LICENSE AUTHORITY FILE COPY

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

January 28, 1981

DO NOT REMOVE

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

Amdt 91
to
DPR-47

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. 91, 91, and 88 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 submittal dated December 29, 1980.

These amendments revise the Technical Specifications by providing for surveillance intervals for certain requirements to be extended from an annual cycle to each reload shutdown, a nominal 18-month cycle.

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

Sincerely,

Robert W. Reid

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

Enclosures:
1. Amendment No. 91 to DPR-38
2. Amendment No. 91 to DPR-47
3. Amendment No. 88 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

cc w/enclosures: See next page

Duke Power Company

cc w/enclosure(s):

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P. O. Box 2178
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Charlotte, North Carolina 28242

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

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

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

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cc w/enclosure(s) & incoming dtd.:
12/29/80

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. 91
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 29, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 91 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
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 29, 1980, 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. 91 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
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 29, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 91 TO DPR-38
AMENDMENT NO. 91 TO DPR-47
AMENDMENT NO. 88 TO DPR-55

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

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 4-1 | 4-1 |
| 4.1-3 | 4.1-3 |
| 4.1-4 | 4.1-4 |
| 4.1-5 | 4.1-5 |
| 4.1-6 | 4.1-6 |
| 4.1-7 | 4.1-7 |
| 4.1-8 | 4.1-8 |
| 4.1-9 | 4.1-9 |
| 4.4-4 | 4.4-4 |
| 4.8-1 | 4.8-1 |

## 4 SURVEILLANCE REQUIREMENTS

### 4.0 SURVEILLANCE STANDARDS

Applicability

Applies to surveillance requirements which relate to tests, calibrations and inspections necessary to assure that the quality of structures, systems and components is maintained and that operation is within the safety limits and limiting conditions for operation.

Objective

To specify minimum acceptable surveillance requirements.

Specification

4.0.1 Surveillance of structures, systems, components and parameters shall be as specified in the various subsections to this Technical Specification section, Section 4.0, except as permitted by Technical Specifications 4.0.2 and 4.0.3 below.

4.0.2 Minimum surveillance frequencies, unless specified otherwise, may be adjusted as follows to facilitate test scheduling:

| Specified Frequency | Maximum Allowable Interval Between Surveillances |
|---|---|
| Five times per week | 2 days |
| Two times per week | 5 days |
| Weekly | 10 days |
| Bi-Weekly | 20 days |
| Monthly | 45 days |
| Bi-Monthly | 90 days |
| Quarterly | 135 days |
| Semiannually | 270 days |
| Annually | 18 months |
| Refueling Outage | 22 months, 15 days |

4.0.3 If conditions exist such that surveillance of an item is not necessary to assure that operation is within the safety limits and limiting conditions for operation, surveillance need not be performed if such conditions continue for a length of time greater than the specified surveillance interval. Surveillance waived as a result of this specification shall be performed prior to returning to conditions for which the surveillance is necessary to assure that operation is within safety limits and limiting conditions for operation.

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

| Channel Description | Check | Test | Calibrate | Remarks |
|---|---|---|---|---|
| 1. Protective Channel Coincidence Logic | NA | MO | NA | |
| 2. Control Rod Drive Trip Breaker | NA | MO | NA | |
| 3. Power Range Amplifier | ES(1) | NA | (1) | (1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent. |
| 4. Power Range | ES | MO | MO(1)(2) | (1) Using incore instrumentation.<br>(2) Axial offset upper and lower chambers after each startup if not done previous week. |
| 5. Intermediate Range | ES(1) | PS | NA | (1) When in service. |
| 6. Source Range | ES(1) | PS | NA | (1) When in service. |
| 7. Reactor Coolant Temperature | ES | MO | RF | |
| 8. High Reactor Coolant Pressure | ES | MO | RF | |
| 9. Low Reactor Coolant Pressure | ES | MO | RF | |
| 10. Flux-Reactor Coolant Flow Comparator | ES | MO | RF | |
| 11. Reactor Coolant Pressure Temperature Comparator | ES | MO | RF | |

Amendments Nos. 91, 91, & 88

4.1-4

Table 4.1-1 (CONTINUED)

| Channel Description | Check | Test | Calibrate | Remarks |
|---|---|---|---|---|
| 12. Pump-Flux Comparator | ES | MO | RF | |
| 13. High Reactor Building Pressure | DA | MO | RF | |
| 14. High Pressure Injection Logic | NA | MO | NA | |
| 15. High Pressure Injection Analog Channels: | | | | |
| a. Reactor Coolant Pressure | ES | MO | RF | |
| b. Reactor Building Pressure (4 psig) | ES | MO | RF | |
| 16. Low Pressure Injection Logic | NA | MO | NA | |
| 17. Low Pressure Injection Analog Channels: | | | | |
| a. Reactor Coolant Pressure | ES | MO | RF | |
| b. Reactor Building Pressure (4 psig) | ES | MO | RF | |
| 18. Reactor Building Emergency Cooling and Isolation System Logic | NA | MO | NA | |
| 19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig) | ES | MO | RF | |

## Table 4.1-1 (CONTINUED)

| Channel Description | Check | Test | Calibrate | Remarks |
|---|---|---|---|---|
| 20. Reactor Building System Logic | NA | MO | NA | |
| 21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure | NA | MO | RF | |
| 22. Pressurizer Temperature | ES | NA | RF | |
| 23. Control Rod Absolute Position | ES(1) | NA | RF(2) | (1) Check with Relative Position Indicator.<br>(2) Calibrate rod misalignment channel. |
| 24. Control Rod Relative Position | ES(1) | NA | RF(2) | (1) Check with Absolute Position Indicator.<br>(2) Calibrate rod misalignment channel. |
| 25. Core Flood Tanks: | | | | |
| a. Pressure | ES | NA | RF | |
| b. Level | ES | NA | RF | |
| 26. Pressurizer Level | ES | NA | RF | |
| 27. Letdown Storage Tank Level | DA | NA | RF | |
| 28. Radiation Monitoring Systems | WE(1) | MO | QU | (1) Check functioning of self-checking feature on each detector. |
| 29. High and Low Pressure Injection Systems Flow Channels | NA | NA | RF | |

Table 4.1-1 (CONTINUED)

| Channel Description | Check | Test | Calibrate | Remarks |
|---|---|---|---|---|
| 30. Borated Water Storage Tank Level Indicator | WE | NA | RF | |
| 31. Boric Acid Mix Tank: | | | | |
| a. Level | NA | NA | AN | |
| b. Temperature | MO | NA | AN | |
| 32. Concentrated Boric Acid Storage Tank: | | | | |
| a. Level | NA | NA | AN | |
| b. Temperature | MO | NA | AN | |
| 33. Containment Temperature | NA | NA | RF | |
| 34. Incore Neutron Detectors | MO(1) | NA | NA | (1) Check functioning; including functioning of computer readout or recorder readout. |
| 35. Emergency Plant Radiation Instruments | MO(1) | NA | RF | (1) Battery check. |
| 36. Environmental Monitors | MO(1) | NA | RF | (1) Check functioning. |
| 37. Reactor Manual Trip | NA | PS | NA | |
| 38. Reactor Building Emergency Sump Level | NA | NA | RF | |
| 39. Steam Generator Water Level | WE | NA | RF | |
| 40. Turbine Overspeed Trip | NA | NA | RF | |

## Table 4.1-1 (CONTINUED)

| Channel Description | Check | Test | Calibrate | Remarks |
|---|---|---|---|---|
| 41. Engineered Safeguards Channel 1 HP Injection Manual Trip | NA | RF | NA | |
| 42. Engineered Safeguards Channel 2 HP Injection Manual Trip | NA | RF | NA | |
| 43. Engineered Safeguards Channel 3 LP Injection Manual Trip | NA | RF | NA | |
| 44. Engineered Safeguards Channel 4 LP Injection Manual Trip | NA | RF | NA | |
| 45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip | NA | RF | NA | |
| 46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip | NA | RF | NA | |
| 47. Engineered Safeguards Channel 7 Spray Manual Trip | NA | RF | NA | |
| 48. Engineered Safeguards Channel 8 Spray Manual Trip | NA | RF | NA | |

## Table 4.1-1 (CONTINUED)

| Channel Description | Check | Test | Calibrate | Remarks |
|---|---|---|---|---|

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly

QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage

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 | Each Refueling (4) |
| 3. Main Steam Safety Valves | Setpoint | Each Refueling (4) |
| 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 | Each Refueling |
| 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.

(4) Number of safety valves to be tested each refueling shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

### 4.4.1.3 Isolation Valve Functional Tests

Quarterly, remotely-operated Reactor Building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during unit operation. The latter valves shall be tested during each refueling shutdown.

### 4.4.1.4 Refueling Outage Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed each refueling outage and prior to any integrated leak rate test, to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak rate test. Results of the inspection shall be reported to the Commission within 90 days of completion.

### 4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.4 and 4.4.1.2.3, respectively.

### Bases

The Reactor Building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 286°F. Prior to initial operation, the containment is strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment is also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests verify that the leak rate from Reactor Building pressurization satisfies the relationships given in the specification.

The performance of a periodic integrated leak rate test during unit life provides a current assessment of potential leakage from the containment, in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 29.5 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leak rate and it duplicates the preoperational leak rate test at 29.5 psig. The specification provides a relationship for relating the measured leakage of air at 29.5 psig to the potential leakage at 59 psig. The frequency of the periodic integrated leak rate test is normally keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leak rate tests is based on three major considerations. First is the low probability of leaks in the

## 4.8 MAIN STEAM STOP VALVES

Applicability

Applies to the main steam stop valves.

Objective

To verify the ability of the main steam stop valves to close upon signal and to verify the leak tightness of the main steam stop valves.

Specification

4.8.1 Using Channels A and B, the operation of each of the main steam stop valves shall be tested during each refueling outage to demonstrate a closure time of one second or less in Channel A and a closure time of 15 seconds or less for Channel B.

4.8.2 The leak rate through the main steam stop valves shall not exceed 25 cubic feet per hour at a pressure of 59 psig and shall be tested during each refueling outage.

Bases

The main steam stop valves limit the Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam line break accident. Their ability to promptly close upon redundant signals will be verified during each refueling outage. Channel A solenoid valves are designed to close all four turbine stop valves in 240 milliseconds. The backup Channel B solenoid valves are designed to close the turbine stop valves in approximately 12 seconds.

Using the maximum 15 second stop valve closing time, the fouled steam generator inventories and the minimum tripped rod worth with the maximum stuck rod worth, an analysis similar to that presented in FSAR Section 14.1.2.9, (but considering a blowdown of both steam generators) shows that the reactor will remain subcritical after reactor trip following a double-ended steam line break.

The main stop valves would become isolation valves in the unlikely event that there should be a rupture of a reactor coolant line concurrent with rupture of the steam generator feedwater header. The allowable leak rate of 25 cubic feet per hour is approximately 25 percent of total allowable containment leakage from all penetrations and isolation valves.

REFERENCES

(1) FSAR Supplement 2, Page 2-7

(2) Technical Specification 4.4.1

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 91 TO FACILITY OPERATING LICENSE NO. DPR-38

## AMENDMENT NO. 91 TO FACILITY OPERATING LICENSE NO. DPR-47

## AMENDMENT NO. 88 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 29, 1980, the Duke Power Company (the licensee or DPC) submitted proposed changes to the Station's common Technical Specifications (TSs) that extend the surveillance interval for certain surveillance requirements from an annual to an 18-month nominal cycle.

### Background

The three Oconee Units are in a transition from a nominal 12-month fuel cycle to a nominal 18-month fuel cycle. The licensee wishes to avoid unnecessary shutdown outages to perform certain surveillances that would be required on an annual basis if the surveillance intervals were not extended.

### Evaluation

The licensee proposed surveillance interval changes in four TSs:

4.0 Surveillance Standards,
4.1 Operational Safety Review,
4.4 Reactor Building, and
4.8 Main Steam Stop Valves.

Each proposal is individually evaluated by TS Section number as follows:

### 4.0 Surveillance Standards

This TS is an overall surveillance requirement for all safety-related equipment and structures. The only proposed change is to add the term "Refueling Outage" and define it as a maximum allowable interval between surveillances of 22 months 15 days. This definition is based on a nominal interval of 18 months plus 25%. This definition is consistent with NUREG-0103, Revision 3, the Standard Technical Specifications (STS) for Babcock and Wilcox (B&W) reactors and is therefore acceptable.

## 4.1 Operational Safety Review

The portions of this TS affected by the December 29, 1980, change request relate to instrument channel calibration for protection system instruments, control system instruments and engineered safety feature instruments, and to the minimum test frequencies for relief valves and safety valves. The proposal is to extend the calibration or test frequencies from annual to each refueling outage which is a nominal 18-month interval plus 25%. The instrument surveillance limits, as proposed, are acceptable as they are consistent with Table 1.2 and Section 4.0.2 of the STS. The proposed test frequencies for the relief and safety valves are acceptable as they are in agreement with Table IWV 3510-1 of Section XI of the ASME Boiler and Pressure Vessel Code.

## 4.4 Reactor Building

The requested change regarding the Reactor Building is to extend the visual examination interval of the accessible interior and exterior surfaces from annual to each refueling interval. This examination is part of the 10 CFR 50 - Appendix J, primary reactor containment leakage testing requirements. Section V.A of Appendix J requires an interval of three examinations within a 10-year period. We find an examination at each refueling interval to be acceptable as it will provide more than three examinations within a 10-year period.

## 4.8 Main Steam Stop Valves

The main steam stop valves become isolation valves in the unlikely event of a concurrent loss-of-coolant accident and a steam generator feedwater header rupture. The closure times of the valves, which determine the rate of reactivity insertion following a main steam line break accident, are not being changed by these license amendments, nor is the currently specified leak rate. The proposed change affects only the closure time test interval; the licensee proposes to extend it from annual to each refueling. Each refueling could result in an interval of 18 months plus 25% or 22 months 15 days. Appendix J of 10 CFR 50, which governs isolation valves, requires a minimum test frequency of two years. We conclude that a test frequency of each refueling is acceptable as it is less than the two-year frequency of Appendix J.

## Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

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: January 28, 1981

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. 91, 91 and 88 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the 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.

These amendments revise the Station's common Technical Specifications by providing for surveillance intervals for certain requirements to be extended from an annual cycle to each reload shutdown, a nominal 18-month cycle.

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

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

For further details with respect to this action, see (1) the application for amendments dated December 29, 1980, (2) Amendments Nos. 91, 91, and 88 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Oconee County Library, 501 West Southbroad, 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 Licensing.

Dated at Bethesda, Maryland, this 28th day of January 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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