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TO: Mr. Benard C. Rusche

FROM: Duke Power Company, Charlotte, North Carolina, Mr. William O. Parker, Jr.

DATE OF DOCUMENT: 11/30/76

DATE RECEIVED: 12/6/76

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PROP

INPUT FORM

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DESCRIPTION

Ltr. re our 8/4/75 and 8/14/75 ltrs. and their 9/5/75 ltr. w/attached....notorized 11/30/76....furnishing required tech spec revisions to assure the proper inplementation of 10CFR50, Appendix J at Oconee.

(13-P)

PLANT NAME:

Oconee Units 1-2-3

ENCLOSURE

ACKNOWLEDGED

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FOR ACTION/INFORMATION 12/7/76 RJL

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CONTROL NUMBER: 12316

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

November 30, 1976

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

In an August 4, 1975 letter from Mr. Karl R. Goller, NRC/ONRR, it was stated that the requirements of 10CFR50, Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" had been published on February 14, 1973. Since many nuclear plants had received either an operating license or their containments had reached an advanced stage of design or construction prior to the issuance of Appendix J, it is possible that they may not be in full compliance with Appendix J, even though they are adhering to the provisions of their Technical Specifications.

In our review of the Oconee Nuclear Station in response to the August 14, 1975 letter, the only discrepancy identified was the testing of the personnel and emergency hatches. A request for exemption to the provisions of 10CFR50, Appendix J was requested in that regard by our letter dated September 5, 1975.

Subsequent review of the Oconee Nuclear Station has revealed that certain additional penetrations may be construed to require Type C testing by the regulations. Accordingly, pursuant to the provisions of 10CFR50, §50.90 the attached Technical Specification revisions are requested to assure the proper implementation of 10CFR50, Appendix J at Oconee. Limits on containment leakage have been moved to Specification 3.6 which is a Limiting Condition for operation. Specification 4.4 "Reactor Building" has also been rewritten to more closely agree with the requirements of 10CFR50, Appendix J.

Since the Oconee Nuclear Station was virtually complete at the time of publication of the 10CFR50, Appendix J, there are penetrations in which full compliance with Appendix J is not possible. Specifically, penetrations exist in which Type C tests cannot be performed due to the absence of test connections or isolation valves. Other penetrations must be tested in a direction opposite to that which would be seen during an accident. A listing of penetrations which cannot be tested, or can only be tested in the reverse direction is shown in Table 4.4-1 of the proposed Technical Specification revision. Pursuant to 10CFR50, §50.12 exemptions are requested from the provisions of 10CFR50, Appendix J in these cases. These penetrations are now tested in conjunction with the integrated leak rate test of the Reactor

12316

Building. Modifications necessary to permit testing of these penetrations on a refueling basis are not considered to provide any significant improvement in the effectiveness of the containment system or to increase the protection now provided for the public health and safety.

Very truly yours,

William O. Parker, Jr.

MST:ge
Attachment

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, -47 and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker

William O. Parker, Jr., Vice President

ATTEST:

Lewis F. Camp, Jr.

Lewis F. Camp, Jr.
Assistant Secretary
(Seal)

Subscribed and sworn to before me this 30th day of November, 1976.

Edna B. Farmer

Notary Public
(Notarial Seal)

My Commission Expires:

October 24, 1977

PROPOSED

TECHNICAL SPECIFICATION

REVISIONS

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.

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

3.6.6 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.

3.6.7 The overall containment integrated leakage rate shall not exceed:

a. 0.25 weight percent of containment air per 24 hours at 59 psig. (La)

b. 0.0775 (Unit 1) weight percent of containment
0.176 (Unit 2)
0.176 (Unit 3)
air per 24 hours at 29.5 psig (L_t)

3.6.8 The combined leakage rate of all penetrations and isolation valves shall not exceed 0.125 weight percent of the containment air per 24 hours at 59 psig.

3.6.9 In the event the combined leakage of all penetrations and isolation valves exceeds Specification 3.6.8, based on the most recent surveillance testing results, while containment integrity is required, repairs shall be initiated immediately and conformance with Specification 3.6.8 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 of 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 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.

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 shall 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 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 acceptable containment leakage rate is specified in Specification 3.6.7. If the reduced pressure leakage rate exceeds 0.75 L_t, a test at peak pressure shall be conducted. If the peak pressure leakage rate exceeds 0.75 La the test schedule applicable to subsequent Type A tests shall be reviewed and approved by the Commission. If leakage during any two consecutive Type A tests exceeds either 0.75 La or 0.75 L_t, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests demonstrate leakage 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 difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t.

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 injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total leakage rate at P_a (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 with the exception of the exemptions from the provisions of the Code noted on Table 4.4-1.

4.4.1.2.2 Frequency of Test

Local leak 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 with the exception of the exemptions from the provisions of the Code noted on Table 4.4-1.

4.4.1.2.3 Acceptance Criteria

The combined leakage rate from all penetrations and isolation valves shall not exceed that permitted by Specification 3.6.8.

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

Quarterly (for Units 2 and 3), remotely-operated Reactor Building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during unit operation. The latter valves shall be tested during each refueling shutdown.

Bases

The Reactor Building is designed for an internal pressure of 59 psig and a steamair mixture temperature of 286°F. Prior to initial operation, the containment is strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment is also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests verify 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 that would pressurize the interior of the containment, 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 with the exception of electrical penetrations which are tested with SF_6. 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.

Table 4.4.1

LOCAL LEAK RATE TESTS

Penetration No.	Valve No.	Description	Exemption to 10CFR50 Appendix J
		Personnel Hatch	Test every 4 months - if unopened, every 12 months
		Emergency Hatch	Test every 4 months - if unopened, every 12 months
		Equipment Hatch Seals	
		Fuel Transfer Tube Seals	
		Electrical Penetrations	

Penetration No.	Valve No.	Description	10CFR50 Exemption to Appendix J
1	RC-5	Pressurizer Steam Sample (Inside)	
	RC-6	Pressurizer Water Sample (Inside)	
	RC-7	Pressurizer Sample (Outside)	
2	FDW-105	OTSG A Sample (Inside)	
	FDW-106	OTSG A Sample (Outside)	
3	CC-20	Component Cooling Inlet (Inside)	
	CC-24	Component Cooling Inlet (Outside)	
4	G-23	B OTSG Drain (Inside)	Valves will not be leak tested.
	FDW-104	B OTSG Drain (Outside)	Valves will not be leak tested.
5	LWD-1	RB Normal Sump	Valve will be tested in reverse direction (Unit 2,3)
	LWD-2	RB Normal Sump	Valve will be tested in reverse direction (Unit 1)
6	HP-3	A Letdown Cooler (Inside)	
	HP-4	B Letdown Cooler (Inside)	
	HP-5	Letdown Isolation (Outside)	
7	HP-20	RC Pump Seal Return (Inside)	Valve will be tested in reverse direction.
	HP-21	RC Pump Seal Return (Outside)	
18	GWD-12	Quench Tank Vent Line (Inside)	Valve will be tested in reverse direction.
	GWD-13	Quench Tank Vent Line (Outside)	
19	PR-6	R.B. Purge Inlet (Inside)	
	PR-5	R.B. Purge Inlet (Outside)	
20	PR-1	R.B. Purge Outlet (Inside)	
	PR-2	R.B. Purge Outlet (Outside)	
21	LPSW-6	RC Pump Oil Cooler Inlet	
22	LPSW-15	RC Pump Oil Cooler Outlet	

Penetration No.	Valve No.	Description	10CFR50 Exemption to Appendix J
29	CS-5	Quench Tank Drain Line (Inside)	Valve will be tested in reverse direction.
	CS-6	Quench Tank Drain Line (Outside)	
41	IA-90	Instrument Air to Reactor Bldg (Outside)	Valve will not be leak tested.
	IA-91	Instrument Air to Reactor Bldg (Inside)	Valve will not be leak tested.
43	G-23	A OTSG Drain (Inside)	Valve will not be leak tested.
	FDW-103	A OTSG Drain (Outside)	Valve will not be leak tested.
49	N-110	NP Nitrogen Supply (Outside)	Valve will not be leak tested.
	N-116	LP Nitrogen Supply (Inside)	Valve will not be leak tested.
	N-119	LP Nitrogen Supply (Inside)	Valve will not be leak tested.
	N-174	LP Nitrogen Supply (Inside)	Valve will not be leak tested.
54	CC-7	Component Cooling Outlet (Inside)	Valve will be tested in reverse direction.
	CC-8	Component Cooling Outlet (Outside)	
58	FDW-107	OTSG B Sample (Inside)	
	FDW-108	OTSG B Sample (Outside)	
60	PR-7	R.B. Radiation Monitor (Inside)	
	PR-8	R.B. Radiation Monitor (Outside)	
61	PR-9	R.B. Radiation Monitor (Inside)	
	PR-10	R.B. Radiation Monitor (Outside)	
38	CS-12	Quench Tank Cooler Inlet (Inside)	
	CS-11	Quench Tank Cooler Outlet (Outside)	
39	CF-10	Nitrogen to CFT (Inside)	Valve will not be leak tested.
	N-121	Nitrogen to Pressurizer (Inside)	Valve will not be leak tested.
	N-130	H.P. Nitrogen Supply (Outside)	Valve will not be leak tested.
	CA-29	Chemical Addition to CFT (Outside)	Valve will not be leak tested.
40	LWD-99	R.B. Emergency Sump Drain	Valve will not be leak tested.
	LWD-103	R.B. Emergency Sump Drain	Valve will not be leak tested.

Penetration No.	Valve No.	Description	10CFR50 Exemption to Appendix J
44	CC-77	CC to CRD (Inside)	
	CC-76	CCto CRD (Outside)	
46	FW-64	Reactor Head Wash System (Outside)	Valve will not be leak tested.
	FW-65	Reactor Head Wash System (Inside)	Valve will not be leak tested.
47	DW-155	RCP Seal Vents (Outside)	Valve will not be leak tested.
	DW-156	RCP Seal Vents (Inside)	Valve will not be leak tested.
48	BA-5	Breathing Air to R.B. (Outside)	Valve will not be leak tested.
	BA-33	Breathing Air to R.B. (Inside)	Valve will not be leak tested.
53	CF-8	Nitrogen to CFT	Valve will not be leak tested.
	N-128	Nitrogen Supply	Valve will not be leak tested.
	CA-127	Chemical Addition to CFT	Valve will not be leak tested.
56	SF-61	S.F. Canal Fill and Drain (Inside)	Valve will not be leak tested.
	SF-60	S.F. Canal Fill and Drain (Outside)	Valve will not be leak tested.
59	CF-3	C.F. Sample Drain	Valve will not be leak tested.
	CF-4	C.F. Sample Drain	Valve will not be leak tested.
	CF-19	C.F. Sample	Valve will not be leak tested.
	CF-38	C.F. Drain	Valve will not be leak tested.
55	DW-59	Demineralized Water Supply (Outside)	Valve will not be leak tested.
	DW-60	Demineralized Water Supply (Inside)	Valve will not be leak tested.
45	LRT-24	Leak Rate Test Valve (Inside)	Valve will not be leak tested.
	LRT-25	Leak Rate Test Valve (Outside)	Valve will not be leak tested.
	LRT-39	Leak Rate Test Valve (Inside)	Valve will not be leak tested.
	LRT-38	Leak Rate Test Valve (Outside)	Valve will not be leak tested.
51	LRT-17	Leak Rate Test Valve (Outside)	Valve will not be leak tested.