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

May 5, 1983

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

Posted
Amdt. 121
to DPR-47

Mr. H. B. Tucker
Vice President - Steam 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. 121, 121, and 118to 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 April 18, 1983, as supplemented on April 22, 1983.

These amendments revise the TSs to delay certain surveillance requirements for Oconee Unit No. 1 until its next scheduled refueling outage.

In connection with this action, the Commission is hereby granting an exemption from the requirements of Sections III.D.2 and III.D.3 of Appendix J to 10 CFR 50. This exempts you from the requirements to perform some of the electrical penetration O-ring seal leak tests and certain mechanical penetration leak rate tests prior to the next scheduled refueling outage for Unit No. 1. In granting this exemption, we have determined that it is authorized by law, will not endanger life or property or the common defense and security and is otherwise in the public interest.

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

Sincerely,

for

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. Amendment No. 121 to DPR-38
2. Amendment No. 121 to DPR-47
3. Amendment No. 118 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

Duke Power Company

cc w/enclosure(s):

Mr. William L. Porter
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

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

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

William T. Orders
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

J. Michael McGarry, III, Esq.
DeBevoise & Liberman
1200 17th Street, N.W.
Washington, D. C. 20036

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. 121
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 April 18, 1983, as supplemented April 22, 1983, 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. 121 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1983

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. 121
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 April 18, 1983, as supplemented April 22, 1983, 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. 121 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1983

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. 118
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 April 18, 1983, as supplemented April 22, 1983, 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. 118are 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1983

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 121 TO DPR-38

AMENDMENT NO. 121 TO DPR-47

AMENDMENT NO. 118 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
4.1-3	4.1-3
4.1-4	4.1-4
4.1-5	4.1-5
4.1-6	4.1-6
4.1-8	4.1-8
4.4-6	4.4-6
4.4-9	4.4-9
4.4-10	4.4-10
4.4-11	4.4-11
4.4-13	4.4-13
4.6-1	4.6-1
4.18-1	4.18-1

Pages 4.4-5 and 4.4-14 are overleaf pages and are included to maintain document completeness.

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. (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)

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 & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
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 (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
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. (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)

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 o 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
49. Emergency Feedwater Flow Indicators	MO	NA	RF	
50. PORV and Safety Valve Position Indicators	MO	NA	RF	
51. RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	MO	RF	
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
a) Control Oil Pressure Switches	NA	MO	RF	
*b) Discharge Pressure Switches	NA	MO	RF	
53. Emergency Feedwater Initiation Circuits				
a) Control Oil Pressure Switches	NA	MO	RF	
*b) Discharge Pressure Switches	NA	MO	RF	

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

* This Technical Specification will become effective as follows:
Unit 1 - at the first convenient outage prior to or at the end of Oconee 1 Cycle 8 Refueling Outage
Unit 2 - end of Oconee 2 Cycle 6 Refueling Outage
Unit 3 - end of Oconee 3 Cycle 7 Refueling Outage

During the interim period, these discharge pressure switches will be tested during cold shutdown not to exceed once per month.

**A one-time extension is granted for the instrument calibration such that it be performed during the 1983 Unit 1 refueling outage, provided that such outage begins no later than July 16, 1983.

When containment integrity is established, the overall containment leak rate of 0.25 weight percent of containment air at 59 psig will assure that the limits of 10CFR100 will not be exceeded should the maximum hypothetical accident occur. In order to assure the integrity of the containment, periodic testing is performed at reduced pressure, 29.5 psig. The permissible leakage rate at this reduced pressure has been established from the initial integrated leak rate tests in conformance with 10CFR50, Appendix J.

The containment air locks (i.e., Personnel Hatch and Emergency Hatch) are tested on a more frequent basis than other penetrations. The air locks are utilized during periods of time when containment integrity is required as well as when the reactor is shutdown. Proper verification of door seal integrity is required to ensure containment integrity. Because the door seals are recessed, damage from tools due to air lock entry is improbable; however, a leak test of the outer door seals has been shown to be an acceptable alternative to the full hatch test to ensure air lock integrity.

REFERENCES

(1) FSAR, Sections 5 and 13.

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
1	Pressurizer liquid sample line (Unit 1 only)	Note 1	Type C	Note 2, 7b
2	OTSG A Sample line	Note 1	Type C	Note 7b
3	Component cooling inlet line	Note 1	Type C	Note 3, 7d
4	OTSG B drain line	Note 1	None required	Note 7b
5	RB normal sump drain line	Note 10	Type C	Note 7a, 7b, 9
6	Letdown line	Note 1	Type C	Note 2, 7b
7	RC Pump seal return line	Note 1	Type C	Note 3, 7b, 9
8	Loop A nozzle warming line	Not Vented	None required	Note 5, 7d
9	RCS normal makeup line and HP injection 'A' loop	Not Vented	None required	Note 5
10	RC Pump seal injection	Not Vented	Type C	Note 5, 7d, 9, 12

TABLE 4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
36 37	RB emergency sump recirculation line	Not Vented	None required	Note 5
38	Quench tank cooler inlet line	Note 1	Type C	Note 2, 7d, 12
39	HP Nitrogen supply	Note 1	None required	Note 3 (manual valves)
(Unit 2, 3) Only	CFT Vent line	Note 1	None required	Note 3 (manual valves)
40	RB emergency sump drain line	Note 1	None required	
41	Instrument air supply & ILRT verification line	Note 1	None required	Note 3 (manual valves)
42	SPARE	Not in Use		
43	OTSG A drain line	Note 1	None required	Note 7b
44	Component cooling to control rod drive inlet line	Note 1	Type C	Note 3, 7d
45	ILRT instrument line	Not Vented	Type C	Note 3, 7a
46	Reactor head-wash filtered water inlet	Note 1	Type C	Note 3 (manual valves)

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
47 (Unit 1 only)	Demineralized water supply to RC pump seal vents	Note 1	Type C	Note 3, 7d
48	Breathing air inlet	Note 1	None required	Note 3 (manual valves)
49 (Unit 1 only)	LP Nitrogen supply	Note 1	None required	Note 3 (manual valves)
50	OTSG A Emergency FDW line	Not Vented	None required	Note 5
51	ILRT Pressurization line	Note 1	None required	Note 6a, 7a
52	HP Injection to 'B' loop	Not Vented	None required	Note 5
53 (All)	HP Nitrogen supply to 'A' core flood tank	Note 1	None required	Note 3 (manual valves)
(Unit 2, 3)	LP Nitrogen supply	Note 2	None required	Note 3 (manual valves)
54	Component cooling outlet line	Note 1	Type C	Note 3, 7b, 9(8)
55	Demineralized water supply	Note 1	Type C	(Unit 1) Note 3, (manual valves), 12 (Unit 2,3) Note 3, 9 (manual valves)
56	Spent fuel canal fill and drain	Note 1	None required	Note 3 (manual valve)
57 (Unit 1 only)	DHR return line	Not Vented	None required	Note 4

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
58 (All)	OTSG B sample line	Note 1	Type C	Note 7b
(Unit 2, 3)	Pressurizer sample line	Note 1	Type C	Note 2, 7b
59	CF tank sample line	Note 1	None required	Note 2
60	RB sample line (outlet)	Note 1	Type C	Note 2, 7b, 9
61	RB sample line (inlet)	Note 1	Type C	Note 3, 7b, 9
62 (Units 2, 3 only)	DHR return line	Not vented	None required	Note 4
	Personnel hatch	Vented	Type B	Note 6b
	Emergency hatch	Vented	Type B	Note 6b
	Equipment hatch	Vented	Type B	Note 6c
	Electrical penetration	Vented	Type B	Note 6a, 12

TABLE 4.4-1
NOTES (continued)

c. Isolation valves are required to operate intermittently under post accident conditions.

d. Check valves used for containment isolation.

NOTE 8 DELETED

NOTE 9 Reverse direction test of inside containment isolation valve authorized. Leakage results are conservative.

NOTE 10 System is submerged during post-accident conditions and performance of Type A test. System will be drained to the extent possible.

NOTE 11 Type B test performed on the blind flanges inside the Reactor Building. The tube drain valves and valves outside the containment are not tested.

NOTE 12 A one-time extension from the local leak test and corresponding exemption from Sections III.D.2 and III.D.3 of Appendix J to 10 CFR Part 50 is granted such that it be performed during the 1983 Unit 1 refueling outage, provided that such outage begins no later than July 16, 1983.

4.4.2 Structural Integrity

Applicability

Applies to the structural integrity of the Reactor Building.

Objective

To define the inservice surveillance program for the Reactor Building.

Specification

4.4.2.1 Tendon Surveillance

For the initial surveillance program, covering the first five years of operation, nine tendons shall be selected for periodic inspection for symptoms of material deterioration or force reduction. The surveillance tendons shall consist of three horizontal tendons, one in each of three 120° sectors of the containment; three vertical tendons located at approximately 120° apart; and three dome tendons located approximately 120° apart. The following nine tendons have been selected as the surveillance tendons:

Dome	1D28 2D28 (Units 1 & 3) 2D29 (Unit 2) 3D28
Horizontal	13H9 51H9 53H10
Vertical	23V14 45V16 61V16

4.4.2.1.1 Lift-Off

Lift-off readings shall be taken for all nine surveillance tendons.

4.4.2.1.2 Wire Inspection and Testing

One surveillance tendon of each directional group shall be relaxed and one wire from each relaxed tendon shall be removed as a sample and visually inspected for corrosion or pitting. Tensile tests shall also be performed on a minimum of three specimens taken from the ends and middle of each of the three wires. The specimens shall be the maximum length acceptable for the test apparatus to be used and shall include areas representative of significant corrosion or pitting.

After the wire removal, the tendons shall be retensioned to the stress level measured at the lift-off reading and then checked by a final lift-off reading.

4.6 EMERGENCY POWER PERIODIC TESTING

Applicability

Applies to the periodic testing surveillance of the emergency power sources.

Objective

To verify that the emergency power sources and equipment will respond promptly and properly when required.

Specification

4.6.1 Monthly, a test of the Keowee Hydro units shall be performed to verify proper operation of these emergency power sources and associated equipment. This test shall assure that:

a. Each hydro unit can be automatically started from the Unit 1 and 2 control room.

b. Each hydro unit can be synchronized through the 230 kV overhead circuit to the startup transformers.

c. Each hydro unit can energize the 13.8 kV underground feeder.

d. The 4160 volt startup transformer main feeder bus breakers and standby bus breaker shall be exercised.

4.6.2 a. Annually, the Keowee Hydro units will be started using the emergency start circuits in each control room to verify that each hydro unit and associated equipment is available to carry load within 25 seconds of a simulated requirement for engineered safety features.

b. Promptly following the above annual test, each hydro unit will be loaded to at least the combined load of the auxiliaries actuated by ESG signal in one unit and the auxiliaries of the other two units in hot shutdown by synchronizing the hydro unit to the offsite power system and assuming the load at the maximum practical rate.

4.6.3 Monthly, the Keowee Underground Feeder Breaker Interlock shall be verified to be operable.

4.6.4 During each refueling outage*, a simulated emergency transfer of the 4160 volt main feeder buses to the startup transformer (i.e., CT1, CT2 or CT3) and to the 4160 volt standby buses shall be made to verify proper operation.

4.6.5 Quarterly, the External Grid Trouble Protection System logic shall be tested to demonstrate its ability to provide an isolated power path between Keowee and Oconee.

4.6.6 Annually and prior to planned extended Keowee outages, it shall be demonstrated that a Lee Station combustion turbine can be started and

*A one-time extension is granted for the simulated emergency transfer of the 4160 volt main feeder buses such that it be performed during the 1983 Unit 1 refueling outage, provided that such outage begins no later than July 16, 1983.

4.18 SNUBBERS

Applicability

Applies to hydraulic and mechanical snubbers used to protect the Reactor Coolant System and other safety-related systems.

Objective

To verify that the required hydraulic and mechanical snubbers are operable.

Specification

4.18.1 Each snubber associated with the Reactor Coolant System and other safety-related systems, as specified in the appropriate Station Procedure shall be visually inspected. Visual inspections shall verify:

(1) that there are no visible indications of damage or impaired OPERABILITY,

(2) attachments to the foundation or supporting structure are secure, and

(3) in those locations where mechanical snubber movement can be manually induced, the snubbers shall be inspected as follows:

(a) At each refueling*, the inaccessible snubbers shall be inspected near the beginning and the end of the outage.

(b) In the event of a severe dynamic event, snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom of motion using one of the following: (i) Manually induced snubber movement, (ii) evaluation of in place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced (or overhauled) before returning to power. Re-inspection shall subsequently be performed according to the schedule listed below.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.18.4. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be tested

*A one-time extension is granted for the inaccessible mechanical snubber inspection such that it be performed during the 1983 Unit 1 refueling outage, provided that such outage begins no later than July 16, 1983.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. DPR-55

AND REQUEST FOR EXEMPTION FROM SECTIONS III.D.2 AND 3

OF APPENDIX J TO 10 CFR PART 50

DUKE POWER COMPANY

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

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

Introduction

By application dated April 18, 1983, as supplemented on April 22, 1983, Duke Power Company (DPC or the licensee) proposed changes to the Oconee Nuclear Station, Unit No. 1 (ONS-1 or the facility) Technical Specifications (TSs). The proposed TS changes would delay certain surveillance requirements until the unit's upcoming refueling outage.

Background

The last ONS-1 refueling outage lasted approximately six months, from June 26 until December 31, 1981. The extended outage was to determine the broken-bolt failure mechanism, engineer a fix and carry out the required modifications to the lower thermal shield. Many equipment tests, calibrations and inspections required by the TSs were performed early in this refueling outage. Because of the long outage and the cycle operating history, some TS surveillance requirements are coming due ahead of the scheduled July 3, 1983 refueling outage. DPC has pointed out that the July 3 shutdown date assumes continuous operation and could be as late as 11:59pm on July 15. Therefore, their request is for one-time TS changes to delay 25 surveillance requirements until July 16, 1983. Each area of the proposed TS changes is evaluated below.

Evaluation

- Instrument Channel Calibrations

  Thirteen different types of instrument channels were calibrated between June 26 and August 10, 1981, in accordance with TS 4.1.1 and Table 4.1-1. TS 4.0.2 gives the maximum allowable interval between surveillance as 22 months 15 days. Therefore, these calibrations are due to be performed between May 11 and June 25, 1983.

DPC has analyzed the instrument channels involved for drift. The channels are subject to only "drift" errors induced within the instrumentation itself and can tolerate long intervals between calibrations. The process system instrumentation errors induced by drift are expected to remain within acceptable FSAR limits until recalibration can be performed during the upcoming refueling or the next forced outage of sufficient length. Various channels involved in the request for relief are one of a series of redundant channels in a given system. The request for relief involves only the formal surveillance requirement and not the additional checks that the channels receive during shift or monthly inspections. Substantial calibration shifts within the channel (essential channel failure) is revealed during routine checking and testing procedures.

Additionally, the results of the previous calibration were investigated and the acceptance criteria were met. In two cases (RPS Channel D Flow Instrument and LPIS Sump Level Instrument) the acceptance criteria were not met for the previous calibration. In the case of the Channel D Flow Instrument, however, the system uses total flow in the RPS calculations and this met the acceptance criteria. In the case of the Sump Level Instrument the drift, although outside of acceptance criteria, was in the conservative direction. This instrument measures leakage from the reactor coolant system and is backed up by redundant systems (i.e., the reactor building air particulate monitor, iodine monitors, gaseous area monitors and water inventory balances).

Based on our review of this data, we find the additional 73 days (maximum possible) of reactor operation acceptable from an instrumentation reliability standpoint.

- Local Leak Rate Testing

TS 4.4.1.2.2 requires local leak rate testing of all containment penetrations at intervals no greater than 24 months. This requirement is in agreement with Sections III.D.2 and 3 of Appendix J to 10 CFR Part 50 (Schedule for Type B and Type C tests). The last electrical penetration O-ring seal leak test and the mechanical penetration leak rate test for Unit 1 began June 30, 1981 and July 4, 1981, respectively. The detailed data show only three of the 61 electrical penetrations and three mechanical penetrations were tested before July 16, 1981 and, therefore, are due before the latest proposed shutdown date, 11:59pm on July 15, 1983. The justification to delay surveillance for each is as follows:

- Electrical Penetration O-ring Seal Test

An operating experience review by DPC has shown that failure of O-ring seals at the plant are relatively low. Of the more than a thousand leak rate tests performed to date, only two failures of O-ring seals have been found.

All electrical penetrations are grouped within or vented to the penetration room which is formed adjacent to the outside surface of each Reactor Building by enclosing the area around the majority of the penetrations. Each penetration room is provided with two fans and two filter assemblies. When the filtration system is placed into operation (upon receipt of an engineered safeguards signal from the Reactor Building), a negative pressure will be maintained in the penetration room to assure inleakage which is collected and discharged through HEPA and charcoal filters to the unit vent.

We, therefore, find reactor operation without performing the TS required Type B test (Appendix J of 10 CFR 50, Section III.B) acceptable based on previous reliability, system design and the short period of time (less than 16 days beyond the scheduled due date). Additionally, justification has been provided for the schedular relief from the 24 month test interval specified in Appendix J.

- Mechanical Penetration Leak Rate Test

There are only three penetrations affected by the request for relief. Penetration 10 is tied to the HPI system outside of containment which is a closed system and seismic designed. Penetration 38 (Quench Tank Cooler Inlet Line) is also a closed system and non-seismic outside the containment isolation valve. Penetration 55 (Demineralized Water Supply) is a closed system and non-seismic outside the containment isolation valve. Additional justification provided by the licensee is the mechanical penetration design and the negative pressure features of the penetration room mentioned above. All mechanical penetrations are grouped within or vented to the penetration room. Any leakage that might occur from these penetrations will be collected and discharged through HEPA filters and charcoal filters to the unit vent. The leakage barrier in the Reactor Building is the one-quarter inch steel liner plate. All penetrations are continously welded to the liner plate before they are embedded in concrete. The penetrations become an integral part of the liner and are designed, installed, and tested as such. Additionally, the steel liner plate is attached to the prestressed concrete Reactor Building and forms an integral part of the structure. The Reactor Building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbable hypothetical accident as well as for other types of loading conditions.

We find continued reactor operation for less than 12 days beyond the test schedule acceptable. Likewise, relief from the Type C test of Appendix J to 10 CFR, Section III.G is justified and should be granted.

- Engineered Safety Features (ESF) Valves

TS 4.5.1.2.2.b requires critical low pressure injection (LPI) valves to be cycled manually each refueling outage. DPC states that valves LP-9, 10, 12, 14, 17 and 18 of the LPI and core flooding systems were last cycled beginning July 21, 1981. Therefore, the retest is due starting June 6, 1983, about 50 days before the latest shutdown date.

Justification provided is satisfactory stroke test results of these valves in the past.

We find the 50-day delay in testing the operability of the LPI and core flooding systems' valves, as listed above, acceptable.

- Emergency Power Circuitry

TS 4.6.4 requires a simulated emergency transfer of the 4160 volt main bases to the startup transformer and to the standby buses during each refueling outage. The last such transfer was performed August 21, 1981, thus it is due by TS on July 6, 1983. The licensee's justification for the needed extension of nine days is no previous operational problems with these breakers.

We find delaying this simulated emergency transfer for nine days maximum will not adversely affect reactor safety and is, therefore, acceptable.

- Mechanical Snubbers

DPC states that five inaccessible mechanical snubbers on the steam generator flush and drain line were previously inspected on July 12, 1981, per TS 4.18.1. Thus, the inspection on these snubbers is due May 28, 1983, 49 days before the latest possible shutdown date. Justification for this inspection delay supplied by DPC is the relative low stress conditions that would exist if some of the snubbers were inoperable, a new snubber installed last outage and one snubber on non-safety grade piping.

We find the proposed TS change delaying inspection of the subject mechanical snubbers for 49 days acceptable.

Conclusion

We conclude that there is adequate justification for the short-time postponement of the TS surveillance of the equipment described above. Some of the TSs should be modified to meet our requirements such as limiting the exemption coverage. This has been discussed with the DPC staff and agreement reached.

We further conclude that an exemption to the schedular requirements of Sections III.D.2 and 3 of Appendix J to 10 CFR Part 50 is justified.

Environmental Consideration

We have determined that this action does 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 this action involves 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 this action.

Safety Conclusions

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 an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, 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.

We have also determined, pursuant to 10 CFR 50.12, that an exemption from the requirements of Sections III.D.2 and 3 of Appendix J to 10 CFR 50 is authorized by law, will not endanger life or property or the common defense and security and is otherwise in the public interest.

Dated: May 5, 1983

The following NRC personnel have contributed to this Safety Evaluation: E. Conner, and John F. Suermann.

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

AND EXEMPTION FROM APPENDIX J, 10 CFR 50

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 121, 121 and 118 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company (the licensee), which revised the Technical Specifications (TSs) 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 TSs to delay certain surveillance requirements for Oconee Unit No. 1 until its next scheduled refueling outage. In connection with this action, the Commission has also granted an exemption from the requirements of Sections III.D.2 and III.D.3 of Appendix J to 10 CFR 50, to perform leak rate tests on certain of the O-ring seal electrical and the mechanical penetrations prior to the next scheduled refueling outage for Oconee Unit No. 1. The exemption is effective as of the date of issuance.

The application for the amendments and request for exemption comply 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 and the letter to the licensee dated May 5, 1983 . 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 and the granting of this exemption will not result in any significant environemntal impact and that pursuant to 10 CFR §51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these actions.

For further details with respect to these actions, see (1) the application for amendments and request for exemption dated April 18, 1983, as supplemented April 22, 1983, (2) Amendments Nos.121 ,121 , and 118 to License Nos. DPR-38, DPR-47 and DPR-55, respectively, (3) the Commission's related Safety Evaluation and (4) the Commission's letter to the Licensee dated May 5, 1983 . 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 Street, Walhalla, South Carolina 29691. A copy of items (2), (3) and (4) 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 5th day of May 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing