February 26, 1998

Mr. William R. McCollum
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

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SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. MA0736, MA0737, AND MA0738)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 228 ,229 , and 225 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications in response to your application dated February 2, 1998, and supplement dated February 18, 1998.

The amendments revise the wording used to specify refueling outage surveillances. The changes clarify that these surveillances are to be performed on an 18-month frequency and need not be constrained to refueling outage conditions. This action supersedes the Notice of Enforcement Discretion Number 98-06-001 issued by the staff on February 2, 1998.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:
David E. LaBarge, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No.228to DPR-38
2. Amendment No.229to DPR-47
3. Amendment No.225to DPR-55
4. Safety Evaluation

1/1 OFO1

cc w/encl: See next page

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OFFICE	PM:PD22	E	LA:PD22		SPLB*		TSB*		OGC
NAME	D.LaBARGE		L.BERRY		L.MARSH		W.BECKNER		UYoung
DATE	2/25/98		2/26/98		2/ 10 /98		2/ 11 /98		2/25/98

OFFICE	PDII-2/D		SCSB*					
NAME	HBerkow		CBerlinger					
DATE	2/26/98		2/ 10 /98		/ /98		/ /98	

*See previous concurrence **OFFICIAL RECORD COPY**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 26, 1998

Mr. William R. McCollum
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
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Docket Nos. 50-269, 50-270, and 50-287

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1. Amendment No. 228 to DPR-38
2. Amendment No. 229 to DPR-47
3. Amendment No. 225 to DPR-55
4. Safety Evaluation

cc w/encl: See next page

Oconee Nuclear Station

cc:

Mr. Paul R. Newton
Legal Department (PBO5E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Mr. Robert B. Borsum
Framatome Technologies
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852-1631

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
7812B Rochester Highway
Seneca, South Carolina 29672

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, S.W., Suite 23T85
Atlanta, Georgia 30303

Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. J. E. Burchfield
Compliance Manager
Duke Energy Corporation
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

L. A. Keller
Manager - Nuclear Regulatory Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28242-0001

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of Environment, Health, and Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 228
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Energy Corporation (the licensee) dated February 2, 1998, as supplemented February 18, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 228 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and will be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: February 26, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.229
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Energy Corporation (the licensee) dated February 2, 1998, as supplemented February 18, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 229 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and will be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: February 26, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 225
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Energy Corporation (the licensee) dated February 2, 1998, as supplemented February 18, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 225 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and will be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: February 26, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 228

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 229

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 225

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

Remove	Insert
4.0-1	4.0-1
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
4.1-8a	4.1-8a
-------	4.1-8b
4.1-9	4.1-9
4.2-2	4.2-2
4.4-17	4.4-17
4.4-20	4.4-20
4.5-1	4.5-1
4.5-2	4.5-2

Remove	Insert
4.5-4	4.5-4
4.5-6	4.5-6
4.5-7	4.5-7
4.5-8	4.5-8
4.5-9	4.5-9
4.6-1	4.6-1
4.7-1	4.7-1
4.8-1	4.8-1
4.9-1	4.9-1
4.12-1	4.12-1
4.14-1	4.14-1
4-14-2	4.14-2
4.18-1	4.18-1
4.18-2	4.18-2
4.18-4	4.18-4
4.20-5	4.20-5

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
18 months	22 months, 15 days
Refueling Outage	22 months, 15 days

Clarifying words in individual specifications such as "every," "at least," or "at least once every" are not intended to alter the frequencies defined by this specification.

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.

4.0.4 Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(l)	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	45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide 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	45 Days STB	18 months	
8. High Reactor Coolant Pressure	ES	45 Days STB	18 months	
9. Low Reactor Coolant Pressure	ES	45 Days STB	18 months	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
10. Flux-Reactor Coolant Flow Comparator	ES	45 Days STB	18 months	
11. Reactor Coolant Pressure Temperature Comparator	ES	45 Days STB	18 months	
12. Pump-Flux Comparator	ES	45 Days STB	18 months	
13. High Reactor Building Pressure	DA	45 Days STB	18 months	
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	18 months	
b. Reactor Building Pressure (4 psig)	ES	MO	18 months	
16. Low Pressure Injection Logic	NA	MO	NA	

TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	18 months	
b. Reactor Building Pressure (4 psig)	ES	MO	18 months	
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	18 months	
20. Reactor Building Spray System Logic	NA	MO	NA	
21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	18 months	
22. Pressurizer Temperature	ES	NA	18 months	
23. Control Rod Absolute Position	ES(1)	NA	18 months (2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.

Channel Description	Check	Test	Calibrate	Remarks
24. Control Rod Relative Position	ES(1)	NA	18 months (2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel.
25. Core Flood Tanks				
a. Pressure	ES	NA	18 months	
b. Level	ES	NA	18 months	
26. Pressurizer Level	ES	NA	18 months	
27. Letdown Storage Tank Level	DA	NA	18 months	
28. Delete				
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	18 months	
30. Borated Water Storage Tank Level Indicator	WE	NA	18 months	
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	

Table 4.1-1 (CONTINUED)

Channel Description		Check	Test	Calibrate	Remarks
33.	Containment Temperature	NA	NA	18 months	
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	18 months	(1) Battery check.
36.	Environmental Monitors	MO(1)	NA	18 months	(1) Check functioning.
37.	Reactor Manual Trip	NA	PS	NA	
38.	Reactor Building Emergency Sump Level	NA	NA	18 months	
39.	Steam Generator Water Level	WE	NA	18 months	
40.	Turbine Overspeed Trip	NA	NA	18 months	
41.	Engineered Safeguards Channel 1 HP Injection & Reactor Building Isolation Manual Trip	NA	18 months	NA	Includes Reactor Building isolation of non-essential systems only
42.	Engineered Safeguards Channel 2 HP Injection & Reactor Building Isolation Manual Trip	NA	18 months	NA	Includes Reactor Building isolation of non-essential systems only
43.	Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	18 months	NA	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
44. Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	18 months	NA	
45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	18 months(1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	18 months(1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
47. Engineered Safeguards Channel 7 Spray Manual Trip	NA	18 months	NA	
48. Engineered Safeguards Channel 8 Spray Manual Trip	NA	18 months	NA	
49. Emergency Feedwater Flow Indicators	MO	NA	18 months	
50. PORV and Safety Valve Position Indicators	MO	NA	18 months	
51. RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	18 months	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
a) Control Oil Pressure Switches	NA	45 Days STB	18 months	
53. Emergency Feedwater Initiation Circuits				
a) Control Oil Pressure Switches	NA	MO	18 months	
54. Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	18 months	TMI Item II.F.1.3
55. Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	18 months	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80,-81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
59. Reactor Vessel Head Level	NA	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
60. Core Exit Thermocouples	MO	NA	18 months(1)	(1) A one-time extension of the calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
61. Subcooling Monitors	MO	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.

ES - Each Shift	QU - Quarterly
DA - Daily	AN - Annually
WE - Weekly	PS - Prior to startup, if not performed previous week
MO - Monthly	NA - Not Applicable
	STB - STAGGERED TEST BASIS

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	18 months[(4)]
3.	Main Steam Safety Valves	Setpoint	18 months[(4)]
4.	Refueling System Interlocks[(5)]	Functional	Prior to Refueling
5.	Main Steam Stop Valves [(1)]	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System [(2)] Leakage	Evaluate	Daily
7.	Condenser Circulating Water[(6)] Flow Test	Functional	18 months
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 [(3)] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months

(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 every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

(5) Applicable only to the interlocks associated with the Reactor Building Purge System.

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

4.2.6 The power operated relief valve (PORV) is used for low temperature overpressure protection of the RCS and shall be demonstrated operable by:

a. Performing an operability test prior to each startup from cold shutdown.

b. Performing a calibration of the actuation circuit every 18 months.

c. Performing an inspection of the PORV at least once every two refueling cycles.

4.2.7 Each shift, the RCS vent(s) (as defined in Specification 3.1.2.9) shall be verified to be open, if the vent(s) is(are) being used for overpressure protection. If the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then these valves will open at least once per 31 days.

Bases

The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10 CFR 50.55(a) to the extent practicable within limitations of design, geometry and materials of construction. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

4.4.3 Containment Hydrogen Control Systems

Applicability

Applies to the Containment Hydrogen Control Systems.

Objective

To verify that the Containment Hydrogen Control Systems are operable.

Specifications

4.4.3.1 Containment Hydrogen Control System Piping

Every 18 months, the permanent piping for the Containment Hydrogen Control System shall be tested as follows:

a. The post-LOCA flow paths shall be verified by connecting and operating either the Hydrogen Purge Unit or the Hydrogen Recombiner through each flow path as follows:

   1. The hydrogen Recombiner flow path circulates Reactor Building atmosphere at a flow greater than 50 SCFM.

   2. The Hydrogen Purge flow path removes Reactor Building atmosphere and discharges to the Unit vent stack at a flow greater than or equal to 45 SCFM.

b. The blind isolation flanges on the Containment Hydrogen Control System permanent piping shall be leak tested after each installation to ensure adequate isolation.

4.4.3.2 Containment Hydrogen Recombiner System Operational Performance Testing

a. The testing requirement of this section may be performed without connecting the system to either of the Reactor Buildings.

b. Every 18 months:

   1. Visual inspection of the unit.

   2. Calibrate all recombiner instrumentation and control circuits.

   3. Operate a recombiner unit at design flow rate 10% and allow unit to reach recombination temperature.

4.4.3.3 Reactor Building Hydrogen Purge System, Pre-Operational Testing

a. Prior to declaring this system operable, a Pre-operational system test shall be performed.

4.4.4 Reactor Building Purge System

Applicability

Applies to the Reactor Building Purge System.

Objective

To verify that the Reactor Building Purge System is operable.

Specification

4.4.4.1 Each shutdown, when the purge valves have been operated, leakage integrity tests shall be performed on the containment purge isolation valves after final closing and prior to going above hot shutdown. If the purge valves have not been operated, leakage integrity tests shall be performed prior to going above hot shutdown unless such tests have been conducted within the proceeding six months. If the acceptance criteria of Specification 4.4.1.2.3 are not met, Specification 3.6.6 shall apply. Unit shutdown to conduct the test and/or effect repairs is specifically not required.

4.4.4.2 Monthly, when the unit is above 250°F and 350 psig, the containment purge isolation valves shall be verified closed.

4.4.4.3 Every 18 months, the valve seals of the containment purge isolation valves shall be visually inspected and adjusted or replaced as appropriate.

4.4.4.4 Prior to use of the purge system at conditions between cold shutdown and 250°F and 350 psig, the isolation valves shall be exercise tested in accordance with the requirements (except test frequency) of the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

4.4.4.5 The pneumatically operated purge isolation valves shall be verified to close in response to a control signal from RIA-45 when the system is tested prior to refueling operations per Specification 3.8.10.

Bases

Leakage integrity tests of the purge supply and isolation valves are conducted in order to identify excessive degradation of the resilient seals. Excessive leakage past resilient seals is typically caused by severe environmental conditions and/or wear due to frequent use.

The pneumatically operated purge isolation valves are tested prior to refueling operations because the only automatic isolation system in service at refueling is through RIA-45, which only closes the pneumatic isolation valves.

4.5 EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirements for the Emergency Core Cooling Systems.

Objective

To verify that the Emergency Core Cooling Systems are operable.

Specification

4.5.1.1 System Tests

4.5.1.1.1 High Pressure Injection System

a. Every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the High Pressure Injection System for emergency core cooling operation.

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

4.5.1.1.2 Low Pressure Injection System

a. Every 18 months, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

   (1) A test signal will be applied to demonstrate actuation of the Low Pressure Injection System for emergency core cooling operation.

   (2) Verification of the engineered safety features function of the Low Pressure Service Water pumps and manual alignment from the control room of valves LPSW-4 and LPSW-5 shall be made to demonstrate operability of the Low Pressure Injection coolers.[1]

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the ES actuation signal properly; all appropriate ES actuated pump breakers shall have opened or closed, and all ES actuated valves shall have completed their travel. In addition, valves LPSW-4 and LPSW-5 shall have completed their travel.

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[1] The ES function of valves LPSW-4 and LPSW-5 shall be verified every 18 months. This surveillance requirement may be discontinued and replaced by the valve surveillance in 4.5.1.1.2.a.(2) when the ES signals are removed from LPSW-4 and LPSW-5. Removal of the ES signal from valves LPSW-4 and LPSW-5 is scheduled in the U3EOC16, U1EOC17, and U2EOC16 refueling outages successively.

4.5.1.1.3 Core Flooding System

a. Every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests

4.5.1.2.1 Valves - Power Operated

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

b. Every 18 months, the following LPI system valves shall be cycled manually to verify the manual operability of these power operated valves:

   (1) LPI pump discharge (ES) LP-17,-18
   (2) LPI discharge throttling LP-12,-14
   (3) LPI discharge header crossover LP-9,-10
   (4) LPI discharge to HPI/RBS LP-15,-16

4.5.1.2.2 Check Valves

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10.

Bases

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

(a)
To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the Reactor Building Cooling Systems.

Objective

To verify that the Reactor Building Cooling Systems are operable.

Specification

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

a. (1) Every 18 months, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the Reactor Building Spray System.

(2) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly; the appropriate pump breakers shall have closed, and all valves shall have completed their travel.

b. Station compressed air will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every ten years.

4.5.2.1.2 Reactor Building Cooling System

a. Every 18 months[1], a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the Reactor Building Cooling System for reactor building cooling operation.

(2) Verification of the engineered safety features function of the Low Pressure Service Water System which supplies coolant to the reactor building coolers shall be made to demonstrate operability of the coolers.

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate valves have completed their travel, and fans are running at half speed.

[1]A one-time extension of the Reactor Building Cooling system test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.

4.5.3 Containment Heat Removal Capability

Applicability

Applies to verification of adequate containment heat removal capability.

Objective

To verify that containment heat removal capability is sufficient to maintain post accident conditions within design limits.

Specification

4.5.3.1 Containment Heat Removal Capability

a. Every 18 months, containment heat removal capability shall be verified to be sufficient to maintain post accident conditions within design limits.

b. In addition to the requirements of 4.5.3.1.a, on a frequency consistent with the LPI cooler and RBCU fouling rate, containment heat removal capability shall be verified to be sufficient to maintain post accident conditions within design limits.

Bases

The safety functions of the LPI system, RB Spray system, and RBCUs include maintaining containment pressure and temperature below design limits following an accident. This surveillance assures that containment heat removal capability is adequate assuming a worst case single failure. Specification 4.5.3.1.a requires that at a minimum the surveillance be performed every 18 months. In addition, since service induced fouling can reduce containment heat removal capability, Specification 4.5.3.1.b requires that a fouling rate be determined in order to establish a more frequent test interval if required.

REFERENCES:

FSAR Section 6.2
FSAR Section 15.14

4.5.4 Penetration Room Ventilation System

Applicability

Applies to testing of the Penetration Room Ventilation System

Objective

To verify that the Penetration Room Ventilation System is operable.

Specification

4.5.4.1 Operational and Performance Testing

a. Monthly, each train of the Penetration Room Ventilation System shall be operated for at least 15 minutes at design flow ±10%.

b. Every 18 months, it shall be demonstrated that:

   1. The Penetration Room Ventilation System fans operate at design flow (± 10%) when tested in accordance with ANSI N510-1975.

   2. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than six inches of water at the system design flow rate (± 10%).

   3. Each branch of the Penetration Room Ventilation System is capable of automatic initiation.

   4. The bypass valve for filter cooling is manually operable.

c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Penetration Room purge filters:

   1. Every 18 months;

   2. After each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank;

   3. After any structural maintenance on the system housing;

   4. After painting, fire, or chemical release in any ventilation zone communicating with the system.

d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

e. Every 18 months, or following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Penetration Room Ventilation system filters for laboratory analysis. Within 31 days of removal, this sample shall be verified to show ≥90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

<u>Bases</u>

Pressure drop across the combined high efficiency particulate air (HEPA) filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per year operating cycle establishes performance capability.

(HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system every month will demonstrate operability of the filters and adsorber system. Operation for 15 minutes demonstrates operability and minimizes the moisture build up during testing.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

4.5.5 Low Pressure Injection System Leakage

Applicability

Applies to Low Pressure Injection System leakage.

Objective

To maintain a preventive leakage rate for the Low Pressure Injection System which will prevent significant off-site exposures.

Specification

4.5.5.1 Acceptance Limit

The maximum allowable leakage from the Low Pressure Injection System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

4.5.5.2 Test

Every 18 months, the following tests of the Low Pressure Injection System shall be conducted to determine leakage:

a. The portion of the Low Pressure Injection System, except as specified in (b), that is outside the containment shall be tested either by use in normal operation or by hydrostatically testing at 350 psig.

b. Piping from the containment emergency sump to the low pressure injection pump suction isolation valve shall be pressure tested at no less than 59 psig.

c. Visual inspection shall be made for excessive leakage from components of the system. Any excessive leakage shall be measured by collection and weighing or by another equivalent method.

Bases

The leakage rate limit for the Low Pressure Injection System is a judgement value based on assuring that the components can be expected to operate with-out mechanical failure for a period on the order of 200 days after a loss of coolant accident. The test pressure (350 psig) achieved either by normal system operation or by hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the Low Pressure Injection System (59 psig) is equivalent to the design pressure of the containment. The dose to the thyroid calculated as a result of this leakage is 0.76 rem for a two-hour exposure at the site boundary.

REFERENCE

FSAR, Section 15.15.4, and 6.3.3.2.2

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.

c. Also, the ability of the Keowee Unit ACBs to close automatically to the underground path will be tested on an annual frequency.

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

4.6.4 Every 18 months, a simulated emergency transfer of the 4160 volt main feeder buses to the startup

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Trip Insertion Time Test

Applicability

Applies to the surveillance of the control rod trip insertion time.

Objective

To assure the control rod trip insertion time is within that used in the safety analyses.

Specification

The control rod insertion time shall be measured at either full flow or no flow conditions as follows:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. For all rods at least once every 18 months.

The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66* seconds at reactor coolant full flow conditions or 1.40 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement.

If the trip insertion time above is not met, the rod shall be declared inoperable.

* - For Unit 1 Cycle 15, Group 1, Rod 8 and Group 2, Rod 5 may be considered operable with an insertion time ≤ 3.00 sec provided:

1) the average insertion time for the remaining rods in Groups 1 and 2 is ≤ 1.50 sec, and

2) the core average negative reactivity insertion rate is within the assumptions of the safety analysis.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR Chapter 15.

A rod is considered inoperable if the trip insertion time is greater than the specified allowable time or the core average negative reactivity insertion rate is less than the assumptions of the safety analysis.

REFERENCES

(1) FSAR, Section 15
(2) Technical Specification 3.5.2

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.

Specification

4.8 Using Channels A and B, the operation of each of the main steam stop valves shall be tested every 18 months 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.

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 every 18 months. 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 15.13, (but considering a blowdown of both steam generators) shows that the reactor will remain sub-critical after reactor trip following a double-ended steam line break.

REFERENCES

(1) FSAR, Section 10.3.4, and 15.13

4.9 EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine-driven and motor-driven emergency feedwater pumps and associated valves.

Objective

To verify that the emergency feedwater pumps and associated valves are operable.

Specification

4.9.1 Pump Test

The turbine-driven and motor-driven feedwater pumps shall be operated on recirculation to the upper surge tank for a minimum of one hour in accordance with the requirements of Specification 4.0.4.

4.9.2 Valve Test

Automatic valves in the emergency feedwater flow path will be determined to be operable in accordance with the requirements of Specification 4.0.4.

4.9.3 System Flow Test

Prior to Unit operation above 25% Full Power following any modifications or repairs to the emergency feedwater system which could degrade the flow path and at least once every 18 months, the emergency feedwater system shall be given either a manual or an automatic initiation signal.

4.9.4 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly. In addition, during operation of the System Flow Test (Item 4.9.3 above), flow to the steam generators shall be verified by control room indication.

Bases

The monthly testing frequency is sufficient to verify that the emergency feed-water pumps are operable. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pumps. The parameters which are observed are detailed in the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI. The System Flow Test verifies correct total system operation following modifications or repairs.

REFERENCES

(1) FSAR, Section 10.4.7.4

4.12 CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM

Applicability

Applies to control room pressurization and filtering system components

Objective

To verify that these systems and components will be able to perform their design functions.

Specification

4.12.1 Operating Tests

a. Control room outside air booster fan system tests shall be performed quarterly. These tests shall consist of an external visual inspection, a flow measurement for each unit and pressure drop measurements across each filter bank. Pressure drop across pre-filter shall not exceed 1 inch H_2O and pressure drop across HEPA shall not exceed 2 inches H_2O. Fan motors shall be operated continuously for at least one hour, and all louvers shall be proven operable.

b. Every 18 months, verify the system maintains the control room at a positive pressure with both outside air booster fans on during system operation.

4.12.2 Filter Tests

Every 18 months, for the Unit 1 and 2 and the Unit 3 control room an in-place leakage test using DOP on HEPA units and Freon-112 (or equivalent) on carbon units shall be performed at design flow on each filter train. Removal of 99.5 percent DOP by each entire HEPA filter unit and removal of 99.0 percent Freon-112 (or equivalent) by each entire carbon adsorber unit shall constitute acceptance performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

Bases

The purpose of the control room pressurization filtering system is to protect the control room operators from the effects of accidental release of radioactive effluents or toxic gases in the Turbine Building or Auxiliary Building only. The system is designed with two 50 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, carbon filters, booster fans, air handling unit fans, and associated ductwork to pressurize the control room with outside air.

Since these systems are not normally operated, a periodic test is required to insure their operability when needed. Quarterly testing of this system will show that the system is available.

Testing of the installed carbon adsorber stage and absolute filters every 18 months will verify the leak integrity of the cleanup system. Testing every 18 months will also verify the ability of the system to maintain the control room at a positive pressure to minimize infiltration of hazardous effluents.

4.14 REACTOR BUILDING PURGE FILTERS AND SPENT FUEL POOL VENTILATION SYSTEM

Applicability

Applies to testing of the Reactor Building purge filters for Units 2 and 3 and the spent fuel pool ventilation systems.

Objective

To verify that the Unit 2 and Unit 3 Reactor Building purge filters will per-form their design function and that when used with the spent fuel pool ventilation system, will reduce the off-site dose due to a fuel handling accident.

Specification

4.14.1 Operational and Performance Testing

a. Monthly, each train of the spent fuel pool ventilation system shall be operated through the respective Reactor Building purge filters for at least 15 minutes at design flow ± 10%.

b. Every 18 months, the spent fuel pool ventilation fans shall be shown to operate at design flow ± 10% when tested in accordance with ANSI N510-1975.

c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Reactor Building purge filters:

1. Every 18 months;
2. After each complete or partial replacement of HEPA filter bank or charcoal adsorber bank;
3. After any structural maintenance on the system housing;
4. After painting, fire, or chemical release in any ventilation zone communicating with the system.

d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

e. Every 18 months, or following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Reactor Building purge filters for laboratory analysis. Within 31 days of removal, this sample shall be verified to show ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

Bases

The Unit 2 Reactor Building purge filter is used in the ventilation system for the common spent fuel pool for Units 1 and 2. The Unit 3 Reactor Building purge filter is used in the Unit 3 spent fuel pool ventilation system. Each filter is constructed with a prefilter, an absolute filter and a charcoal filter in series. The high efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine.

Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the doses for a fuel handling accident would be minimized.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the spent fuel pool ventilation system every month will demonstrate operability of the fans, filters and adsorber system.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

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) Every 18 months, the inaccessible snubbers shall be inspected near the beginning and the end of an 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 by starting with the piston at the as found setting and extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. Snubber operability will be verified in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	4 months ± 25%
5,6,7	2 months ± 25%
≥8	1 month ± 25%

Note: (1) The required inspection interval shall not be lengthened more than two steps per inspection.

(2) Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility during reactor operation. These two groups may be inspected independently according to be above schedule.

(3) Hydraulic and mechanical snubber inspection schedules are independent.

4.18.2 The seal service life of hydraulic snubbers shall be monitored to ensure that the seals do not exceed their expected service life by more than 10% between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information, and the seals shall be replaced so that the maximum expected service life is not exceeded by more than 10% during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.5.1.m.

4.18.3 At least once every 18 months, a representative sample, a minimum of 10% of the total of hydraulic snubbers in use in the plant, shall be functionally tested either in place or in a bench test. For each hydraulic snubber that does not meet the functional test acceptance criteria of Specification 4.18.4, an additional minimum of 10% of the hydraulic snubbers shall be functionally tested until none are found inoperative or all have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of hydraulic snubbers. The representative sample shall be selected randomly from the total population of safety-related hydraulic snubbers.

4.18.6 Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions. Snubbers so exempted shall be listed in a permanent record which references the exemption letter date.

Bases

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval unless so determined, by the engineer, from a previous window of a schedule. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber.

To provide assurance of snubber functional reliability, a representative sample of the installed hydraulic snubbers will be functionally tested every 18 months. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

TABLE 4.20-1
SSF INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

		Check	Calibrate	Remarks
1.	RCS Pressure (3)	WE	18 months	Loop A, B
2.	SSF RC Makeup Pump (3)			
	Suction Pressure	QU(1)	18 months	
	Discharge Pressure	QU(1)	18 months	
	Suction Temperature	QU(1)	18 months	
	Discharge Flow	QU(1)	18 months	
3.	RC System Temperature (3)	NA(2)	18 months	Loop A, B Hot, Cold
4.	Pressurizer Water Level (3)	WE	18 months	
5.	SSF Auxiliary Service Water Pump			
	Suction Pressure	QU(1)	AN	
	Discharge Pressure	QU(1)	AN	
	Unit 1 Discharge Pressure	NA	AN	
	Unit 2 Discharge Pressure	NA	AN	
	Unit 3 Discharge Pressure	NA	AN	
	Discharge Test Flow	QU(1)	AN	
	Suction Temperature	QU(1)	AN	
6.	Steam Generator Levels (3)	WE	18 months	A,B
7.	Underground Fuel Oil Storage Tank Inventory	NA	AN	
8.	D/G Service Water Pump			
	Discharge Flow	QU(1)	AN	
	Discharge Pressure	QU(1)	AN	
9.	D/G Air Start System Pressure	WE	AN	

(1) Check when pump operated/tested per IST.

(2) This instrumentation is normally aligned through a transfer/isolation device to each Unit Control Room and is thus checked in accordance with Specification 4.1, Table 4.1-1, Item 7. Every 18 months, the instrument string to the SSF Control Room will be checked and calibrated.

(3) Units 1, 2, 3.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 228 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 225 TO FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated February 2, 1998, as supplemented by letter dated February 18, 1998, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TS). The requested changes would revise the wording that is presently used to specify refueling outage surveillances. The changes would indicate that these surveillances are to be performed on an 18-month frequency and need not be constrained to refueling outage conditions. These changes were addressed in a Notice of Enforcement Discretion (NOED) Number 98-06-001 issued by the staff on February 2, 1998, which is effective until superseded by these amendments.

The February 18, 1998, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The Oconee TS define Refueling Shutdown as a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods. It does not define a refueling outage, but provides in TS 4.0.2 that the maximum duration for the surveillances is 22 months, 15 days. However, there are many surveillance requirements that are to be performed "during refueling outages," which indicates that refueling conditions must exist. The current TS wording is unduly restrictive since it is possible to perform many valid surveillances during plant conditions other than during a refueling outage.

The proposed TS amendments would change the terminology that is presently used to specify when the surveillances are to be performed. This would be accomplished by replacing phrases containing "refueling" terminology such as "refueling outage," "refueling shutdown," or "refueling frequency," with wording that clearly states the required surveillance frequency of 18 months, where such a distinction is appropriate. This would allow credit to be taken for surveillances that are performed at plant conditions other than during refueling outages.

The maximum interval will continue to be 22 months, 15 days, as specified in TS 4.0.2, for surveillances conducted at 18-month intervals. In addition, a sentence would be added to TS 4.0.2 to indicate that words such as "each" and "every" that are used in some surveillance specifications are not intended to alter the frequencies described in the specifications. In other words, if a specification states that the surveillance must be performed every (or each) 18 months, the surveillance can be performed at any time as long as the duration between the last two surveillances does not exceed 22 months, 15 days. This would clarify that the surveillance need not be performed at exactly 18-month intervals.

In addition to TS 4.0.2, the licensee proposed changes to the following TS sections to replace the refueling outage surveillance intervals with 18-month intervals: (1) Table 4.1-1, Item Numbers 7, 8, 9, 10, 11, 12, 13, 15a, 15b, 17a, 17b, 19, 21, 22, 23, 24, 25a, 25b, 26, 27, 29, 30, 33, 35, 36, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52a, 53a, 54, 56, 58, 59, 60, and 61; (2) Table 4.1-2, Item Numbers 2, 3, 7, 11, and Note 4; (3) TS 4.2.6.b, TS 4.4.3.1, TS 4.4.3.2.b, TS 4.4.4.3, 4.5.1.1.1.a, and 4.5.1.1.2.a; (4) Note 1 at the bottom of TS Page 4.5-1; (5) TS 4.5.1.1.3.a, 4.5.1.2.1.b, 4.5.2.1.1.a(1), 4.5.2.1.2.a, 4.5.3.1.a, 4.5.3 Bases, 4.5.4.1.b, 4.5.4.1.c.1, 4.5.4.1.e, 4.5.5.2, and 4.6.4; (6) TS 4.7.1, Specification c; (7) Specification 4.8 and 4.8 Bases; (8) TS 4.9.3, 4.12.1.b, 4.12.2, and 4.12 Bases; (9) TS 4.14.1.b, 4.14.1.c.1, 4.14.1.e, 4.18.1(3)(a), 4.18.3, 4.18 Bases; and (10) Table 4.20-1, Item Numbers 1, 2, 3, 4, 6, and Note (2). Also, "RF - Refueling Outage" would be deleted from the list of acronyms at the end of Table 4.1-1.

The proposed changes are consistent with the Oconee Improved Technical Specifications (ITS), that were written using the guidance in NUREG-1430, Revision 1, and submitted by letter dated October 28, 1997. This submittal is under staff review.

The proposed changes are limited to clarifying surveillance intervals and do not modify every usage of wording that refers to refueling conditions or refueling activities. Some TS requirements are event-driven rather than frequency-driven. Other TS clearly indicate that testing may be performed during plant conditions other than refueling outages.

The staff finds the proposed changes acceptable since they are consistent with the standard TS for Oconee, more clearly specify and identify the frequency rather than the plant condition for performing the surveillances, do not alter the maximum frequency (22 months, 15 days in accordance with TS 4.0.2) at which frequency many of the tests have been performed in the past, and will not adversely impact public health and safety.

3.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case where the Commission and licensee need to act promptly and time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment, and it is determined that the amendment involves no significant hazards consideration.

Under such circumstances, the Commission notifies the public in one of two ways: by using a Federal Register notice providing an opportunity for hearing and allowing at least 2 weeks for prior public comments, or by issuing a press release discussing the proposed changes, using the local media. In this case, the Commission used the first approach.

The licensee submitted the request for amendments on February 2, 1998. It was noticed in the Federal Register on February 10, 1998 (63 FR 6784), at which time the staff proposed a no significant hazards consideration determination. The licensee requested that the amendments be issued on an exigent basis in accordance with the staff policy for processing a NOED.

The original Oconee TS required that certain surveillances be performed annually and, therefore, were not constrained to performance with a unit in the refueling condition. As a result, the licensee has not interpreted a surveillance that is specified to be performed at refueling outage frequency as meaning that the unit must be in a refueling outage to satisfy the requirement. Therefore, some surveillances specified at a refueling outage frequency were performed at times other than during a refueling outage. In discussions with the NRC staff on January 29, 1998, the licensee was informed of the staff's interpretation of Oconee's TS that concluded any surveillance that was specified to be performed during refueling outages must be performed with the unit in a refueling outage. Thus, any surveillances performed at power, in past forced outages, or during planned shutdowns, would not satisfy the TS requirements.

Prior to January 29, 1998, the licensee did not recognize that terminology specifying the frequency of surveillances (i.e., the refueling outage frequency established in the surveillance specifications) also defined the condition at which the surveillance must be performed. Once notified of this requirement, the licensee immediately began to evaluate the impact of this interpretation of the TS. On January 30, 1998, the licensee confirmed that certain surveillances had been performed at times other than during a refueling outage and that implementation of the staff's interpretation of the surveillances designated in the TS as "refueling outage" would result in exceeding the time constraints allowed in the TS for these surveillances and, in accordance with TS 3.0, would result in the forced shutdown of Units 2 and 3 and interfere with the planned startup of Unit 1. When these findings were discussed with the staff on January 30, 1998, an NOED was issued verbally, which allowed the exercise of discretion not to enforce compliance with TS 3.0 for these surveillances for the period from 3:30 p.m. on January 30, 1998, until issuance of these amendments. The NOED was confirmed in writing on February 2, 1998.

These amendments complete the review process and implement the proposed TS changes, pursuant to the NRC's policy regarding exercising discretion for an operating facility set out in Section VII.c of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, for processing NOEDs. The staff has determined that, because compliance with the refueling outage surveillances would necessitate either plant shutdown or delayed startup, and in light of the NOED, issuance of these amendments is needed in less than the 30-day comment period normally allowed for processing amendments to the TS. The licensee promptly submitted its application letter after being advised of the staff's interpretation of the surveillance TS. Therefore, pursuant to 10 CFR 50.9(a)(6), the staff has determined that exigent circumstances exist and the amendments are being processed accordingly.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92, state that the Commission may make a final determination that license amendments involve no significant hazards consideration if operation of the facility, in accordance with the proposed amendments, would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

> This proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to involve no significant hazards, in that operation of the facility in accordance with the proposed amendment would not:
>
> 1. Involve a significant increase in the probability or consequences of an accident previously evaluated?
>
>   No. The proposed change will revise the surveillance requirements for selected surveillances which have a refueling outage surveillance frequency with a maximum interval of 22 months and 15 days. The proposed change will replace the refueling outage requirement with a comparable requirement to perform the surveillance every 18 months which has a maximum interval of 22 months and 15 days. The proposed change does not increase the maximum interval between surveillances and does not change any surveillance acceptance criteria. Thus, the probability and consequences of an accident previously evaluated will not be significantly increased.
>
> 2. Create the possibility of a new or different kind of accident from the accidents previously evaluated?
>
>   No. Since the proposed change does not increase the maximum interval between surveillances and does not change any surveillance acceptance criteria, a new or different kind of accident from the accidents which were previously evaluated will not occur.
>
> 3. Involve a significant reduction in a margin of safety?
>
>   No. The margin of safety will not be significantly reduced by this amendment request because the maximum interval between the surveillances and the surveillance acceptance criteria are not changed. Thus, the operability of the plant equipment and systems will be verified within the same surveillance interval and to the same acceptance criteria.

Based on the above considerations, the NRC staff concludes that the amendments meet the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendments do not involve a significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final finding that the amendments involve no significant hazards consideration. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: David E. LaBarge

Date: February 26, 1998