



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:

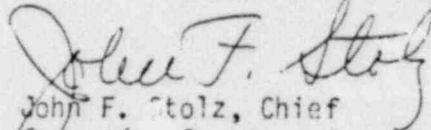
3.B Technical Specifications

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.

B104100/112

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

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: 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

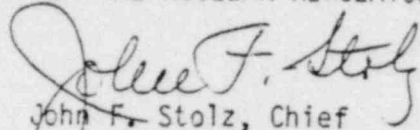
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The applications for amendment by Duke Power Company (the licensee) dated May 1, 1979, February 16, 1981 and March 6, 1981, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and 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, 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

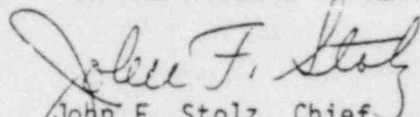
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The applications for amendment by Duke Power Company (the licensee) dated May 1, 1979, February 16, 1981 and March 6, 1981, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and 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, 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 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

<u>Functional Unit</u>	<u>(A) Minimum Operable Channels</u>	<u>(B) Minimum Degree of Redundancy</u>	<u>(C) Operator Action If Conditions Of Column A and B Cannot Be Met</u>
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



TABLE 3.5.1-1

## INSTRUMENTS OPERATING CONDITIONS (cont'd)

<u>Functional Unit</u>	(A) <u>Minimum Operable Channels</u>	(B) <u>Minimum Degree of Redundancy</u>	(C) <u>Operator Action If Conditions Of Column A and B Cannot Be Met</u>
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)

<u>Functional Unit</u>	(A) <u>Minimum Operable Channels</u>	(B) <u>Minimum Degree of Redundancy</u>	(C) <u>Operator Action If Conditions Of Column A and B Cannot Be Met</u>
b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
15. 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)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
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  
 HO - 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 ( $\pm 10\%$ ).
  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 ( $\pm 10\%$ ) 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  $>90\%$  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<sub>2</sub> Detector Test

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

### 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  $\pm 10$  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

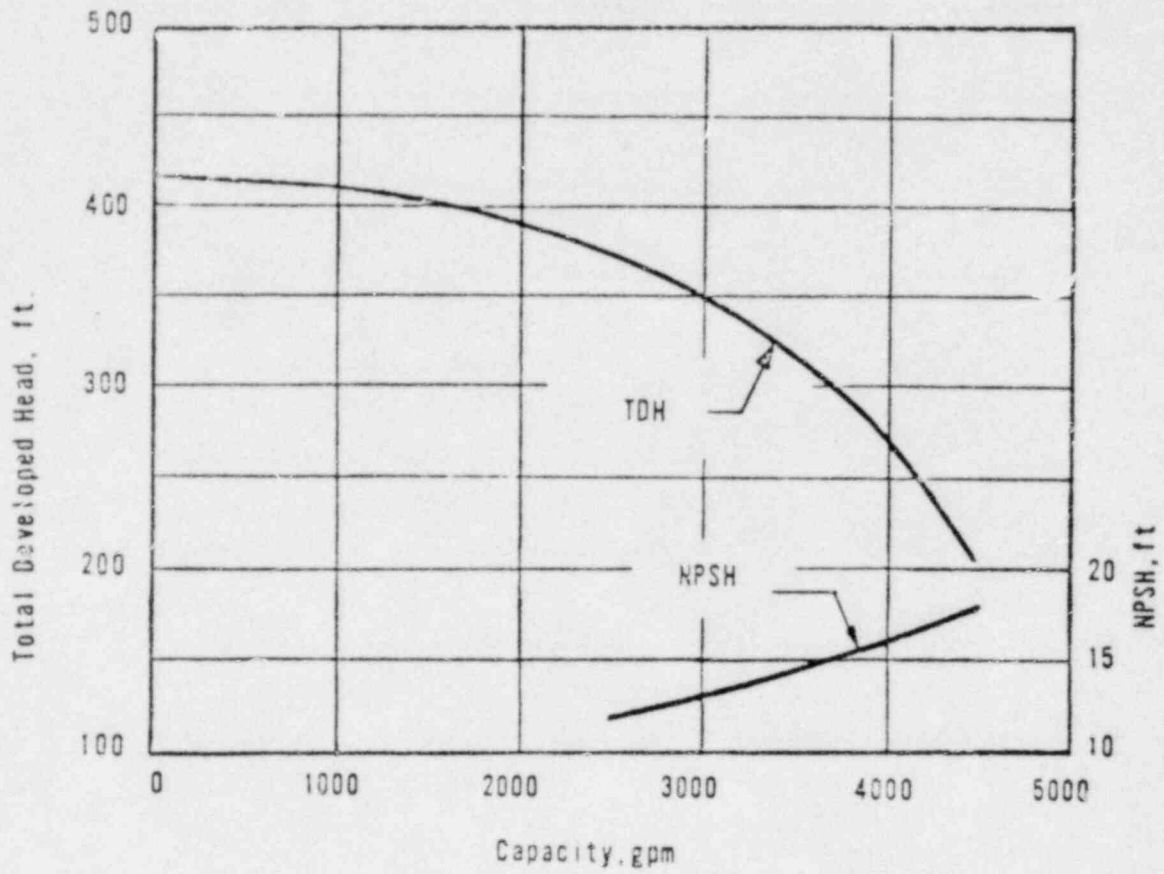
The Emergency Core Cooling Systems are the principle reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

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 ( $\pm 10\%$ ) 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 ( $\pm 10\%$ )
  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. 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  $>99\%$  DOP removal and  $>99\%$  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 >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.

#### ases

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°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°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<sub>2</sub>O and pressure drop across HEPA shall not exceed 2 inches H<sub>2</sub>O. 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.

##### 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  $\pm$  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  $>$  99% DOP removal and  $>$  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.

### ases

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:

<u>Item</u>	<u>Frequency</u>
(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.