

JANUARY 28 1981

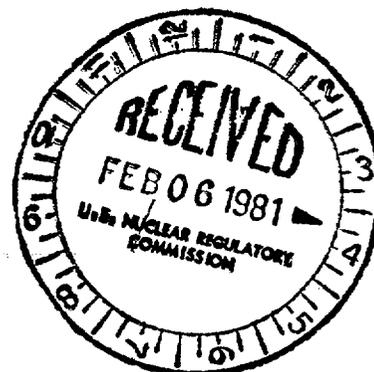
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Dockets Nos. 50-269, 50-270  
and 50-287

Mr. William O. Parker, Jr.  
Vice President - Steam Production  
Duke Power Company  
P. O. Box 2178  
422 South Church Street  
Charlotte, North Carolina 28242



Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 91, 92, 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,

Original signed by  
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. 92 to DPR-47
3. Amendment No. 88 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

cc w/enclosures: See next page

8102200809 P

*E.K.* See comment on FR, notes of p. 2.

OFFICE	ORB#4:DL <i>M</i>	ORB#4:DL <i>MBS</i>	C-ORB#4:DL <i>REid</i>	AD-OR:DL <i>TNovak</i>	OELD		
SURNAME	RIngram/cb	MFairtile	<i>REid</i>	<i>TNovak</i>	<i>Letcher</i>		
DATE	1/19/81	1/21/81	1/21/81	1/21/81	1/23/81		



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

January 28, 1981

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

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Vice President - Steam Production  
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A handwritten signature in cursive script that reads "Robert W. Reid".

Robert W. Reid, Chief  
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5. Notice of Issuance

cc w/enclosures: See next page

Duke Power Company

cc w/enclosure(s):

Mr. William L. Porter  
Duke Power Company  
P. O. Box 2178  
422 South Church Street  
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

U. S. Environmental Protection Agency  
Region IV Office  
ATTN: EIS COORDINATOR  
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Atlanta, Georgia 30308

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U.S. Nuclear Regulatory Commission  
Route 2, Box 610  
Seneca, South Carolina 29678

Mr. Robert B. Borsum  
Babcock & Wilcox  
Nuclear Power Generation Division  
Suite 420, 7735 Old Georgetown Road  
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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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in cursive script, reading "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 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

A handwritten signature in cursive script, appearing to read "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

A handwritten signature in black ink, appearing to read "Robert W. Reid". The signature is written in a cursive style with a large initial "R".

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:

<u>Remove Pages</u>	<u>Insert Pages</u>
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:

<u>Specified Frequency</u>	<u>Maximum Allowable Interval Between Surveillances</u>
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

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
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. (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	

Table 4.1-1 (CONTINUED)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
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)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
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. (2) Calibrate rod misalignment channel.
24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator. (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)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
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)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
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)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
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

<u>Item</u>	<u>Test</u>	<u>Frequency</u>
1. Control Rod Movement <sup>(1)</sup>	Movement of Each Rod	Monthly
2. Pressurizer Safety Valves	Setpoint	Each Refueling <sup>(4)</sup>
3. Main Steam Safety Valves	Setpoint	Each Refueling <sup>(4)</sup>
4. Refueling System Interlocks	Functional	Prior to Refueling
5. Main Steam Stop Valves <sup>(1)</sup>	Movement of Each Stop Valve	Monthly
6. Reactor Coolant System <sup>(2)</sup> Leakage	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 <sup>(3)</sup> Pressure Injection System	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.

8102200817

#### 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 COMMISSIONDOCKETS NOS. 50-269, 50-270 AND 50-287DUKE POWER COMPANYNOTICE 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.

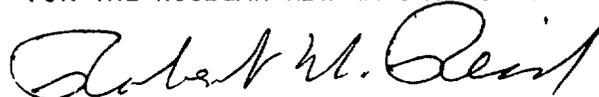
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- 2 -

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 **FEB 05 1981** 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, Chief  
Operating Reactors Branch #4  
Division of Licensing



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

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Docket No. 50-269/270/287

January 28, 1981

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- Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- Notice of Availability of Applicant's Environmental Report.
- Notice of Proposed Issuance of Amendment to Facility Operating License.
- Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- Notice of Availability of NRC Draft/Final Environmental Statement.
- Notice of Limited Work Authorization.
- Notice of Availability of Safety Evaluation Report.
- Notice of Issuance of Construction Permit(s).
- Notice of Issuance of Facility Operating License(s) or Amendment(s).
- Other: Amendments Nos. 91, 91 & 88  
Referenced documents have been provided PDR

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