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. 104
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated November 30, 1976, as supplemented on October 24 and December 29, 1980, and July 24 and September 3, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 104 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: November 6, 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. 104
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated November 30, 1976, as supplemented on October 24 and December 29, 1980, and July 24 and September 3, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 104 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: November 6, 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. 101
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated November 30, 1976, as supplemented on October 24 and December 29, 1980, and July 24 and September 3, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I:

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 101 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: November 6, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 104 TO DPR-38

AMENDMENT NO. 104 TO DPR-47

AMENDMENT NO. 101 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
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vi	vi
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--	3.6-3
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4.4-8	4.4-8
4.4-9	4.4-9
4.4-10	4.4-10
4.4-11	4.4-11
4.4-12	4.4-12
--	4.4-13
--	4.4-14 (was 4.4-6)
--	4.4-15 (was 4.4-7)
--	4.4-16 (was 4.4-8)
--	4.4-17 (was 4.4-10)
--	4.4-18 (was 4.4-11)
--	4.4-19 (was 4.4-12)

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3. The affected penetration is isolated within four hours by the use of a closed manual valve or blind flange.

4. The reactor is in the hot shutdown condition within 12 hours and cold shutdown within 24 hours.

3.6.4 The reactor building internal pressure shall not exceed 1.5 psig or five inches of Hg if the reactor is critical.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and tagged.

3.6.6 The combined leakage rate for all penetrations and valves shall be determined in accordance with Specification 4.4.1.2. If, based on the most recent surveillance testing results the combined leakage rate exceeds the specified value and containment integrity is required then, repairs shall be initiated immediately and conformance with specified value shall be demonstrated within 48 hours or the reactor shall be in cold shutdown within an additional 36 hours.

Bases

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the Reactor Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here are conservative since an evaluation of National Weather Service records for this area indicates that from 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest is 30.85 inches of Hg.

Operation with a personnel or emergency hatch inoperable does not impair containment integrity since either door meets the design specifications for structural integrity and leak rate. Momentary passage through the outer door is necessary should the inner door gasket be inoperative to install or remove auxiliary restraint beams on the inner door to allow testing of the hatch. The time limits imposed permit completion of maintenance action and the performance of a local leak rate test when required or the orderly shutdown and cooldown of the reactor. Timely corrective action for an inoperable containment isolation valve is also specified.

When containment integrity is established, the limits of 10CFR100 will not be exceeded should the maximum hypothetical accident occur.

REFERENCES

FSAR, Section 5

4.4 REACTOR BUILDING

4.4.1 Containment Leakage Tests

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leak Rate Tests

4.4.1.1.1 Test Pressure

The periodic integrated leak rate test may be performed at a test pressure of not less than 29.5 psig. The containment leakage rate shall be determined in conformance with the criteria specified in Appendix J of 10CFR50 using the methods and provisions of ANSI N45.4-1972.

4.4.1.1.2 Frequency of Test

After the preoperational leakage rate tests, a set of three Type A tests shall be performed with the unit in a shutdown condition at approximately equal intervals during each 10 year service period. The third test of each set shall be conducted when the plant is shutdown for the 10 year inservice inspections.

4.4.1.1.3 Acceptance Criteria

The overall acceptance containment leakage rate is determined by the preoperational leakage rate test and shall not exceed 0.25 weight percent of containment air per 24 hours at 59 psig. Any leakage in excess of 50% of the total allowed containment leakage shall be demonstrated to be to the penetration room. If the reduced pressure leakage rate 95% Upper Confidence Level (UCL) exceeds 0.75 L_t, a test at peak pressure shall be conducted. If the peak pressure leakage rate 95% UCL exceeds 0.75 La, the test schedule applicable to subsequent Type A tests shall be reviewed and approved by the Commission. If leakage rate 95% UCL during any two consecutive Type A tests exceeds either 0.75 La or 0.75 L_t, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests demonstrate leakage rate 95% UCL is less than 0.75 La or 0.75 L_t, at which time the normal testing schedule may be resumed.

4.4.1.1.4 Accuracy

The accuracy of each Type A test shall be verified by a supplemental test which:

a. Confirms the accuracy of the Type A test by verifying that the absolute difference between supplemental and Type A test data is within 0.25 La or 0.25 L_t, as appropriate.

b. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

c. Requires the quantity of gas bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total leakage rate at Pa (59 psig) or P_t (29.5 psig).

4.4.1.1.5 Report of Test Results

The results of periodic tests shall be the subject of a summary technical report which shall be submitted to the Commission within 90 days of completion of the test.

4.4.1.2 Local Leak Rate Testing

4.4.1.2.1 Scope of Testing

The local leak rate shall be measured for the components listed in Table 4.4-1 in accordance with the criteria specified in Appendix J of 10CFR50.

4.4.1.2.2 Frequency of Test

Local leak rate tests shall be conducted with gas at a pressure of not less than 59 psig during each reactor shutdown for refueling or other convenient interval but in no case at intervals greater than 24 months.

4.4.1.2.3 Acceptance Criteria

The combined leakage rate from all penetrations and isolation valves shall not exceed 0.125 weight percent of the postulated post-accident containment air mass per 24 hours at 59 psig.

4.4.1.3 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.3 and 4.4.1.2.3, respectively.

4.4.1.4 Isolation Valve Functional Tests

Inservice testing of ASME Code Class 1, 2, and 3 valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

4.4.1.5 Containment Air Lock Testing

4.4.1.5.1 Scope of Testing

The Personnel Air Lock and Emergency Air Lock shall be tested as required by the following:

4.4.1.5.2 Frequency of Test

(a) The Personnel Air Lock and Emergency Air Lock shall be tested quarterly at an internal pressure of not less than 59 psig.

(b) Air locks opened during periods when containment integrity is not required shall be tested at the end of such periods by a full hatch leak test at not less than 59 psig. If the full hatch test has been performed within the previous 3 days, the leak test can be performed between the double seal of the outer door at not less than 59 psig.

(c) When containment integrity is required, either a full hatch leak test or a leak test of the outer door double seal will be performed within 3 days of initial opening, and during periods of frequent use, at least once every 3 days. Each leak test will be performed at not less than 59 psig.

4.4.1.5.3 Acceptance Criteria

The acceptance criteria for the air lock leakage test is as stated in Specification 4.4.1.2.3.

Bases

The Reactor Building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 286°F. This corresponds to a post-accident containment atmosphere mass of 5.1277×10^5 lbm. Prior to initial operation, the containment was strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment was also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests verified that the leak rate from Reactor Building pressurization satisfies the relationships given in the specification.

The performance of a periodic integrated leak rate test during unit life provides a current assessment of potential leakage from the containment, in case of an accident. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 29.5 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leak rate and it duplicates the preoperational leak rate test at 29.5 psig. The frequency of the periodic integrated leak rate test is normally keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leak rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.25 percent leakage rate at 59 psig during preoperational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.125 percent) of leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the containment is maintained.

Leakage to the penetration room, which is permitted to be up to 50 percent of the total allowable containment leakage, is discharged through high efficiency particulate air (HEPA) and charcoal filters to the unit vent. The filters are conservatively said to be 90 percent efficient for iodine removal.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the Reactor Building liner due to the mechanical closure involved. Testing of these penetrations is performed with air or nitrogen. The basis for specifying a maximum leak rate of 0.125 percent from penetrations and isolation valves is that one-half of the actual integrated leak rate is expected from those sources. Valve operability tests are specified to assure proper closure or opening of the Reactor Building isolation valves to provide for isolation of functioning of Engineered Safety Features systems.

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

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

REFERENCES

(1) FSAR, Sections 5 and 13.

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

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

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

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
11	Fuel transfer tube	Not Vented	Type B	Note 6a, 11
12	Fuel transfer tube	Not Vented	Type B	Note 6a, 11
13	RB Spray inlet line	Not Vented	None required	Note 5, 7d
14	RB Spray inlet line	Not Vented	None required	Note 5, 7d
15	LPI and DHR inlet line	Not Vented	None required	Note 4, 5
16	LPI and DHR inlet line	Not Vented	None required	Note 4, 5
17	OTSG B Emergency FDW line	Not Vented	None required	Note 5, 7d
18	Quench tank vent line	Note 1	Type C	Note 3, 7b, 9
19	RB purge inlet line	Note 1	Type B	Note 6a, 7a, 7b 9
20	RB purge outlet line	Note 1	Type B	Note 6a, 7a, 7b 9
21	LPSW to RC Pump motors and lube oil coolers inlet	Not Vented	None required	Note 7b, 9

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

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
22	LPSW from RC Pump motors and lube oil coolers outlet	Not Vented	None required	Note 7b, 9
23	RC Pump seal injection	Not Vented	Type C	Note 5, 7d, 9
24	SPARE	Not in Use		
25	OTSG B Feedwater line	Not Vented	None required	Note 5
26	OTSG A Main steam line	Not Vented	None required	Note 5, MS Stop valve leak test performed
27	OTSG A Feedwater line	Not Vented	None required	Note 5
28	OTSG B Main steam line	Not Vented	None required	Note 5, MS Stop valve leak test performed
29	Quench tank drain line	Note 1	Type C	Note 3, 7b, 9
30 31 32	LPSW for RB Cooling units inlet line	Not Vented	None required	Note 5
33 34 35	LPSW for RB cooling units outlet line	Not Vented	None required	Note 5

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

PENETRATION NUMBER	SYSTEM	TYPE A SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
36 37	RB emergency sump recirculation line	Not Vented	None required	Note 5
38	Quench tank cooler inlet line	Note 1	Type C	Note 2, 7d
39	HP Nitrogen supply	Note 1	None required	Note 3 (manual valves)
(Unit 2, 3) Only	CFT Vent line	Note 1	None required	Note 3 (manual valves
40	RB emergency sump drain line	Note 1	None required	
41	Instrument air supply & ILRT verification line	Note 1	None required	Note 3 (manual valves)
42	SPARE	Not in Use		
43	OTSG A drain line	Note 1	None required	Note 7b
44	Component cooling to control rod drive inlet line	Note 1	Type C	Note 3, 7d
45	ILRT instrument line	Not Vented	Type C	Note 3, 7a
46	Reactor head-wash filtered water inlet	Note 1	Type B	Note 3, 6a

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

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

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

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

TABLE 4.4-1
(NOTES)

NOTE 1 All vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment atmosphere and to assure they will be subjected to the test differential pressure.

NOTE 2 Fluid system that is part of the reactor coolant pressure boundary and open directly to the containment atmosphere under post-accident conditions (vented to containment atmosphere during Type A test).

NOTE 3 Closed system inside containment that penetrates containment and postulated to rupture as a result of a loss of coolant accident (vented to containment atmosphere during Type A test).

NOTE 4 System required to maintain the plant in a safe condition during the test (need not be vented).

NOTE 5 System normally filled with water and operating under post-accident condition (need not be vented). Type C test required with report to NRC.

NOTE 6 a. Containment penetration whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetration filled with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

b. Air lock door seals including door operating mechanisms which are part of the containment pressure boundary.

c. Doors with resilient seals or gaskets except for seal welded doors.

d. Components other than those above which must meet the acceptance criteria of Type B tests.

NOTE 7 a. Isolation valves provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves.

b. Isolation valves are required to close automatically upon receipt of a containment isolation signal in response to controls intended to affect containment isolation.

TABLE 4.4-1
NOTES (continued)

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

d. Check valves used for containment isolation.

NOTE 8 DELETED

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

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

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

4.4.2 Structural Integrity

Applicability

Applies to the structural integrity of the Reactor Building.

Objective

To define the inservice surveillance program for the Reactor Building.

Specification

4.4.2.1 Tendon Surveillance

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

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

4.4.2.1.1 Lift-Off

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

4.4.2.1.2 Wire Inspection and Testing

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

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

Should the inspection of one of the wires reveal any significant corrosion (pitting or loss of area), further inspection of the other two sets in that directional group will be made to determine the extent of the corrosion and its significance to the load-carrying capability of the structure. The sheathing filler will be sampled and inspected for changes in physical appearance.

Wire samples shall be selected in such a manner that with the third inspection, wires from all nine surveillance tendons shall have been inspected and tested.

4.4.2.2 Inspection Intervals and Reports

For Unit 1, the initial inspection shall be within 18 months of the initial Reactor Building Structural Integrity Test. The inspection intervals, measured from the date of the initial inspection, shall be two years, four years and every five years thereafter or as modified based on experience. For Units 2 and 3 the inspection intervals measured from the date of the initial structural test shall be one year, three years and every five years thereafter or as modified based on experience. Tendon surveillance may be conducted during reactor operation provided design conditions regarding loss of adjacent tendons are satisfied at all times.

A quantitative analytical report covering results of each inspection shall be submitted to the Commission within 90 days of completion, and shall especially address the following conditions, should they develop.

a. Broken wires.

b. The force-time trend line for any tendon, when extrapolated, that extends beyond either the upper or lower bounds of the predicted design band.

c. Unexpected changes in corrosion conditions or sheathing filler properties.

Bases

Provisions have been made for an in-service surveillance program, covering the first several years of the life of the unit, intended to provide sufficient evidence to maintain confidence that the integrity of the Reactor Building is being preserved. This program consists of tendon, tendon anchorage and liner plate surveillance. The first year tendon anchorage and liner plate surveillance programs have been successfully completed.

To accomplish these programs, the following representative tendon groups have been selected for surveillance:

Horizontal - Three 120° tendons comprising one complete hoop system below grade

Vertical - Three tendons spaced approximately 120° apart.

Dome - Three tendons spaced approximately 120° apart.

The inspection during this initial period of at least one wire from each of the nine surveillance tendons (one wire per group per inspection) is considered sufficient representation to detect the presence of any wide spread tendon corrosion or pitting conditions in the structure. This program will be subject to review and revision as warranted based on studies and on results obtained for this and other prestressed concrete reactor buildings during this period of time.

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 amounts 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. 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 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.