Attachment I
Technical Specification Mark-ups
and
Revised Originals

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

, AS REQUIRED BY 10 CFR 50.54(o) AND 10 CFR 50, APPENDIX J, OPTION B, INCLUDING ANY APPROVED EXEMPTIONS,

~~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 or the mass-plot method.*~~

GUIDELINES OF REGULATORY GUIDE 1.163, DATED SEPTEMBER, 1995.

~~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~~.

4.4.1.1.1 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 shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests demonstrate leakage rate 95% UCL is less than 0.75 La or 0.75 L_t, at which time the normal testing schedule may be resumed~~.

~~4.4.1.1.4 Accuracy~~

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

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

~~*The mass-plot method may be used for each unit until the presently proposed changes to Appendix J (51 FR 39538) become effective. Thereafter, the licensee shall comply with the provisions of such rule.~~

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 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 containment penetrations in accordance with the criteria specified in Appendix J of 10CFR50, OPTION A

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.

NO CHANGES; INFO ONLY

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 semiannually 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 x 105 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.

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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 3.8.1.7.4, 6.2.4, and 14

(2) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50, Appendix J", Revision 0; July 26, 1996

(3) Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program"; September 1995.

(4) NUREG 1493, "Performance-Based Containment Leak-Test Program", Revision 0, September 1, 1995.

INSERT FOR PREVIOUS PAGE:

The NRC approved an amendment to 10CFR50, Appendix J, "Leak Rate Testing of Containment of Light-Water-Cooled Nuclear Power Plants", to implement a performance-based option for leakage testing of containment.

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Specification

4.4.1.1 Integrated Leak Rate Tests

The containment leakage rate shall be determined, as required by 10CFR50.54 (o) and 10CFR50, Appendix J, Option B, including any approved exemptions, using the guidelines of Regulatory Guide 1.163, dated September, 1995.

4.4.1.1.1 Acceptance Criteria

The overall acceptance containment leakage rate is determined by the preoperational leakage rate test and shall not exceed 0.25 weight percent of containment air per 24 hours at 59 psig. Any leakage in excess of 50% of the total allowed containment leakage shall be demonstrated to be to the penetration room.

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 containment penetrations in accordance with the criteria specified in Appendix J of 10CFR50, Option A.

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 4.4.1.1.1 and 4.4.1.2.3, respectively.

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(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.

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The NRC approved an amendment to 10CFR50, Appendix J, "Leak Rate Testing of Containment of Light-Water-Cooled Nuclear Power Plants", to implement a performance-based option for leakage testing of containment.

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.

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.

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Attachment II
Description of and Justification for
Technical Specification Change

Description of Changes

The changes included in Attachment I will implement the NRC's revision to 10 CFR 50, Appendix J, which became effective on October 26, 1995. The revision to the regulation represents a shift away from prescriptive testing requirements in Appendix J, Option A, to a performance-based approach (Option B). Specifically, upon completion of two consecutive successful Type A tests, the licensee may extend the test interval up to 10 years between Type A tests. (Option B also provides for test interval extensions for Type B and C testing, but these changes are not being requested at this time.)

The changes requested herein include:

Specification 4.4.1.1 is revised to refer to the controlling regulations (10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B) and the Regulatory Guide that provides guidance by which Integrated Leak Tests will be performed.

Specifications 4.4.1.1.1, 4.4.1.1.2, 4.4.1.1.4 and 4.4.1.1.5 are being deleted. These specifications contain provisions which are redundant to, or in conflict with (e.g., reduced pressure testing is no longer acceptable), the guidance contained in Regulatory Guide 1.163.

References to reduced pressure tests in Specification 4.4.1.1.3 are deleted.

An obsolete footnote, to allow the use of the mass-plot test method, is being deleted.

The reference to Appendix J in Specification 4.4.1.2.1 is clarified to reflect that local (Type B and C) testing will continue to be performed pursuant to Option A (current practices).

The Bases are revised to delete references to reduced-pressure testing and to acknowledge the performance-based nature of the rule change.

A change will be made to UFSAR Section 3.8.1.7.4, to delete reference to half-pressure testing, in the periodic update following approval of this amendment request.

Technical Justification

The proposed changes are based on approved guidance documents from the NRC and Nuclear Energy Institute (NEI), including NEI 94-01, dated July 26, 1995; Regulatory Guide 1.163, dated September, 1995; and sample Improved Standard Technical Specifications (ISTS) developed by NEI, with NRC cooperation. The sample ISTS provided guidance on the scope of changes that the NRC expects to see from each of the utilities who elect to pursue Option B. The changes presented in this application meet the intent of the changes, relative to Type A testing, that have been approved in concept by the NRC. The NRC has determined that the industry guideline (NEI 94-01) referenced in the Regulatory Guide, with some exceptions, is an acceptable means of demonstrating compliance with the requirements of Option B. Duke Power intends to comply with the provisions of the NEI document, except as modified by the Regulatory Guide. NUREG-1493 contains a detailed technical justification for conversion to the performance-based containment leak rate testing as specified in 10CFR50 Appendix J, Option B.

The acceptance criteria for Type A tests, as specified in TS 4.4.1.1.3 has not changed.

Deleting the details of the test program from TSs, and providing a reference to the guidance document (RG 1.163) is consistent with the recommendations of the Regulatory Guide, and with the criteria of 10 CFR 50.36.

The change in the test interval, based on the performance of the containment structure in previous tests, has been determined by the NRC's own analysis, presented in NUREG-1493, to have a minimal impact on safety.

Attachment III

No Significant Hazards Analysis

The following analysis is presented, pursuant to 10 CFR 50.91, to demonstrate that the proposed change will not create a Significant Hazard Consideration.

1. The proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Containment leak rate testing is not an initiator of any accident; the proposed change does not affect reactor operations or accident analysis, and has no significant radiological consequences. Therefore, this proposed change will not involve an increase in the probability or consequences of any previously-evaluated accident.

2. The proposed change will not create the possibility of any new accident not previously evaluated.

The proposed change does not affect normal plant operations or configuration, or change any design basis. The proposed changes will not affect the response of containment during a design basis accident.

3. There is no significant reduction in a margin of safety.

The proposed changes are based on NRC-accepted provisions, and maintain necessary levels of reliability of containment integrity. The performanced-based approach to leakage rate testing recognizes that historically good results of containment testing provide appropriate assurance of future containment integrity; this supports the conclusion that the impact on the health and safety of the public as a result of extended test intervals is negligible.

Based on the above, no significant hazards consideration is created by the proposed change.

Environmental Assessment

The proposed change has been evaluated to determine if any environmental impact would be created. The change is considered to meet the criteria (presented in 10 CFR 51.22(c)(9)) for categorical exclusion from the requirements for an environmental assessment, because:

A. As documented above, the change will create No Significant Hazards Consideration.

B. There is no change in the type, or significant increase in the amounts, of any effluent that may be released offsite.

The change will create no new mechanism by which effluents are released, and will provide continued assurances that leakage remains within the existing allowed leakage.

C. There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not change methods by which radioactive materials, including effluents, are handled, processed, or disposed of. Normal radiation levels within the nuclear station will not increase, and this change will not result in personnel spending additional time in radiation areas. Therefore, there will be no increase in individual or cumulative radiation exposure.