NOVEMBER 0 6 1981

DISTRIBUTION:
Docket File-3
NRC PDR
L PDR
TERA
NSIC
ORB#4 Rdg
DEisenhut
PWagner
RIngram
OELD
AEOD
IE-4
ACRS-10
Gray File +4
RDiggs
CMiles
JWetmore
ASLAB
HOrnstein
EBlackwood
WPaulson
BScharf-10

RECEIVED NOV 13 1981 U.S. NUCLEAR REGULATORY COMMISSION

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. 104, 104, and 101 to Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated November 30, 1976, as supplemented by letters dated October 24 and December 29, 1980, July 24 and September 3, 1981.

These amendments revise the TSs to incorporate the containment penetration testing requirements of Appendix J to 10 CFR Part 50.

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

Sincerely,

Original signed by

Philip C. Wagner, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 104 to DPR-38
2. Amendment No. 104 to DPR-47
3. Amendment No. 101 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

8111250088 811106
PDR ADOCK 05000269
P PDR

E.K. F.R. notice & Amendment only.

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD-OR:DL	OELD	C-CSB:DSI
SURNAME	RIngram/cab	PWagner	JStolz	TNovak	KETCHEN	WButler
DATE	10/21/81	10/23/81	10/21/81	10/31/81	11/2/81	10/30/81

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
iv	iv
vi	vi
3.6-2	3.6-2
--	3.6-3
4.4-1	4.4-1
4.4-2	4.4-2
4.4-3	4.4-3
4.4-4	4.4-4
4.4-5	4.4-5
4.4-6	4.4-6
4.4-7	4.4-7
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)

Section		Page
4.4.1	Containment Leakage Tests	4.4-1
4.4.2	Structural Integrity	4.4-14
4.4.3	Hydrogen Purge System	4.4-17
4.5	EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEMS PERIODIC TESTING	4.5-1
4.5.1	Emergency Core Cooling Systems	4.5-1
4.5.2	Reactor Building Cooling Systems	4.5-6
4.5.3	Penetration Room Ventilation System	4.5-10
4.5.4	Low Pressure Injection System Leakage	4.5-12
4.6	EMERGENCY POWER PERIODIC TESTING	4.6-1
4.7	REACTOR CONTROL ROD SYSTEM TESTS	4.7-1
4.7.1	Control Rod Trip Insertion Time	4.7-1
4.7.2	Control Rod Program Verification	4.7-2
4.8	MAIN STEAM STOP VALVES	4.8-1
4.9	EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING	4.9-1
4.10	REACTIVITY ANOMALIES	4.10-1
4.11	ENVIRONMENTAL SURVEILLANCE	4.11-1
4.12	CONTROL ROOM FILTERING SYSTEM	4.12-1
	(INTENTIONALLY BLANK)	4.13-1
4.14	REACTOR BUILDING PURGE FILTERS AND THE SPENT FUEL POOL VENTILATION SYSTEM	4.14-1
4.15	IODINE RADIATION MONITORING FILTERS	4.15-1
4.16	RADIOACTIVE MATERIALS SOURCES	4.16-1
4.17	STEAM GENERATOR TUBING SURVEILLANCE	4.17-1
4.18	HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS)	4.18-1
4.19	FIRE PROTECTION AND DETECTION SYSTEM	4.19-1
4.20	REACTOR VESSEL INTERNALS VENT VALVES	4.20-1

LIST OF TABLES

Table No.		Page
2.3-1A	Reactor Protective System Trip Setting Limits - Unit 1	2.3-11
2.3-1B	Reactor Protective System Trip Setting Limits - Unit 2	2.3-12
2.3-1C	Reactor Protective System Trip Setting Limits - Unit 3	2.3-13
3.5-1-1	Instruments Operating Conditions	3.5-4
3.5-1	Quadrant Power Tilt Limits	3.5-14
3.17-1	Fire Protection & Detection Systems	3.17-5
4.1-1	Instrument Surveillance Requirements	4.1-3
4.1-2	Minimum Equipment Test Frequency	4.1-9
4.1-3	Minimum Sampling Frequency	4.1-10
4.2-1	Oconee Nuclear Station Capsule Assembly Withdrawal Schedule at Crystal River Unit No. 3	4.2-3
4.4-1	List of Penetrations with 10CFR50 Appendix J Test Requirements	4.4-6
4.11-1	Oconee Environmental Radioactivity Monitoring Program	4.11-3
4.11-2	Offsite Radiological Monitoring Program	4.11-4
4.11-3	Analytical Sensitivities	4.11-5
4.17-1	Steam Generator Tube Inspection	4.17-6
4.18-1	Safety Related Shock Suppressors (Snubbers)	4.18-3
6.1-1	Minimum Operating Shift Requirements with Fuel in Three Reactor Vessels	6.1-6
6.6-1	Report of Radioactive Effluents	6.6-8

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 pre-operational 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 shut-down 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) Note 3, 6a (Unit 2,3) 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 ≥99% DOP removal and ≥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_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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 101 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

1.0 Introduction

By letter dated November 30, 1976, Duke Power Company (Duke or the licensee) submitted an application which proposed revisions to the common Oconee Nuclear Station (ONS) Technical Specifications (TSs) related to the testing requirements of Appendix J to 10 CFR Part 50. By letters dated October 24 and December 29, 1980, Duke submitted revisions and supplements to the above application, but did not include air lock leak rate testing requirements. On October 22, 1980, Appendix J was revised regarding Type B tests of air locks. Duke submitted a supplement to the above application on July 24, 1981, to incorporate air lock leak testing. The NRC provided a preliminary Safety Evaluation Report (SER) of the Appendix J review to Duke by letter dated July 29, 1981. In response to the open items contained in this preliminary SER, Duke submitted a revised application on September 3, 1981, which included a composite resubmittal of the previously proposed TSs.

2.0 Background

Included in the preliminary SER mentioned above, was a Draft Technical Evaluation Report (TER) dated February 1981 provided by the NRC's consultant, Franklin Research Center. The NRC has recently received the final TER on Containment Leakage Rate Testing dated August 6, 1981, (copy attached). The NRC has reviewed this TER and agrees with the findings and conclusions contained therein with the exception of the testing requirements for Penetration 59. In addition, the NRC has reviewed the July 24, 1981, application related to containment airlock testing which was incorporated into the September 3, 1981 resubmittal.

3.0 Evaluation

3.1 Airlocks

By letters dated September 5, 1975, February 15 and September 14, 1977, Duke requested an exemption to the leak testing requirements of Appendix J to 10 CFR Part 50. On October 22, 1980, Appendix J was revised to allow testing of air lock door seals in lieu of full pressure tests for those doors in frequent use, provided full pressure tests are performed at least once each six months.

By letter dated July 24, 1981, Duke submitted an application to include air lock leakage testing requirements in the common ONS TSs. The proposed TS would require that full pressure tests (i.e., Pa=59 psig) be performed quarterly and at the end of periods when containment integrity is not required if the airlock was opened. In addition, the proposal would require that within three days, either a full hatch leak test or a leak test of the outer door double seal at a pressure of 59 psig be performed if the airlock door is opened when containment integrity is required. The NRC has reviewed this proposal and finds it to be in accordance with the requirements of Appendix J to 10 CFR Part 50 and, therefore, to be acceptable.

3.2 Other Penetrations

As mentioned above, the NRC's consultant reviewed the Containment Leakage Rate Testing requirements for the ONS with the exception of the air locks. We have reviewed the attached August 6, 1981, TER and agree with the conclusions contained therein with the exception of Penetration 59, Core Flood Tank sample lines. Other conclusions contained in Section 4 of the TER relate to: 1) leak testing of valves in Penetration 47 for Unit 1, 2) justification for reverse direction testing of certain isolation valves, and 3) the acceptance of the submitted TSs subject to certain corrections.

By letter dated September 3, 1981, Duke provided additional information regarding the conclusions contained in the Draft TER and a complete resubmittal of TSs related to this subject. The NRC evaluation of this submittal is as follows:

1) Duke reevaluated the necessity of performing a Type C test on the Unit 1 valves associated with Penetration 47 (Demineralized Water Supply to Reactor Coolant Pump (RCP) seal vents) and determined that modifications necessary to allow Type C testing were not required. Nevertheless, Duke accepted the NRC's position and committed to modify Penetration 47 on ONS Unit 1 to allow Type C leak testing. The NRC finds this change to be acceptable.

2) The justification for reverse direction testing of certain containment isolation valves has been reviewed and approved by the NRC's Office of Inspection and Enforcement.

3) Modifications to the TSs to incorporate the corrections contained in the Draft TER (and subsequently the TER) were included by Duke in this supplement. The NRC has reviewed the revised TSs and finds them to be acceptable.

Included in the September 3, 1981, submittal is a refutation of the position taken in the TER that the valves associated with Penetration 59 should be Type C leak tested. The basis for this position (TER pages 18 & 19) is that the core flood tank (CFT) sample isolation valves can become a barrier to the escape of containment air when the location of the Loss of Coolant Accident (LOCA) break causes the contents of a tank to be discharged into the containment. In this case, a leaking sample line could allow the CFT nitrogen to be vented such that containment atmosphere can then enter the CFT by leaking through check valve CF-11 or CF-13. Since the isolation valves may be relied upon to prevent the escape of containment air in this situation, Type C testing is required.

Duke's response to this position stated that: "for a postulated break between valves CF-11(-13), CF-12(-14) and LP-47(-48), a core flood tank (CFT) would depressurize to containment but it would most likely not be a LOCA. Check valves CF-12 and -14 tend to seat with RC pressure and would prevent any loss of coolant from occurring. They are also periodically leak checked pursuant to Technical Specifications 3.1.6.10 and 4.5.1.2.3. If the break is postulated to continue, operators would isolate the affected core flood tank (CFT) by closing CF-1, -2 when directed by procedure. With a core flood tank (CFT) isolated, a unit shutdown would then be required by Specifications 3.0 and 3.3.

"If the break were postulated to occur between CF-12(-14) and the reactor vessel, a LOCA would occur and the CFTs would depressurize and Low Pressure Injection would be initiated. As the Reactor Coolant System is depressurized, coolant would flow out the break and make-up would be provided by ECC systems. In all cases it is predicted that ambient pressure in the containment is less than, or at most, equal to system pressure. Furthermore, by the design of the system, this piping is low in the containment relative to the entry point in the vessel. CF-12, -14 are located in vertical runs of piping, just prior to entry into the Reactor Vessel. Also, operators are directed to isolate the depressurized CFTs by closing CF-1, -2. Regardless of where the break is, cooling water would tend to seat CF-11, -13. Thus, regardless of break location, it is not credible to conclude that the CF Tank Sample isolation valves will ever see containment atmosphere following a postulated break that discharges the content of a tank into the containment."

Duke concluded that, based on the above, Type C testing need not be performed on Penetration 59.

The NRC has reviewed Duke's response, and in light of the additional TS surveillance requirements incorporated by Order dated April 20, 1981, agrees with the conclusion that Type C testing need not be performed on Penetration 59.

Based on the above findings, we conclude that the TSs submitted on September 3, 1981, are acceptable and the requirements of Appendix J to 10 CFR Part 50 have been met at the ONS.

By letter dated November 6, 1981, Duke requested a change to the September 3, 1981 application related to the local leak test requirements for Penetrations 21 and 22. These Penetrations serve the Low Pressure Service Water (LPSW) to the Reactor Coolant Pumps motors and lube oil coolers (21-inlet and 22-outlet). The September 3, 1981 application indicates a Type C leakage rate test should be performed on the associated valves (21-LPSW 6 and 22-LPSW 15) which are located outside of the containment. On further review of the leak testing requirements, Duke concluded that Type C testing of these valves was not required since the outboard side of both valves would remain pressurized by the LPSW system throughout a LOCA. We have evaluated the leak testing requirements for these Penetrations and have concluded that sufficient assurance exists that pressurized LPSW will be maintained on the outboard of both of these penetrations to preclude leakage of containment atmosphere. Therefore, we find this proposed change to be acceptable.

4.0 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.

5.0 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.

Attachment: TER

Dated: November 6, 1981

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE RATE TESTING

DUKE POWER COMPANY
OCONEE UNITS 1, 2, AND 3

NRC DOCKET NO. 50-269, 50-270, 50-287

NRC TAC NO. 10114, 10115, 10116

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 33, 34, 35

Prepared by

Franklin Research Center
The Parkway at Twentieth Street
Philadelphia, PA 19103

Author: T. J. DelGaizo

FRC Group Leader: T. J. DelGaizo

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: Y. S. Huang

August 6, 1981

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

8108130163 XA

CONTENTS

Section	Title	Page
1	BACKGROUND.	1
2	EVALUATION CRITERIA	3
3	TECHNICAL EVALUATION	4
	3.1 Licensee-Proposed Exemptions from the Type C Testing Requirements of Appendix J	4
	3.1.1 Penetrations 4, 43 - OTSG B, A Drain Lines. .	4
	3.1.2 Penetrations 8, 9, 52 - Loop A Nozzle Warming Line; High Pressure Injection Lines, A, B . .	6
	3.1.3 Penetrations 13, 14 - Reactor Building Spray Inlet Lines, A, B	7
	3.1.4 Penetrations 15, 16 - Low Pressure Injection and Decay Heat Removal Inlet Lines, A, B . .	9
	3.1.5 Penetrations 17, 50 - OTSG, B, A Emergency FDW Lines	11
	3.1.6 OTSG Feedwater and Steam Penetrations a. Penetrations 25, 27 b. Penetrations 26, 28	12
	3.1.7 Penetrations 30, 31, 32 LPSW for RB Cooling Units Inlet Line; 33, 34, 35 LPSW for RB Cooling Units Outlet Line	13
	3.1.8 Reactor Building Emergency Sump Penetrations	
	a. Penetrations 36, 37	14
	b. Penetration 40.	14
	3.1.9 Penetration 47 (Unit 1 Only) - Demineralized Water Supply to RC Pump Seal Vents.	16

CONTENTS (CONT.)

Section	Title	Page
	3.1.10 Penetration 51 - Leak Rate Test Line	16
	3.1.11 Penetrations 57 (Unit 1), 62 (Units 2, 3) - Decay Heat Removal Return Line	17
	3.1.12 Penetration 59 - CF Tank Sample Line	18
	3.2 Reverse Direction Testing of Isolation Valves. . .	19
	3.3 Proposed Technical Specification Changes	21
4	CONCLUSIONS	23
5	REFERENCES.	25

1. BACKGROUND

On August 4, 1975 [1], the NRC requested that the Duke Power Company (DPC) review the containment leakage testing program at Oconee Units 1, 2, and 3 (Oconee) and provide a plan for achieving full compliance with 10CFR50, Appendix J, Containment Leakage Testing, including appropriate design modifications, changes to technical specifications, or requests for exemption from requirements pursuant to 10CFR50.12, where necessary.

On September 5, 1975 [2], DPC responded to the NRC's request stating that the Oconee Technical Specifications were in compliance with 10CFR50, Appendix J, with one exception. DPC requested an exemption for this deviation pursuant to 10CFR50.12.

On November 30, 1976 [3], DPC reported that subsequent review of the Oconee testing program revealed that certain additional penetrations may be construed to require Type C testing. DPC stated that these penetrations were tested in conjunction with the integrated leak rate test of the reactor building and requested exemption for these penetrations pursuant to 10CFR50.12.

On December 28, 1976 [4], DPC acknowledged that an approved modified method to meet the objective of 10CFR50, Appendix J, for airlock testing was considered to be a suitable alternative for Oconee.

On February 15, 1977 [5], DPC reported that efforts to test airlocks according to the modified method were unsuccessful and requested an exemption to the provision of 10CFR50, Appendix J, as previously requested in Reference 2.

On August 15, 1977 [6], the NRC notified DPC that (1) reverse direction testing of the five penetrations identified by DPC in Reference 3 was acceptable and no exemption was required, (2) more information was required to evaluate the acceptability of not testing seven penetrations so identified in Reference 3, (3) DPC must supply evidence to justify not including 23 specified containment penetrations in the Technical Specifications listing of penetrations requiring local leak rate testing, and (4) an exemption with respect to airlock testing was declined.

On September 14, 1977 [7], DPC provided justification for not testing 31 penetrations (the 30 identified in Reference 6 and penetration number 57) and, following a substantial restatement of the problem associated with compliance with the NRC's position concerning airlock testing, reiterated DPC's position requesting that an exemption in this matter be granted to allow continuation of the existing airlock testing program. Further, on October 24, 1980 [8], DPC submitted to the NRC a proposed revision to its Technical Specifications which supplemented the original submittal of Reference 3.

On December 29, 1980 [9], DPC submitted a supplement to Reference 8, revising part of its Technical Specifications, and stated that the review of containment airlock test procedures is continuing.

The purpose of this report is to provide technical evaluations of all outstanding submittals regarding the containment leakage testing program at Oconee. Since DPC has indicated in Reference 9 that it is reviewing its airlock test program in view of the October 22, 1980 rule change regarding airlock testing, FRC has not included an evaluation of any airlock submittals.

1. BACKGROUND

On August 4, 1975 [1], the NRC requested that the Duke Power Company (DPC) review the containment leakage testing program at Oconee Units 1, 2, and 3 (Oconee) and provide a plan for achieving full compliance with 10CFR50, Appendix J, Containment Leakage Testing, including appropriate design modifications, changes to technical specifications, or requests for exemption from requirements pursuant to 10CFR50.12, where necessary.

On September 5, 1975 [2], DPC responded to the NRC's request stating that the Oconee Technical Specifications were in compliance with 10CFR50, Appendix J, with one exception. DPC requested an exemption for this deviation pursuant to 10CFR50.12.

On November 30, 1976 [3], DPC reported that subsequent review of the Oconee testing program revealed that certain additional penetrations may be construed to require Type C testing. DPC stated that these penetrations were tested in conjunction with the integrated leak rate test of the reactor building and requested exemption for these penetrations pursuant to 10CFR50.12.

On December 28, 1976 [4], DPC acknowledged that an approved modified method to meet the objective of 10CFR50, Appendix J, for airlock testing was considered to be a suitable alternative for Oconee.

On February 15, 1977 [5], DPC reported that efforts to test airlocks according to the modified method were unsuccessful and requested an exemption to the provision of 10CFR50, Appendix J, as previously requested in Reference 2.

On August 15, 1977 [6], the NRC notified DPC that (1) reverse direction testing of the five penetrations identified by DPC in Reference 3 was acceptable and no exemption was required, (2) more information was required to evaluate the acceptability of not testing seven penetrations so identified in Reference 3, (3) DPC must supply evidence to justify not including 23 specified containment penetrations in the Technical Specifications listing of penetrations requiring local leak rate testing, and (4) an exemption with respect to airlock testing was declined.

On September 14, 1977 [7], DPC provided justification for not testing 31 penetrations (the 30 identified in Reference 6 and penetration number 57) and, following a substantial restatement of the problem associated with compliance with the NRC's position concerning airlock testing, reiterated DPC's position requesting that an exemption in this matter be granted to allow continuation of the existing airlock testing program. Further, on October 24, 1980 [8], DPC submitted to the NRC a proposed revision to its Technical Specifications which supplemented the original submittal of Reference 3.

On December 29, 1980 [9], DPC submitted a supplement to Reference 8, revising part of its Technical Specifications, and stated that the review of containment airlock test procedures is continuing.

The purpose of this report is to provide technical evaluations of all outstanding submittals regarding the containment leakage testing program at Oconee. Since DPC has indicated in Reference 9 that it is reviewing its airlock test program in view of the October 22, 1980 rule change regarding airlock testing, FRC has not included an evaluation of any airlock submittals.

2. EVALUATION CRITERIA

Code of Federal Regulations, Title 10 Part 50 (10CFR50), Appendix J, Containment Leakage Testing, contains the basic criteria used for the following evaluation. Recognition that plant-specific conditions can lead to variations not explicitly covered by existing regulations has dictated that this review emphasize the basic intent of Appendix J, that potential containment atmosphere leakage paths be identified, monitored, and maintained below established limits. Where applied in the following evaluation, criteria used have been referenced or briefly stated, as necessary, to support the conclusions.

3. TECHNICAL EVALUATION

In Reference 9, DPC revised its latest proposals regarding containment leakage testing. To facilitate the NRC's review, Reference 9 contained all of DPC's outstanding requests regarding the implementation of Appendix J. Consequently, FRC has reviewed and evaluated only the Reference 9 submittal.

These evaluations have been conducted in three categories:

1. Exemptions from the requirements of Appendix J for testing of containment isolation valves (Type C testing) as provided in Table 4.4-1 of Reference 9.

2. Justification for reverse direction valve testing as provided in Table 4.4-1 of Reference 9.

3. Proposed technical specification changes as included in Reference 9.

3.1 LICENSEE-PROPOSED EXEMPTIONS FROM THE TYPE C TESTING REQUIREMENTS OF APPENDIX J

Exemptions from the Type C testing requirements of Appendix J, which are listed under "Test Requirement Bases" in Reference 9, are evaluated by FRC in the following sections.

3.1.1 Penetrations 4, 43 - OTSG B, A Drain Lines

Licensee Position - "This system can be isolated from the OTSG's and is drained and vented during a Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"The inside containment isolation valves are normally closed manual gate valves. Outside containment isolation valve is a normally closed motor-operated gate valve which receives an ES signal to close. The manual isolation valves provide the containment isolation function but are not required to be tested based on the definition of containment isolation valves in Appendix J, II.H. The ES closure signal to outside isolation valve is provided as a backup method to assure containment isolation. During normal operation, the primary means to assure containment isolation is by having the system valves closed as this system is normally used only when the unit is shutdown and for a limited period of time during the unit heat-up and prior to criticality. Furthermore, the drain lines are connected to a seismically designed system, which does not communicate with the containment, and which operates at conditions well above postulated accident pressure and temperature conditions. Any containment leakage associated with this system would be included in the Type A test. It is considered that a Type C test is neither necessary nor required for this system."

FRC EVALUATION

The Licensee has correctly stated that Section III.A.1.(d) of Appendix J requires Type C testing of the containment isolation valves in the OTSG drain lines. Section II of Appendix J, however, defines containment isolation valves as those valves relied upon to prevent the escape of containment air to the surrounding environment.

Provided these drain lines do not rupture as a result of a LOCA, there is no possibility of leakage of containment atmosphere through these penetrations because the steam generators will be operating at pressures in excess of post-accident containment pressure. If, however, they may rupture as a result of the LOCA (e.g., LOCA missiles or pipe whip), then the isolation valves are relied upon to prevent the escape of containment atmosphere and must be Type C tested. In this case, the fact that the valves are normally shut or are Type A tested is immaterial.

In Reference 8, the Licensee indicated that these drain lines were postulated to rupture after an accident; however, in Reference 9, indication that the lines rupture was omitted without comment. FRC assumes that the omission was based upon a determination by the Licensee that the lines are not liable to rupture as a result of a LOCA.

FRC concludes that the OTSG drain line isolation valves (both the manual valve inside containment and the motor-operated valve outside containment) do not require Type C testing because they are not relied upon to perform a containment isolation function. No exemption from Appendix J is required. This conclusion is valid, provided that the Licensee has determined that the drain lines inside containment are not liable to rupture as a result of a LOCA.

3.1.2 Penetrations 8, 9, 52 - Loop A Nozzle Warming Line; High Pressure Injection Lines, A, B

<u>Licensee Position</u> - "This system is normally filled with water and operating under post-accident conditions. Thus, it need not be drained and vented during the Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"For the Loop A nozzle warming line, the inside containment isolation valve is a normally open stop-check valve. The outside containment isolation valves are normally open stop-check valve in series with a normally throttled needle valve.

"For the HP injection lines, the inside containment valves are a single swing check in series with two parallel stop-check valves. The outside containment valve is a motor-operated globe valve (A loop - normally closed, B loop - normally open) which receives an ES signal to open. These valves do not perform a containment isolation function as defined in Appendix J, II.4 and thus a Type C test need not be performed."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1.(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1.(d) that are relied upon to prevent escape of containment air to the outside require Type C testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1.(d) system which is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

The Loop A nozzle warming lines (penetration 8) and the high head safety injection lines (penetrations 9 and 52) will normally be pressurized with water at a pressure significantly in excess of containment accident pressure (Pa) at the beginning of an accident. However, the high head safety injection does not operate continuously throughout the postaccident period since it may be secured when pressure has been lowered sufficiently to permit low pressure injection. In addition, failure of pump HP-P1C to start would prevent the pressurization of penetration 52.

Should the system be secured before containment pressure is returned to ambient or should pump HP-P1C fail to operate, however, containment atmospheric leakage would be contained within the system outside containment. The escape of containment air from a closed loop outside containment can further be prevented by the opening of remotely controlled valves in the emergency core cooling system (ECCS), which would continuously apply water pressure from the operating low pressure safety injection system to the isolated portion of the high pressure injection system. This, in addition to the fact that a portion of the system is constantly pressurized by the head of the containment recirculation sump, would preclude leakage of containment air through this system.

Consequently, FRC finds that the isolation valves of the high pressure safety injection system (penetrations 8, 9, and 52) do not require Type C testing, not because testing is excluded by Section II.H but because these valves are not relied upon to prevent the escape of containment air in accordance with Sections II.B, II.D, and II.H.

3.1.3 Penetrations 13, 14 - Reactor Building Spray Inlet Lines, A, B

Licensee Position - "Reactor Building spray system is normally filled with water and operating under post-accident conditions and thus, need not be drained and vented during the Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1.(d).

"The inside containment valve is a tilting disc check valve. Outside containment valve is a normally closed motor-operated globe valve which receives an ES signal to open. These valves do not perform a containment isolation function as defined in Appendix J, II.H and thus a Type C test need not be performed."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1.(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1.(d) that are relied upon to prevent escape of containment air to the outside require Type C testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1.(d) system which is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

Each reactor building (RB) spray system at Oconee consists of two independent loops, each of which delivers water from the borated water storage tank (BWST) or RB emergency sump to the containment spray nozzles. Normally, both loops will be in operation following an accident in which containment pressure exceeds atmospheric pressure. When the loops are in operation, water at pressures greater than Pa will be delivered to the containment, which will prevent any possible out-leakage of containment air through this piping system. Should this system be intermittently operated after an accident or if one loop were to fail (e.g., failure of a pump motor), then there is a question of potential leakage of containment air to the surrounding environment.

Should the system be secure before containment pressure is returned to ambient or should one of the two loops fail to operate, however, containment atmospheric leakage would be contained within the system outside containment. The escape of containment air from the closed loop outside containment can be further prevented by the opening of remotely controlled valves in the ECCS, which would continuously apply water pressure from the operating low pressure safety injection system to the isolated portion of the spray system. This, in addition to the fact that a portion of the spray system is constantly pressurized by the head of the containment recirculation sump, would preclude leakage of containment air through this system.

Consequently, FRC finds that the isolation valves of the RB spray system do not require Type C testing, not because testing is excluded by Section II.H, but because these valves are not relied upon to prevent the escape of containment air in accordance with Sections II.B, II.D, and II.H.

3.1.4 Penetrations 15, 16 - Low Pressure Injection and Decay Heat Removal Inlet Lines, A, B

<u>Licensee Position</u> - "This system is required to be filled with water to maintain the plant in a safe condition during the Type A test. Additionally, this system is normally filled with water and operating under post-accident conditions. Thus, it need not be drained and vented during the Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1.(d).

"The inside containment valve is a swing check valve. The outside containment valve is a normally closed motor-operated gate valve which receives an ES signal to open. These valves do not perform a containment isolation function as defined in Appendix J, II.H and thus a Type C test need not be performed."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type

C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1.(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1(d) that are relied upon to prevent escape of containment air to the outside require Type C Testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1(d) system which is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

The low pressure coolant injection system consists of two injection headers being supplied by three pumps. Once initiated following an accident, this system will remain operational throughout both the injection phase and the long-term postaccident cooling recirculation phase. Furthermore, there is no single active failure which can prevent the operation of the system. In the worst-case scenario, with one of the two motor-operated injection valves failing to open (LP-V4A or LP-V4B), the piping will still be water-pressurized by the operating pump or pumps. In any event, there is no potential for leakage of containment air to atmosphere through penetrations 15 or 16 because of the presence in the lines of water at a pressure greater than Pa.

Consequently, FRC finds that these isolation valves do not require Type C testing, not because testing is excluded by Section II.H, but because the valves are not relied upon to prevent the escape of containment air in accordance with Sections II.B, II.D, and II.H.

3.1.5 Penetrations 17, 50 - OTSG, B, A Emergency FDW Lines

Licensee Position - "This system is normally filled with water and operating under post-accident conditions, and thus, need not be drained and vented during the Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"The inside containment valve is a tilting disc check valve. The outside containment valves are a tilting disc check valve in series with a normally closed pneumatically opened gate valve. These valves do not perform a containment isolation function as defined in Appendix J, II.H and thus a Type C test need not be performed."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1(d) that are relied upon to prevent escape of containment air to the outside require Type C testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1(d) system which is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

The emergency feedwater (EFW) system is a safety-related system designed to provide steam generator feedwater following an accident. The design of this system is such that it is capable of providing feedwater, at pressure higher than Pa, to the steam generators despite a possible single active failure. In addition, in the unlikely event that the plant is cooled down and EFW is secured before containment pressure is reduced to ambient, the system represents a closed loop inside containment which does not communicate with the containment atmosphere.

Consequently, FRC finds that the isolation valves in penetrations 17 and 50 are not relied upon to prevent the escape of containment air after an accident, and therefore Appendix J does not require testing. No Appendix J exemption is necessary.

3.1.6 OTSG Feedwater and Steam Penetrations

a. Penetrations 25, 27 - OTSG B, A Feedwater Lines

Licensee Position - "The OTSG is required to be filled with water to maintain it in a safe condition during the Type A test and thus, the feedwater lines cannot be drained and vented. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"No inside containment isolation valves exist. The outside containment valve is a tilting disc check valve. The feedwater lines are connected to a seismically designed system which does not communicate with the containment atmosphere. The feedwater lines are seismically qualified through the outside containment valve. It is not postulated that this system will rupture during a postulated LOCA condition. However, even if it were to rupture, the operating pressure and temperature are well above that expected in the containment. Thus, it is considered that a Type C test is neither necessary nor required for this system."

b. Penetrations 26, 28 - OTSG B, A Main Steam Lines

Licensee Position - "The OTSG is required to be filled with water to maintain it in a safe condition during the Type A test and thus, the main steam line is not vented. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"No inside containment isolation valves exist. The outside containment valves are two electro-hydraulic turbine stop valves in parallel per main steam line. The steam lines are connected to the seismically designed system which does not communicate with the containment atmosphere. The steam lines are seismically qualified through the stop valves. It is not postulated that this system will rupture during the postulated LOCA condition. However, even it were to rupture, the operating pressure and temperature are well above that expected in the containment. Thus, it is considered that a Type C test is neither necessary nor required for this system."

FRC EVALUATION

FRC concurs with the Licensee's analysis that Appendix J does not require testing of these lines because they are part of a closed system which does not communicate with the reactor coolant pressure boundary or the containment atmosphere and which is not liable to rupture as result of a LOCA.

3.1.7 Penetrations 30, 31, 32 LPSW for RB Cooling Units Inlet Line
33, 34, 35 LPSW for RB Cooling Units Outlet Line

Licensee Position - "This system is normally filled with water and operating under post-accident conditions and, thus, need not be drained and vented during the Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"No inside containment isolation valves exist. The outside containment valve is normally open motor-operated gate valve which also receives an ES signal to open. These valves do not perform a containment isolation function as defined in Appendix J, II.H and, thus, a Type C test need not be performed."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1.(d) that are relied upon to prevent escape of containment air to the outside require Type C testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1.(d) system which is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

The reactor building closed cooling water system forms a closed system inside containment which is designed to operate throughout the postaccident period. Consequently, the isolation valves of this system are not relied upon to perform a postaccident containment isolation function as defined by Appendix J and therefore do not require testing. No exemption from Appendix J requirements is necessary.

3.1.8 Reactor Building Emergency Sump Penetrations

a. Penetrations 36, 37 - Reactor Building Emergency Sump Recirculation Line

Licensee Position - "This system is normally filled with water and operating under post-accident conditions and, thus, need not be drained and vented during the Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"No inside containment isolation valves exist. The outside containment valve for each penetration is normally closed motor-operated gate valve. This valve does not perform a containment isolation function as defined in Appendix J, II.H and, thus, a Type C test need not be performed."

b. Penetration 40 - RB Emergency Sump Drain Line

Licensee Position - "This system is drained and vented during a Type A test. During postulated accident conditions, the RB Emergency Sump contains water but this line would not be in operation. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"No inside containment isolation valves exist. All inside containment piping is imbedded in concrete. The outside containment valves are two normally closed manual gate valves in series. Any containment leakage associated with this system would be included in the Type A test. Therefore, it is considered that the additional Type C test is not necessary."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1.(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1.(d) that are relied upon to prevent escape of containment air to the outside require Type C testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1.(d) system which is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

Following any accident in which the containment is pressurized with radioactive air, the RB emergency sump will be filled with water by the ECCS system. Recirculation lines and drain lines will therefore remain water covered throughout the postaccident period. This water seal precludes the escape of containment air to the outside atmosphere, and therefore the isolation valves are not containment isolation valves as defined by Appendix J. Consequently, these valves do not require Type C testing and no exemption is required.

3.1.9 Penetration 47 (Unit 1 Only) - Demineralized Water Supply to RC Pump Seal Vents

Licensee Position - "This system is drained and vented during a Type A test. A Type C test is required for containment isolation valves by Appendix J, III.A.1(d).

"Both the inside and outside containment valves are tilting disc check valves. Any containment leakage associated with this system would be included in the Type A test. Therefore, it is considered that the additional Type C test is not necessary."

FRC EVALUATION

Generally, Type A testing is not an adequate substitute for Type C testing for two reasons:

1. Type C testing is performed twice as often as Type A testing.

2. Type C testing tests valves individually, whereas Type A testing tests penetrations (i.e., two shut valves in series). This is necessary to ensure that, when one isolation valve fails to shut following an accident, the penetration is adequately isolated. Type A testing is insufficient for this purpose since the leaktightness of the penetration is established by the more leaktight of the two shut valves.

In view of the foregoing, FRC finds that Type C testing of these valves is required.

3.1.10 Penetration 51 - Leak Rate Test Line

Licensee Position - "This air system is vented during the Type A test. Draining of fluids is not required. A Type B test is also required by Appendix J, III.B.

"The inside containment device is a gasketed blind flange which is removed only to perform the Type A test. The outside containment valve is a normally closed air-operated Saunders diaphragm valve. During the performance of the Type A test, this valve is closed and the connecting line vented. Any containment leakage associated with this system would be included in the Type A test. Therefore, it is considered that the additional Type B test is not necessary."

FRC EVALUATION

In view of the comparison of integrated leakage testing (Type A) and local leakage testing (Type B or C) given in the FRC evaluation in Section 3.1.9, and since this penetration includes a blind flange and single valve (tested by Type A test) which is used only during Type A testing, FRC concurs with the Licensee's conclusion that Type A testing of this penetration is sufficient for the purposes of Appendix J.

3.1.11 Penetrations 57 (Unit 1), 62 (Unit 2, 3) - Decay Heat Removal Return Line

Licensee Position - "This system is required to be filled with water to maintain the plant in a safe condition during the Type A test. Additionally this system is normally filled with water and operating under post-accident conditions. Thus, it need not be drained and vented during the Type A test. A Type test is required for containment isolation valves by Appendix J, III.A.1(d).

"The inside containment valves are two normally closed motor-operated gate valves in series. The outside containment valve is a normally closed motor-operated gate valve. These valves do not perform a containment isolation function as defined in Appendix J, II.H and, thus, a Type C test need not be performed."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1.(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1.(d) that are relied upon to prevent escape of containment air to the outside require Type C testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1.(d) system that is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

The decay heat removal return line is directly connected to the low pressure coolant injection (LPCI) system which will be in operation throughout the entire postaccident period as discussed in the FRC evaluation in Section 3.1.4. Leakage of containment air through this penetration is prevented by the water seal created by the operating LPCI system. Consequently, Type C testing is not required by Appendix J because the isolation valves are not relied upon to perform a containment isolation function.

3.1.12 Penetration 59 - CF Tank Sample Line

> Licensee Position - "This system is vented and drained during the Type A test. A Type C test is also required for containment isolation valves by Appendix J, III.A.1(d).
>
> "The inside containment valves are two normally closed motor-operated gate valves in parallel, one to each core flood tank. The outside containment valves are two normally closed manual globe valves in parallel. Any containment leakage associated with this system would be included in the Type A test. Furthermore, these valves do not perform a containment isolation function as defined in Appendix J, II.H, and thus, it is considered that a Type C test need not be performed."

FRC EVALUATION

The Licensee's position implies that valves which do not perform a containment isolation function as defined in Section II.H do not require Type

C testing. FRC does not agree with this interpretation of the Type C testing requirements of Appendix J.

Section II.H defines Type C testing as the measurement of containment isolation valve leakage rates. This section further describes four types of valves which are included as containment isolation valves. Section III.A.1.(d) also identifies systems for which the containment isolation valves must be Type C tested.

Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function. Combined with the definition of leakage in Section II.D, containment isolation valves may be further described as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, the valves of Section II.H or Section III.A.1.(d) that are relied upon to prevent escape of containment air to the outside require Type C testing.

One of the obvious differences between FRC's interpretation of these requirements and the Licensee's interpretation is that FRC would conclude that a normally shut isolation valve in a Section III.A.1.(d) system that is relied upon to prevent leakage of containment air to the outside must be Type C tested, whereas the Licensee would conclude that testing is not required.

Core flood tank (CFT) sample isolation valves can become a barrier to the escape of containment air when the location of the LOCA break causes the contents of a tank to be discharged into the containment. In this case, a leaking sample line will allow the CFT nitrogen to be vented such that containment atmosphere can then enter the CFT by leaking through check valve CF-11 or CF-13.

Since the isolation valves may be relied upon to prevent the escape of containment air in this situation, Type C testing is required.

3.2 REVERSE DIRECTION TESTING OF ISOLATION VALVES

In Table 4.4-1 of Reference 9, DPC lists 14 penetrations for which reverse direction testing is planned. A justification for this testing is provided for each penetration.

In each case, test connections for inboard and outboard penetration isolation valves exist between the valves so that the inboard valve is tested in the reverse direction. Also, Type A test requirements for each penetration are fully met. DPC apparently believes that, because the measured leakage when pressurizing between the two valves results in a leakage rate for both valves, the test is conservative.

FRC EVALUATION

Type A procedures test containment penetrations, i.e., penetrations with two shut isolation valves in series. Compared to Type A testing, DPC's procedure is a conservative measurement of penetration leakage. Type C procedures, however, test individual isolation valves. In this case, DPC's procedure is not necessarily conservative.

Appendix J permits reverse direction testing of an isolation valve when it can be determined that leakage rates measured in the reverse direction are equivalent to or more conservative than leakage rates measured in the direction of accident pressure for that particular valve. This determination, therefore, is contingent upon the type of valve and possibly the design of the particular valve as well. Once the Licensee has made a determination that reverse direction testing is equivalent to or more conservative than testing in the direction of accident pressure for a particular valve, reverse direction testing is authorized by Appendix J. No report to the NRC is necessary nor is a request for exemption necessary. However, the Licensee must be prepared to justify the determination, if so requested.

In view of the foregoing, FRC does not concur with the justification presented by the Licensee in Reference 9 for reverse direction testing of these valves. The acceptability of reverse direction testing, however, remains a matter for Licensee determination.

To assist the Licensee in these determinations, the following observations relative to commonly encountered valves are provided based upon FRC's experience in reviewing containment leakage testing submittals from various operating reactors:

1. Gate valves - generally not capable of reverse direction testing because the seating surfaces relied upon to prevent accident leakage are not tested by reverse direction pressure.

2. Globe valves - where reverse direction testing tends to unseat the valve, testing may be considered conservative.

   Where reverse direction testing tends to seat the valve, testing may still be considered equivalent if the seating force exerted by the valve stem (with normal torque applied) is substantially larger than the seating force exerted by the test pressure.

3. Butterfly valves - generally, measured leakage is independent of the direction of test pressure both from a force-exerted standpoint and a seating-surface standpoint.

4. Stop-check valves - generally, reverse direction testing is conservative although an evaluation of differential forces may be appropriate for certain valves.

5. Ball/plug valves - generally not capable of reverse direction testing for reasons similar to those in the gate valve discussion above.

6. Diaphragm valves - often similar to globe valves but require evaluation on a case-by-case basis.

3.3 PROPOSED TECHNICAL SPECIFICATION CHANGES

In Reference 9, DPC provided proposed revisions to Sections 3.6.6 and 4.4.1 of the Technical Specifications for the Oconee plants. These sections provide for the pressure, frequency, and acceptance criteria, and the accuracy and reporting requirements of the integrated leak rate test; the scope of testing, frequency, and acceptance criteria of the local leak rate tests; reactor building modification requirements; and isolation valve functional test requirements.

FRC EVALUATION

Subparagraph 4.4.1.2.1 (Scope of Testing) requires that local leak rate tests be performed in accordance with Appendix J with the exception of the exemptions from Appendix J noted in Table 4.4-1. FRC's evaluations of these proposed exemptions are provided in Section 3.1 of this report.

Subject to the evaluations of Section 3.1 of this report regarding DPC's proposed exemptions of Table 4.4-1, FRC finds that the proposed Technical Specifications are in conformance with the requirements of Appendix J and are therefore acceptable.

4. CONCLUSIONS

Technical evaluations of Licensee-proposed exemptions from the requirements of Appendix J, justifications for continued reverse direction testing, and proposed Technical Specification changes for Oconee as submitted in Reference 9 have resulted in the following conclusions:

- Proposed exemptions for the following isolation valves identified in Table 4.4-1 have been found unacceptable. These valves should be tested in accordance with Appendix J:

  Penetration 47 (Unit 1 only) - Demineralized water supply to RC pump seal vents
  Penetration 59 - CF tank sample lines.

- Proposed exemptions for the following isolation valves are not necessary because Type C testing is not required by Appendix J:

  Penetrations 4, 43 - OTSG B, A drain lines

  Penetrations 8, 9, 52 - Loop A nozzle warming line; high pressure injection lines, A, B

  Penetrations 13, 14 - Reactor building spray inlet lines, A, B

  Penetrations 15, 16 - Low pressure injection and decay heat removal inlet lines, A, B

  Penetrations 17, 50 - OTSG B, A emergency FDW lines

  Penetrations 25, 27 - OTSG B, A feedwater lines

  Penetrations 26, 28 - OTSG B, A main steam lines

  Penetrations 30, 31, 32 - LPSW for RB cooling units inlet lines

  Penetrations 33, 34, 35 - LPSW for RB cooling units outlet lines

  Penetrations 36, 37 - Reactor building emergency sump recirculation line

  Penetration 40 - RB emergency sump drain line

  Penetration 51 - Leak rate test line

  Penetration 57 (Unit 1), 62 (Units 2, 3) - Decay heat removal line

- Justification for reverse direction testing of certain isolation valves provided in Reference 9 was found to be insufficient. The acceptability of reverse direction testing in accordance with Appendix J remains a matter for Licensee determination.

- Proposed Technical Specification changes submitted in Reference 9 were found to be acceptable subject to modification of Table 4.4-1 in accordance with the findings of this report regarding exemption of isolation valves for penetrations 47 (Unit 1 only) and 59.

5. REFERENCES

1. K. Goller (NRC)
   Letter to W. O. Parker, Jr. (DPC)
   August 4, 1975

2. W. O. Parker, Jr. (DPC)
   Letter to R. Boyd (NRC)
   September 5, 1975

3. W. O. Parker, Jr. (DPC)
   Letter to B. Rusche (NRC)
   November 30, 1976

4. W. O. Parker, Jr. (DPC)
   Letter to B. Rusche (NRC)
   December 28, 1976

5. W. O. Parker, Jr. (DPC)
   Letter to B. Rusche (NRC)
   February 15, 1977

6. A. Schwencer (NRC)
   Letter to W. O. Parker, Jr. (DPC)
   August 15, 1977

7. W. O. Parker, Jr. (DPC)
   Letter to E. Case (NRC, NRR)
   September 14, 1977

8. W. O. Parker, Jr. (DPC)
   Letter to R. W. Reid (NRC, ORB)
   October 24, 1980

9. W. O. Parker, Jr. (DPC)
   Letter to R. W. Reid (NRC, ORB)
   December 29, 1980

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. 104,104 and 101 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the Technical Specifications (TSs) for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments revise the common Oconee Nuclear Station TSs to incorporate the containment penetration testing requirements of Appendix J to 10 CFR Part 50.

The application for the amendments complies 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 application for amendments dated November 30, 1976, as supplemented by letters dated October 24 and December 29, 1980, and July 24 and September 3, 1981, (2) Amendments Nos. 104, 104, and 101 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 6th day of November 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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