Docket-3

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

April 1, 1981

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

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 96 ,96 , and 93 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your requests dated May 1, 1979, February 16, 1981 and March 6, 1981.

These amendments revise the TSs to upgrade the Engineering Safety Features ventilation filter systems surveillance requirements, revise various surveillance requirement testing intervals from annually to refueling cycle to correspond with the 18-month refueling cycle interval, and incorporate requirements for the anticipatory reactor trip system.

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

Sincerely,

John F. Stolz

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

Enclosures:
1. Amendment No.96 to DPR-38
2. Amendment No.96 to DPR-47
3. Amendment No.93 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

Oconee County Library
501 West Southbroad Street
Walhalla, South Carolina 29691

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

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

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

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

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

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

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

5/1/79, 2/16/81 & 3/6/81
Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
License No. DPR- 38

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

    A. The applications for amendment by Duke Power Company (the licensee) dated May 1, 1979, February 16, 1981 and March 6, 1981, comply 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 applications, 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. 96 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: April 1, 1981

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
License No. DPR-47

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

   A. The applications for amendment by Duke Power Company (the licensee) dated May 1, 1979, February 16, 1981 and March 6, 1981, comply 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 applications, 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. 96 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: April 1, 1981

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. DPR-55

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

   A. The applications for amendment by Duke Power Company (the licensee) dated May 1, 1979, February 16, 1981 and March 6, 1981, comply 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 applications, 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. 93 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: April 1, 1981

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 96 TO DPR-38

AMENDMENT NO. 96 TO DPR-47

AMENDMENT NO. 93 TO DPR-55

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

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

REMOVE PAGES	INSERT PAGES
3.5-4	3.5-4
3.5-5	3.5-5
3.5-5a	3.5-5a
4.1-8	4.1-8
4.4-10	4.4-10
4.4-11	4.4-11
-	4.4-12
4.5-1	4.5-1
4.5-2	4.5-2
4.5-5	4.5-5 *
4.5-6	4.5-6
4.5-10	4.5-10
4.5-11	4.5-11
4.5-12	4.5-12
4.6-1	4.6-1
4.6-2	4.6-2
4.6-3	4.6-3
4.7-1	4.7-1
4.7-2	4.7-2*
4.10-1	4.10-1
4.12-1	4.12-1
4.14-1	4.14-1
4.14-2	4.14-2
4.19-1	4.19-1

*No change on this page; provided for convenience only.

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS

	Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
1.	Nuclear Instrumentation Intermediate Range Channels	1	0	Bring to hot shutdown within 12 hours (b)
2.	Nuclear Instrumentation Source Range Channels	1	0	Bring to hot shutdown within 12 hours (b)(c)
3.	RPS Manual Pushbutton	1	0	Bring to hot shutdown within 12 hours
4.	RPS Power Range Instrument Channels	3(a)	1(a)	Bring to hot shutdown within 12 hours
5.	RPS Reactor Coolant Temperature Instrument Channels	2(d)	1	Bring to hot shutdown within 12 hours
6.	RPS Pressure-Temperature Instruments Channels	2(d)	1	Bring to hot shutdown within 12 hours
7.	RPS Flux Imbalance Flow Instrument Channels	2	1	Bring to hot shutdown within 12 hours
8.	RPS Reactor Coolant Pressure			
	a. High Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours
	b. Low Reactor Coolant Pressure Channels	2	1	Bring to hot shutdown within 12 hours
9.	RPS Power-Number of Pumps Instrument Channels	2	1	Bring to hot shutdown within 12 hours
10.	RPS High Reactor Building Pressure Channels	2	1	Bring to hot shutdown within 12 hours
11.	RPS Anticipatory Reactor Trip System (g)			
	a. Loss of Turbine	2	1	Bring to hot shutdown within 12 hours
	b. Loss of Main Feedwater	2	1	Bring to hot shutdown within 12 hours

Amendments Nos. 96, 96, 93

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (cont'd)

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
12. ESF High Pressure Injection System and Reactor Building Isolation (Non-essential Systems)			
a. Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
b. Reactor Building 4 PSIG Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
13. ESF Low Pressure Injection System			
a. Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
b. Reactor Building 4 PSIG Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
14. ESF Reactor Building Isolation (Essential Systems) & Reactor Building Cooling System			
a. Reactor Building 4 PSIG Instrument Channel	2	1	Bring to hot shutdown within 12 hours (e)
b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
15. ESF Reactor Building Spray System			
a. Reactor Building High Pressure Instrument Channel	2	1	Bring to hot shutdown within 12 hours (e)

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

	Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
	b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
16.	Turbine Stop Valves Closure	2	1	Bring to hot shutdown within 12 hours (f)

(a) For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

(d) Single loop operation at power (after testing and approval by the NRC/DOL) is not permitted unless the operating channels are the two receiving Reactor Coolant Temperature from operating loop.

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in the cold shutdown condition within 24 hours.

(f) One operable channel with zero minimum degree of redundancy is allowed for 24 hours before going to the hot shutdown condition.

(g) This requirement is applicable as follows:

Unit 1 - following Summer 1981 refueling outage

Unit 2 - following Fall 1981 refueling outage

Unit 3 - immediately upon the effective date of this license amendment

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	NA	MO	RF	
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater	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

4.4.3 Hydrogen Purge System

Applicability

Applies to the Reactor Building Hydrogen Purge System.

Objective

To verify that the Reactor Building Hydrogen Purge System is operable.

Specification

4.4.3.1 In-place Testing

a. During each refueling outage, an in-place system test shall be performed. This test shall demonstrate that under simulated emergency conditions, the system can be taken from storage and placed into operation within 48 hours.

b. This refueling outage test shall consist of:

1. Visual inspection of the system.

2. Hook-up of the system to one of the three Reactor Buildings.

3. Flow measurement using flow instruments in the portable purging station.

4. Verification that 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%).

5. Verification of the operability of the heater at rated power when tested in accordance with ANSI N510-1975.

4.4.3.2 Operational Performance Testing

a. The testing requirements of this section may be performed without hooking-up the system to one of the Reactor Buildings.

b. Monthly, the hydrogen purge system shall be operated with the heaters on for at least ten hours.

c. During each refueling outage, the hydrogen purge system fans shall be shown to operate at design flow (±10%) when tested in accordance with ANSI N510-1975.

d. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the hydrogen purge filters:

1. During each refueling outage;

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.

e. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975. Otherwise, the filter system shall be declared inoperable.

f. During each refueling outage, 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 $\geq$90% radioactive methyl iodide removal when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

4.4.3.3 H_2 Detector Test

Hydrogen concentration instruments shall be calibrated each refueling outage with proper consideration to moisture effect.

Bases

Pressure drop across the combined high efficiency particulate air (HEPA) filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amount of foreign matter. A test frequency of once per year establishes system 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. Operations 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 at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system 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 ten hours is used to reduce the moisture built up on the adsorbent.

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.

4.5 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. During each refueling outage, 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. During each refueling outage, 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 System which supplies cooling water to the low pressure 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; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.

4.5.1.1.3 Core Flooding System

a. During each refueling outage, 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 Pumps

Quarterly, the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for 15 minutes, and the discharge pressure and flow are within ± 10 percent of a point on the pump head curve. (Figures 4.5.1-1 and 4.5.1-2)

4.5.1.2.2 Valves - Power Operated

a. Quarterly, each Engineered Safety Features valve in the Emergency Core Cooling Systems and each Engineered Safety Features valve associated with emergency core cooling in the Low Pressure Service Water System shall be tested to verify operability.

b. The acceptable performance of each power-operated valve will be that motion is indicated upon actuation by appropriate signals.

c. During each refueling outage, low pressure injection pump discharge (engineered safety features) valves, low pressure injection discharge throttling valves, and low pressure injection discharge header crossover valves shall be cycled manually to verify the manual operability of these power-operated valves.

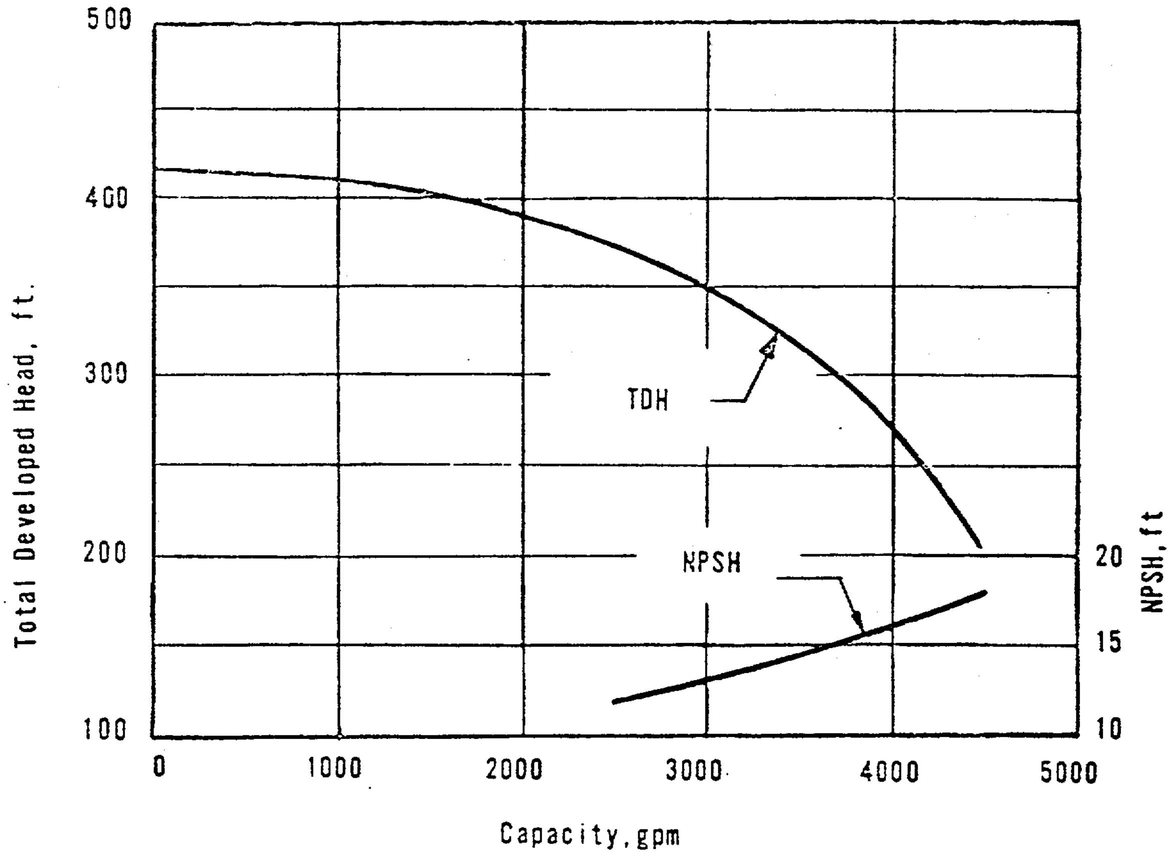
Bases

The Emergency Core Cooling Systems are the principle reactor safety features in the event of a 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. At least once per month operation is rotated to another high pressure injection pump. This verifies that the high pressure injection pumps are operable.

The requirements of the Low Pressure Service Water System for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis verifies that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.



LOW PRESSURE INJECTION
PUMP CHARACTERISTICS

OCONEE NUCLEAR STATION
Figure 4.5.1-2

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. During each refueling outage, 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 (except for reactor building inlet valves to prevent water entering nozzles). Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

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

c. 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.

4.5.2.1.2 Reactor Building Cooling System

a. During each refueling outage, 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.

4.5.3 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

4.5.3.1 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 $\pm$10%.

b. During each refueling outage, 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 Pentration 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. During each refueling outage;

   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 $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

e. During each refueling outage, 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 $\geq$90% radioactive methyl iodide removal when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

Bases

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 operating cycle establishes system 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.

4.5.4 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

4.5.4.1 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.

4.5.4.2 Test

During each refueling outage, 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 judgment value based on assuring that the components can be expected to operate without 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 14.2.2.4.4

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 off-site 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 connected to the 100 kV line. It shall be demonstrated that the 100 kV line can be separated from the rest of the system and supply power to the 4160 volt main feeder buses.

4.6.7 At least once every 18 months, it shall be demonstrated that a Lee station combustion turbine can be started and connected to the isolated 100 kV line and carry the equivalent of the maximum safeguards load of one Oconee unit (4.8 MVA) within one hour.

4.6.8 Annually, it shall be demonstrated that a Lee station combustion turbine can be started and carry the equivalent of the maximum safeguards load of one Oconee unit plus the safe shutdown loads of two Oconee units on the system grid.

4.6.9 Batteries in the Instrumentation and Control, Keowee, and Switching Station shall have the following periodic inspections performed to assure maximum battery life. Any battery or cell not in compliance with these periodic inspection requirements shall be corrected to meet the requirements within 90 days or the battery shall be declared inoperable.

a. Weekly verify that:

(1) The electrolyte level of each pilot cell is in between the minimum and maximum level indication marks.

(2) The pilot cell specific gravity, corrected to $77^{\circ}F$ and full electrolyte level, is $\geq$ 1.200.

(3) The pilot cell float voltage is $\geq$ 2.12 VDC.

(4) The overall battery float voltage is $\geq$ 125 VDC.

b. Quarterly verify that:

(1) The specific gravity of each cell corrected to $77^{\circ}F$ and full electrolyte level, is $\geq$ 1.200 and is not less than 0.010 below the average of all cells measured.

(2) The voltage of each cell under float charge is $\geq$ 2.12 VDC.

(3) The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. Annually verify that:

(1) The cells, end-cell plates and battery racks show no visual indication of structural damage or degradation.

(2) The cell to cell and terminal connections are clean, tight and coated with anti-corrosion grease.

4.6.10 Annually, a one hour discharge service test at the required maximum load shall be made on the instrument and control batteries, the Keowee batteries, and the switching station batteries.

4.6.11 Monthly, the operability of the individual diode monitors in the Instrument and Control Power System shall be verified by imposing a simulated diode failure signal on the monitor.

4.6.12 Semiannually, the peak inverse voltage capability of each auctioneering diode in the 125 VDC Instrument and Control Power System shall be measured and recorded.

Bases

The Keowee Hydro units, in addition to serving as the emergency power sources for the Oconee Nuclear Station, are power generating sources for the Duke system requirements. As power generating units, they are operated frequently, normally on a daily basis at loads equal to or greater than required by Table 8.5 of the FSAR for ESF bus loads. Normal as well as emergency startup and operation of these units will be from the Oconee Unit 1 and 2 Control Room. The frequent starting and loading of these units to meet Duke system power requirements assures the continuous availability for emergency power for the Oconee auxiliaries and engineered safety features equipment. It will be verified that these units will carry the equipment of the maximum safeguards load within 25 seconds, including instrumentation lag, after a simulated requirement for engineered safety features. To further assure the reliability of these units as emergency power sources, they will be, as specified, tested for automatic start on a monthly basis from the Oconee control room. These tests will include verification that each unit can be synchronized to the 230 kV bus and that each unit can energize the 13.8 kV underground feeder.

The interval specified for testing of transfer to emergency power sources is based on maintaining maximum availability of redundant power sources.

Starting a Lee Station gas turbine, separation of the 100 kV line from the remainder of the system, and charging of the 4160 volt main feeder buses are specified to assure the continuity and operability of this equipment. The one hour time limit is considered the absolute maximum time limit that would be required to accomplish this.

REFERENCE

FSAR Section 8

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 following each refueling outage.

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.

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 14.

A rod is considered inoperable if the trip insertion time is greater than the specified allowable time.

REFERENCES

(1) FSAR, Section 14

(2) Technical Specification 3.5.2

4.7.2 Control Rod Program Verification

Applicability

Applies to surveillance of the control rod systems.

Objective

To verify that the designated control rod (by core position 1 through 69) is operating in its programmed functional position and group. (Rod 1 through 12, Groups 1-8)

Specification

4.7.2.1 Whenever the control rod drive patch panel is locked (after inspection, test, reprogramming, or maintenance) each control rod drive mechanism shall be selected from the control room and exercised by a movement of approximately two inches to verify that the proper rod has responded as shown on the unit computer printout of that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

Bases

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer, identified by a unique number (1 through 69) associated with only one core position. The other set of outputs goes to a programmable bank of 69 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or to the control room meter bank are improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by: (1) selecting a specific rod from one group (e.g., Rod 1 in Regulating Group 6), (2) noting that the program-approved core position for this rod of the group (assume the approved core position is No. 53), (3) exercising the selected rod and (4) noting that the computer prints out both absolute and relative position response for the approved core position (assumed to be position No. 53) and that the proper meter responds in the control room display bank (assumed to be Rod 1 in Group 6) for both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the reactor building. For these, it is necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require the evaluation of reactivity anomalies of a specified magnitude occurring during the operation of the unit.

Specification

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of discrepancy shall be made and reported to the Nuclear Regulatory Commission.

Bases

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

4.12 CONTROL ROOM FILTERING SYSTEM

Applicability

Applies to control room 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

System tests shall be performed quarterly. These tests shall consist of visual inspection, a flow measurement at the outlet of 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 and other mechanical systems shall be proven operable.

4.12.2 Filter Tests

During each refueling outage, 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 charcoal 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 charcoal adsorber unit shall constitute acceptable 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 Filtering System is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with two 100 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, charcoal filters and a booster fan 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 for its safety action. During this test the system will be inspected for such things as water, oil, or other foreign material, gasket deterioration, adhesive deterioration in the HEPA units, and unusual or excessive noise or vibration when the fan motor is running.

Refueling outage testing will verify the efficiency of the charcoal and absolute filters.

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 respective spent fuel pool ventilation systems.

Objective

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

Specificiation

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 $\pm$ 10%.

b. During each refueling outage, the spent fuel pool ventilation fans shall be shown to operate at design flow $\pm$ 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. During each refueling outage;

   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 $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

e. During each refueling outage, 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 $\geq$90% radioactive methyl iodide removal when tested in accordance with ANSI N510-1975 (130°, 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.

4.19 FIRE PROTECTION AND DETECTION SYSTEM

Applicability

Applies to the fire protection and detection systems which protect systems and equipment required for safe shutdown.

Objective

To verify the operability of fire protection and detection systems.

Specifications

4.19.1 The High Pressure Fire Protection System components shall be tested as follows:

	Item	Frequency
(a)	High pressure service water pump functional test	Monthly
(b)	System functional test	Every 18 months
(c)	High pressure service water pump capacity test to verify flow of 3000 gpm	Annually
(d)	System Flow Test in Accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, NFPA	Every 3 years
(e)	Alignment of fire protection valves	Monthly
(f)	Sprinkler systems in safety related areas	
	1. System functional test	Each refueling
	2. Inspection of spray headers	Annually*
	3. Inspection of spray nozzle	Annually*
(g)	Fire hose stations	
	1. Visual inspection	Monthly*
	2. Maintenance inspection	Annually*
	3. Partial opening of fire hose station valve	Every 3 years
	4. Hose Hydrostatic test at least 50 psig greater than the maximum pressure at the station	Every 3 years

*This frequency applies only for areas which are normally accessible during operation. If an area is inaccessible during operation, inspections shall be performed in those areas during each refueling outage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO.96 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO.93 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

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

Introduction

By letter dated May 1, 1979, Duke Power Company (the licensee) proposed revisions to the Oconee Nuclear Station (Oconee) Technical Specifications (TSs) which alter Sections 4.4.3, 4.5.3 and 4.14. These changes are primarily administrative in nature, modifying the format of the specifications. Changes to Sections 4.5.3.1a and 4.14.1a were subsequently made following telephone discussions with the licensee. By letter dated February 16, 1981, the licensee proposed revising various surveillance requirement intervals from annually to refueling cycle to coincide with the extended (18-month) refueling cycle. By letter dated March 6, 1981, the licensee proposed requirements for the operability and testing of the anticipatory reactor trip system.

Evaluation

I. Filter Testing

We have reviewed the proposed changes to Sections 4.4.3 (Hydrogen Purge System), 4.5.3. (Penetration Room Ventilation System) and 4.14 (Reactor Building Purge Filters and Spent Fuel Pool Ventilation Systems) of the Oconee TSs requested by letter dated May 1, 1979. These sections specify the limiting conditions for operation and surveillance requirements on three Engineering Safety Features (ESF) ventilation filter systems which are used to mitigate the radiological consequences of accidents at Oconee. Most of the changes are only to modify the format of the three above sections in the ESF ventilation filter system and do not reduce any of the requirements in the present Oconee TSs on the Hydrogen Purge System, Penetration Room Ventilation System and Reactor Building Purge Filters and Spent Fuel Pool Ventilation System.

In addition to the proposed changes to modify the present format of Sections 4.4.3, 4.5.3 and 4.14 of the Oconee TSs, the licensee requested changes to (1) delete the prefix "cold" from references to DOP tests in the Oconee TSs, (2) allow 31 days following removal of a carbon sample to verify that the sample has an acceptable methyl iodide removal efficiency, and (3) for only the Hydrogen Purge System, require removing a charcoal sample after every 720 hours of system operation to test the sample's methyl iodide removal efficiency. The requested change to delete the prefix "cold" from references to DOP tests is consistent with the in-place testing criteria of Regulatory Guide 1.52 (Revision 2), "Design Testing and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Clean-up

System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants". The proposed change to allow 31 days between removing the charcoal sample from the ventilation filter systems and verifying the methyl iodide removal efficiency of the charcoal sample is the standard time allowed for verification of the charcoal radioiodine removal efficiency. If this efficiency is too low, the system would be declared inoperable until the charcoal in the system was replaced. The proposed change, to require removing a charcoal sample once every 720 hours of Hydrogen Purge System operation to test the sample's methyl iodide removal efficiency, is in accordance with the requirement specified in footnote c. to Table 2 of Regulatory Guide 1.52.

Based on the considerations given above, we conclude that these proposed changes to Sections 4.4.3, 4.5.3 and 4.14 of the Oconee TSs are acceptable.

The licensee was asked to amend Sections 4.5.3.1a and 4.14.1a to require that monthly each train of the Penetration Room and Spent Fuel Pool Ventilation System be started from the control room and verified operable at design flow within ± 10%. Previous to the suggested alteration of the licensee's submittal, no operability requirement was stated in the Oconee TSs. These changes increase the assurances that these ventilation filter systems will be available at an acceptable flow rate when needed. The licensee has agreed to include this additional requirement.

The proposed changes discussed above to the Oconee ESF ventilation filter systems do not change any of the assumptions made to calculate the potential consequences of postulated design basis accidents at Oconee. The potential consequences of these postulated accidents, which are not changed by these proposed changes to the Oconee TSs, are given in Safety Evaluations (SEs) dated December 1970 and July 1973 for Oconee.

Licensee Event Reports (LERs) relevant to ESF air filtration and adsorption systems have also been reviewed. LER RO-287/78-19 discusses failure of the penetration filters due to moisture saturation caused by steam leakage on Oconee Unit 3. LER 79-023/03L-9 and 79-030/03L-0 discuss declaring the penetration room ventilation system inoperable due to high humidity from steam leaks on Oconee Unit 1. In the Oconee Unit 1 SE, 50% of all containment leakage is assumed to go through the penetration room filtration system which is considered 90% efficient in iodine removal. Should the Unit 1 filtration system be inoperative, our calculated design basis Loss of Coolant Accident (LOCA) 2-hour thyroid dose (refer to Regulatory Guide 1.4) would increase from 190 Rem to 345 Rem. For Oconee Units 2 and 3, the iodine removal efficiency for the 50% containment leakage to the penetration room filtration system is assumed in the SE to be 90% for elemental and particulate iodine and 70% for organic iodine. Should the Unit 2 or 3 filtration system be inoperative, our calculated design basis LOCA 2-hour thyroid dose would increase from 235 Rem to 424 Rem. It is, therefore, concluded that the penetration room filtration system must be operational to prevent the design basis LOCA 2-hour site exclusion boundary thyroid dose from exceeding the 300 Rem limit in 10 CFR Part 100. All three LERs conclude that the offsite release during a LOCA would be well within the guidelines of 10 CFR Part 100 without the penetration room ventilation system in operation since the licensee presumably did not use the design basis LOCA assumptions that are defined in Regulatory Guide 1.4. We, therefore, determined that additional assurance of the operability of the penetration room filters was necessary. The licensee investigated the possibility of including demisters and heaters or cooling coils designed to reduce the inlet stream relative humidity to less than 70% and found these modifications to be impractical. The licensee has, however, replaced the check valves in the main feedwater lines on Unit 3 (these valves are a major source of the humidity problem due to leakage) and has initiated engineering schedules for similar replacement of these valves on Units 1 and 2 at the next available unit outage. The licensee has also implemented procedures to monitor the humidity in the penetration room and take prompt action to reduce the humidity to less than 70% whenever this value is exceeded. We find that these modifications and procedures provide sufficient additional assurance that the penetration room filter will remain operable and are acceptable.

II. Surveillance Testing Intervals

By letter dated February 16, 1981, the licensee proposed to revise the surveillance interval for the presently required annual tests for the filter system in Section 4.5.3 to a refueling cycle interval. Discussions with the licensee disclosed that the same change was requested for the filter systems in Sections 4.4.3 and 4.14. (A similar request to extend surveillance intervals was approved by Amendments Nos. 91, 91 and 88 which were issued on January 28, 1981, for the Oconee Units 1, 2 and 3, respectively). Since it is the NRC staff's intent that such tests be performed at least once per operating cycle and since the refueling cycle interval has been defined, by the previously mentioned Amendments, to be in accordance with the latest NRC guidance contained in NUREG-0103, Revision 4, "Standard Technical Specifications for B&W PWRs", we find these changes to be acceptable.

The licensee also included revisions to Sections 4.5.2 (Reactor Building Cooling Systems), 4.5.4 (Low Pressure Injection System Leakage), 4.6 (Emergency Power Periodic Testing), 4.7 (Reactor Control Rod System Tests), 4.12 (Control Room Filtering System) and 4.19 (Fire Protection and Detection System) to extend various surveillance tests from annually to at least once per refueling outage. We have reviewed these changes and find them to be in accordance with the requirements given in NUREG-0103, Revision 4, and have concluded that they are acceptable. It should be noted that the surveillance testing required to be performed "during each refueling outage" need not be performed more frequently than once every 22-1/2 months, even though a special circumstance may arise which requires refueling operations at a shorter interval, and may be performed at times other than a refueling outage. This interpretation is consistent with the requirements of NUREG-0103.

An editorial change was also included in the licensee's February 16, 1981, request which revises the requirement to report Reactivity Anomalies (in Section 4.10) to the Nuclear Regulatory Commission instead of the predecessor agency, the Atomic Energy Commission. This change is desirable and acceptable.

III. Anticipatory Reactor Trip System Requirements

By letter dated March 6, 1981, the licensee proposed TSs to require the operability and testing of the anticipatory reactor trip system. Approval of the system was provided by a letter to the licensee from the NRC dated December 4, 1980, which attached the NRC staff's SE and requested that TS requirements be submitted. The licensee's March 6, 1981, proposal was in response to the staff's December 4, 1980 request.

We have reviewed the licensee's submittal and find that it is in accordance with and responsive to our request. We further find that the proposed TSs contain the requirements which are applicable to other similar systems in use at other nuclear plants and those contained in the Standard Technical Specifications for Babcock and Wilcox PWRs. We, therefore, conclude that these additional requirements are acceptable.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 1, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 96 , 96 and 93 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the Technical Specifications for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications to upgrade the Engineered Safety Features ventilation filter systems surveillance requirements, revise various surveillance requirement testing intervals from annually to refueling cycle to correspond with the 18-month refueling cycle interval, and incorporate requirements for the anticipatory reactor trip system.

The applications for the amendments 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. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the applications for amendments dated May 1, 1979, February 16, 1981, March 6, 1981, (2) Amendments Nos. 96 , 96 , and 93 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oconee County Library, 501 West Southbroad Street, Walhalla, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 1st day of April 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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