DAVE BAXTER
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP / 7800 Rochester Highway
Seneca, SC 29672

864-873-4460
864-873-4208 fax
dave.baxter@duke-energy.com

July 14, 2010

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
License Amendment Request for Adoption of NRC Approved Technical Specification Task Force Change to the Standard Technical Specifications, TSTF-52, Implement 10 CFR 50, Appendix J, Option B
License Amendment Request (LAR) No. 2009-08

In accordance with 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend Appendix A, Technical Specifications (TS), for Renewed Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station (ONS), Units 1, 2, and 3. This LAR requests the Nuclear Regulatory Commission (NRC) to review and approve a License Amendment Request (LAR) that will adopt NRC Approved Technical Specification Task Force (TSTF) Change to the Standard TS, TSTF-52 to implement 10 CFR 50, Appendix J, Option B.

10 CFR 50, Appendix J, Option B governs performance-based containment leakage testing requirements for Type B and C testing. Oconee previously implemented 10 CFR 50, Appendix J, Option B, Type A testing. This was approved by the NRC in a safety evaluation dated October 30, 1996.

In accordance with Duke Energy administrative procedures and the Quality Assurance Program Topical Report, these proposed changes have been reviewed and approved by the Plant Operations Review Committee. Additionally, a copy of this LAR is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Duke Energy requests that this proposed license amendment be reviewed and approved by August 31, 2011, in support of the Fall, 2011 refueling outage and that a 90 day implementation period be granted. Duke Energy will also update applicable sections of the Oconee UFSAR, as necessary, and submit these changes per 10 CFR 50.71(e). There are no new commitments being made as a result of this proposed change.

Inquiries on this proposed amendment request should be directed to Reene' Gambrell of the Oconee Regulatory Compliance Group at (864) 873-3364.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 14, 2010.

Sincerely,

Dave Baxter, Vice President
Oconee Nuclear Site

Enclosure:
1. Evaluation of Proposed Change

Attachments:
1. Technical Specifications – Mark Ups
2. Technical Specifications - Reprinted Pages

bc w/enclosures and attachments:

Mr. Luis Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, Georgia 30303-1257

Mr. John Stang, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8 G9A
Washington, D. C. 20555

Mr. Andy Sabisch
Senior Resident Inspector
Oconee Nuclear Site

Mrs. Susan E. Jenkins, Manager
Infectious and Radioactive Waste Management Section
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

ENCLOSURE 1

EVALUATION OF PROPOSED CHANGE

Subject: License Amendment Request for Adoption of NRC Approved Technical Specification Task Force Change to the Standard Technical Specifications, TSTF-52, Implement 10 CFR 50, Appendix J, Option B

1.0 SUMMARY DESCRIPTION

This LAR requests the Nuclear Regulatory Commission (NRC) to review and approve changes to the Technical Specifications (TS) to implement the performance based Option B of 10 CFR 50, Appendix J for Types B and C testing.

In September 1995, the NRC issued NUREG-1493, "Performance-Based Containment Leak-Test Program." This document contained findings that supported extending the containment leak testing intervals. With Regulatory Guide 1.163, dated September 1995, "Performance-Based Containment Leak-Test Program," the NRC amended its regulations to provide a performance based option, Option B, for leakage rate testing of containments of light water cooled nuclear power plants. Regulatory Guide 1.163 endorses with exceptions NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and ANSI/ANS 56.8-1994," Containment System Leakage Testing Requirements." Licensees may voluntarily comply with Option B as an alternative to the prescriptive requirements in Appendix J, Option A. Oconee proposes to revise its TS and Containment Leakage Rate Testing Program to implement the performance based option of 10 CFR 50, Appendix J for Types B and C testing. The proposed changes have been prepared in accordance with the guidance provided in Regulatory Guide 1.163, NEI 94-01, and ANSI/ANS-56.8-1994.

Option B of 10 CFR 50, Appendix J will allow an extended test interval for Types B and C testing. Types B and C extended test intervals are based upon satisfactory performance of two "As Found" tests (test performance prior to any maintenance on the component).

2.0 DETAILED DESCRIPTION

2.1 Containment Leakage Rate Testing Program

The Containment Leakage Rate Testing Program provides controls for implementation as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing. This was approved in a NRC safety evaluation dated October 30, 1996.

The program is in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995.

Type B and C testing is currently implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

2.2 TS Change Description

The proposed TS and associated Bases changes required for 10 CFR 50, Appendix J, Option B implementation are consistent with the model changes made to Improved Standard TS and Bases, as delineated in TSTF-52, Revision 3.

TS 3.6.1

TS Surveillance Requirements (SRs) 3.6.1.1 and 3.6.1.2 delineate the requirements for periodic leakage rate testing for Types A, B, and C tests. Type A testing is presently conducted according to the requirements of 10 CFR 50, Appendix J, Option B. Types B and C testing are presently conducted according to the requirements of 10 CFR 50, Appendix J, Option A. The TS split these testing requirements into two distinct SRs, one for Appendix J, Option B, Type A testing and one for Appendix J, Option A, Types B and C testing. This proposed amendment combines all requirements for testing into SR 3.6.1.1 and will reference the Containment Leakage Rate Testing Program. The frequency note in SR 3.6.1.2 concerning SR 3.0.2 is no longer applicable when Option B is employed and is not carried forward.

Compliance with the requirements of 10 CFR 50, Appendix J, is still assured. The Containment Leakage Rate Testing Program will contain specifics concerning Oconee compliance with the requirements of 10 CFR 50, Appendix J, Option B and the exemptions that have been approved by the NRC. The referenced Containment Leakage Rate Testing Program establishment, implementation, and maintenance are required by the program description in TS 5.5.2. Specific exemptions will be controlled in the Containment Leakage Rate Testing Program. The relocation of Type B and C acceptance criteria to the TS Administrative Controls and the Containment Leakage Rate Testing Program is consistent with NUREG-1430, "Standard Technical Specifications, Babcock & Wilcox Plants." 10 CFR 50, Appendix J, Option B allows longer intervals between leakage tests based on performance trends but does not allow an increase in the leakage acceptance criteria.

SR 3.6.1.3, which verifies containment structural integrity, will be renumbered to SR 3.6.1.2 as a result of combining SR 3.6.1.1 and SR 3.6.1.2 into one SR.

TS 3.6.2

TS SR 3.6.2.1 delineates the requirements for periodic leakage rate testing for primary containment air locks. Air lock testing is presently conducted according to the requirements of 10 CFR 50, Appendix J, Option A. This proposed amendment deletes the reference to 10 CFR 50, Appendix J, Option A and the air lock testing acceptance criteria and provides a reference to the Containment Leakage Rate

Testing Program. SR Note 2 is being revised to state that SR 3.6.1.1 acceptance criteria are applicable. The FREQUENCY note concerning SR 3.0.2 not being applicable is also being deleted, as this note is not used when Option B is employed.

The requirements of 10 CFR 50, Appendix J, Option B revise the SRs regarding containment air locks. Under Option B, containment air locks shall be tested at an internal pressure of not less than a specified pressure prior to a pre-operational Type A test. Subsequent periodic tests shall be performed at a frequency of at least once per 30 months. When containment integrity is required, air lock door seals should be tested within 7 days after each containment access. For periods of multiple containment entries where the air lock doors are routinely used for access more frequently than once every 7 days (e.g., each shift, daily inspection tours of the containment, or more than once within a 7-day period), door seals may be tested once per 30 days.

This change does not adversely affect the safe operation of the facility, since the acceptance criteria are not being changed.

The specific requirements contained in SR 3.6.2.1 are encompassed within the Containment Leakage Rate Testing Program, which provides reference to Option B of 10 CFR 50, Appendix J. The appropriate reference to the Containment Leakage Rate Testing Program within SR 3.6.2.1 ensures sufficient information is retained within the TS. Because the proposed changes are consistent with the current plant configuration, NUREG-1430, Revision 3, and Option B of 10 CFR 50, Appendix J, the proposed changes do not adversely affect existing plant safety margins.

TS 5.5.2

TS 5.5.2, Containment Leakage Rate Testing Program, is modified to delete the specific reference to Appendix J, Option B, Type A testing. Appendix J, Option B has been extended to include Type B and C testing. All references to Appendix J, Option A testing have been deleted. This change is made for consistency with TSTF-52. The following minor changes have been added for TSTF-52 consistency purposes: 1) a sentence is added to the calculated peak containment pressure that states the containment design pressure is 59 psig; and 2) a statement is added at the end of section 5.2.2 that states "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J."

The statement regarding the first Unit 3 SR 3.6.1.1,Type A testing is being deleted because it is an expired one time extension allowance that is no longer applicable. The Containment Leakage Rate Testing Program is defined in accordance with the requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified

by currently approved exemptions. The program is in accordance with the guidelines contained within Regulatory Guide 1.163.

No changes were made to acceptance criteria in their incorporation within the TS Administrative Controls; therefore, existing safety margins remain unaffected by these changes.

TS Bases 3.0

The Bases for SR 3.0.2 have been modified to be consistent with TSTF-52 concerning the fact that SR 3.0.2 (which provides for a 25% grace period on surveillance intervals) cannot be applied to frequencies contained in the Containment Leakage Rate Testing Program. The test intervals delineated in the Containment Leakage Rate Testing Program are specified in regulations; therefore, the 25% grace period allowed by SR 3.0.2 does not apply. The sentence was removed to conform to TSTF-52.

TS Bases 3.6.1 and 3.6.2

The Bases have been revised to appropriately reflect the changes to TS 3.6.1 and TS 3.6.2 described above.

An editorial correction has also been made to revise the allowable leakage rate provided in the Applicable Safety Analyses of TS Bases 3.6.1 and 3.6.2 from 0.25% to 0.20%. These corrections were inadvertently missed in the TS Bases change associated with adoption of Alternate Source Term in license amendment 338, 339, & 339.

2.3 Deviations From TSTF-52, Revision 3

TS 1.1, Definitions

TSTF-52 deletes L_a from the definitions in TS 1.1.

Deviation - There is no definition for L_a in TS 1.1 of Oconee's TS, so this change is unnecessary.

TS 5.5.2, Containment Leakage Rate Testing Program

TSTF-52 revises TS 5.5.2 to include the plant specific air lock testing acceptance criteria.

Deviation – Oconee's airlock testing acceptance criteria is already evaluated in the containment leakage rate acceptance criterion currently provided in TS 5.5.2.

Oconee used BWOG STS Rev. 1 (NUREG 1430) as the basis for conversion to ITS. Oconee chose not to adopt the airlock testing acceptance criteria provided in TS 3.6.2.1 of the BWOG STS Rev. 1 because no requirement existed in the Oconee TSs at that time. Oconee justified this by stating that the air lock's contribution to Type B and C containment leakage is limited such that the Type B and C containment leakage cannot exceed the applicable Type B and C containment leakage limits. Oconee added containment leakage rate testing requirements to the administrative section (TS 5.5.2) based on the NRC model for implementation of 10 CFR 50 Appendix J Option B enclosed in a letter from the NRC to NEI dated November 2, 1995. The recommended format did not contain specific airlock acceptance criteria for the administrative section. Containment leakage testing was based on meeting overall Containment leakage criterion. The NRC approved this approach in Amendment 300, 300, & 300 dated December 16, 1998. Duke will continue to maintain this portion of its licensing basis as is; thus, Oconee will deviate from TSTF-52. Compliance with the requirements of 10 CFR 50, Appendix J, is still assured in that Containment airlock leakage is a subset of overall Containment leakage. Therefore, meeting the acceptance criterion for Containment leakage bounds Containment airlock leakage.

The relocation of Type B and C acceptance criteria to the TS Administrative Controls and the Containment Leakage Rate Testing Program is still consistent with NUREG-1430, "Standard Technical Specifications, Babcock & Wilcox Plants."

<u>TS Bases 3.6.3, Containment Isolation Valves</u>

TSTF-52 revises SR 3.6.3.6 to reflect Appendix J, Option [A] [B].

Deviation – There is no equivalent SR 3.6.3.6 in Oconee's TS 3.6.3; therefore, there is no bases equivalent.

3.0 TECHNICAL EVALUATION

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS. Allowable leakage rates are determined such that the containment leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in FR (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the NRC undertook a study of possible changes to this regulation. Previous study histories of containment were studied and the results of that study

are reported in NUREG-1493, "Performance-Based Leak-Test Program."

As a result of the study, the NRC developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of a revision to 10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate and individual component performance. The previous rule was retained as Option A.

The NRC developed Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September, 1995, to specify a method acceptable for complying with Option B. RG 1.163 endorses with exceptions NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and ANSI/ANS 56.8-1994," Containment System Leakage Testing Requirements."

Licensees may voluntarily comply with Option B as an alternative to the prescriptive requirements in Appendix J, Option A. Oconee proposes to revise its TS and Containment Leakage Rate Testing Program to implement the performance based option of 10 CFR 50, Appendix J for Types B and C testing. The proposed changes have been prepared in accordance with the guidance provided in Regulatory Guide 1.163, NEI 94-01, and ANSI/ANS-56.8-1994.

Option B of 10 CFR 50, Appendix J will allow an extended test interval for Types B and C testing. Types B and C extended test intervals are based upon satisfactory performance of two "As Found" tests (test performance prior to any maintenance on the component).

4.0 REGULATORY EVALUATION

4.1 Significant Hazards Consideration

Duke Energy Carolinas, LLC, has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

No. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within acceptance limits as delineated in 10 CFR 50, Appendix J, Option B. The changes are consistent with current safety analyses. Although some of the proposed changes represent minor relaxation to existing TS requirements, they are consistent with the requirements specified by Option B of 10 CFR 50, Appendix J. The systems affecting containment integrity related to this proposed amendment request are not assumed in any safety analyses to initiate any accident sequence. Therefore, the probability of any accident previously evaluated is not increased by this proposed amendment. The proposed changes maintain an equivalent level of reliability and availability for all affected systems. In addition, maintaining leakage within analyzed limits assumed in accident analyses does not adversely affect either onsite or offsite dose consequences.

Therefore, adopting Appendix J, Option B does not significantly increase the probability or consequences of any accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. No changes are being proposed which will introduce any physical changes to the existing plant design. The proposed changes are consistent with the current safety analyses. Some of the changes may involve revision in the testing of components; however, these are in accordance with the current safety analyses and provide for appropriate testing or surveillance that is consistent with 10 CFR 50, Appendix J, Option B. The proposed changes will not introduce new failure mechanisms beyond those already considered in the current accident analyses. No new modes of operation are introduced by the proposed changes. The proposed changes maintain, at minimum, the present level of operability of any system that affects containment integrity.

Therefore, adoption of Appendix J, Option B will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The provisions specified in Option B of 10 CFR 50, Appendix J allow changes to Type B and Type C test intervals based upon the performance of past leak rate tests. 10 CFR 50, Appendix J, Option B allows longer intervals between leakage tests based on performance trends, but does not relax the leakage acceptance criteria. Changing test intervals from those currently provided in the TS to those provided in 10 CFR 50, Appendix J, Option B does not increase any risks above and beyond those that the NRC has deemed acceptable for the performance based option. In addition, there are risk reduction benefits associated with reduction in component cycling, stress, and

wear associated with increased test intervals. The proposed changes provide continued assurance of leakage integrity of containment without adversely affecting the public health and safety and will not significantly reduce existing safety margins.

Therefore, adoption of Appendix J, option B does not involve a significant reduction in a margin of safety.

4.2 Applicable Regulatory Requirements/Criteria

NUREG-1493, Performance-Based Containment Leak-Test Program, September, 1995.
RG 1.163, Performance-Based Containment Leak Test Program, September, 1995.
10 CFR 50 Appendix J, Option B
TS 5.5.2, Containment Leakage Rate Testing Program
UFSAR, Chapter 3, Section 3.8 and Chapter 6, Section 6.3.

4.3 Precedent

October 28, 1997	Grand Gulf - License Amendment Request to adopt Appendix J Option B Testing.
April 6, 1998	Issuance of Amendment No. 135 to Facility Operating License No. NPF-29 – Grand Gulf Nuclear Station, Unit 1 (TAC NO. M99879).
March 1, 2001	Catawba Nuclear Station – Implementation of 10 CFR 50, Appendix J, Option B for Type B and C Testing.
July 31, 2001	Catawba Nuclear Station - Issuance of Amendments 192 and 184 to Facility Operating License No. NPF-35 and NPF-52.
December 7, 2001	McGuire Nuclear Station – Implementation of 10 CFR 50, Appendix J, Option B for Type B and C Testing.
September 4, 2002	McGuire Nuclear Station - Issuance of Amendments 207 and 188 to Facility Operating License No. NPF-9 and NPF-17.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the

amendment will not be adverse to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

Duke Energy Carolinas, LLC, has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke Energy Carolinas, LLC has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 4.1, adopting Appendix J, Option B does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

Adopting Appendix J, Option B will not impact effluents released offsite. Therefore, there will be no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

Adopting Appendix J, Option B will not impact occupational radiation exposure. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT 1

TECHNICAL SPECIFICATIONS AND TECHNICAL SPECIFICATIONS BASES

MARK UPS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and ~~Type A~~ leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
~~SR 3.6.1.2~~	~~Perform required Type B and C leakage rate testing~~ except for containment air lock testing, ~~in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~ ~~The leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests.~~	~~NOTE~~ ~~SR 3.0.2 is not applicable~~ ~~In accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~
SR 3.6.1.~~3~~ 2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria ~~of~~ applicable to SR 3.6.1.~~2~~ 1. ~~in accordance with 10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~ ----------	~~----------NOTE----------~~ ~~SR 3.0.2 is not applicable~~ ~~----------~~
	Perform required air lock leakage rate testing in accordance with ~~10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~ the Containment Leakage Rate Testing Program.	In accordance with ~~10 CFR 50, Appendix J, Option A, as modified by approved exemptions.~~
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	18 months

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Station Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

~~This~~ A program ~~provides controls for implementation of~~ shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions ~~for Type A testing~~.

~~The first Unit 3 SR 3.6.1.1 Type A test that is performed after the September 11, 1992, test, shall be performed no later than April 11, 2005.~~

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

Add to 1st paragraph of 5.5.2

This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Containment system visual examinations required by Regulatory Guide 1.163, Regulatory Position C.3 shall be performed as follows:

1. Accessible concrete surfaces and post-tensioning system component surfaces of the concrete containment shall be visually examined prior to initiating SR 3.6.1.1 Type A test. These visual examinations, or any portion thereof, shall be performed no earlier than 90 days prior to the start of refueling outages in which Type A tests will be performed. The validity of these visual examinations will be evaluated should any event or condition capable of affecting the integrity of the containment system occur between the completion of the visual examinations and the Type A test.

2. Accessible interior and exterior surfaces of metallic pressure retaining components of the containment system shall be visually examined at least three times every ten years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

~~Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig. The containment design pressure is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day.

Leakage rate acceptance criterion is:

a. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests, and ≤ 0.75 L_a for Type A tests;

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing frequencies required by 10 CFR 50, Appendix J.

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is ~~a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions."~~ The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations.

~~Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."~~

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..."basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the reactor building (RB) structure, its steel liner, and the penetrations of this liner and structure. The containment is designed to contain radioactive material that may be released from the reactor core following an accident (LOCA). Additionally, the containment provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

design basis Loss of coolant

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The containment design includes ungrouted tendons where the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

Design Bases

The reinforced concrete structure is required for structural integrity of the containment under accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity ~~from the containment to the environment. SR 3.6.1.1 and SR 3.6.1.2~~ leakage rate requirements comply with 10 CFR 50, Appendix J, Option ~~A and~~ B ~~as applicable~~ (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. The equipment hatch is closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting accident without exceeding the design leakage rate.

The accidents that result in a challenge to containment from high pressures and temperatures are a ~~loss of coolant accident~~ (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the accident analyses, it is assumed that the containment is OPERABLE such that, for the accidents involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~0.25~~ 20% of containment air weight per day (Ref. 3). This leakage rate, ~~used in~~ the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option ~~A and~~ B ~~(Ref. 1), as~~ L_a: the maximum allowable leakage rate at the ~~calculated~~ design bases LOCA maximum peak containment pressure (P_a) resulting from the limiting ~~accident~~. The allowable leakage rate represented by L_a forms the basis ~~for the acceptance criteria~~ imposed on all containment leakage rate testing. L_a is assumed to be ~~0.25~~ 20% per day in the safety analysis at P_a = 59.0 psig (Ref. 3).

The containment satisfies Criterion 3 of the 10 CFR 50.36 (Ref. 4).

LCO

Containment Leakage Rate Testing Program

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required ~~10 CFR 50, Appendix J,~~ leakage test. At this time, the applicable leakage limits must be met. Compliance with this LCO will ensure a ~~containment~~ configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

BASES (continued)

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If the Required Action and associated Completion Time is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Containment Leakage Rate Testing Program for Option B

Maintaining the containment OPERABLE requires compliance with the visual examinations and ~~Type A~~ leakage rate test requirements of the Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be ≤ 0.75 L_a for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

< 0.6 La for combined Type B and C leakage, and

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.6.1.2~~

~~Maintaining the containment OPERABLE requires compliance with the Type B and C leakage rate test requirements of 10 CFR 50, Appendix J, Option A (Ref. 1), as modified by approved exemptions. As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, Option A, leakage test is required to be < 0.6 L_a for combined Type B and C leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Appendix J, Option A, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.~~

SR 3.6.1.~~3~~ 2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are as described in Specification 5.5.7, "Pre-stressed Concrete Containment Tendon Surveillance Program."

REFERENCES

1. 10 CFR 50, Appendix J, Option ~~A and~~ B.
2. UFSAR, Sections 15.13 and 15.14.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36.
5. UFSAR Chapter 18, Table 18-1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks, also known as the personnel hatch and the emergency hatch, form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following an accident in containment. As such, closure of a single door supports containment OPERABILITY. Each of the outer doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). Each personnel air lock door is provided with limit switches that provide control room indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the containment leakage rate within limit in the event of an accident. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The accident that results in a release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed ~~with an allowable~~ leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option ~~A and~~ B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated

.28%

BASES

APPLICABLE SAFETY ANALYSES (continued)

maximum peak containment pressure (P_a) following a ~~accident~~ design bases LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

Each containment air lock forms part of the containment pressure boundary. As a part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from an accident. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are normally closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. An inoperable inner door can be accessed from inside containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is

BASES

ACTIONS

D.1 and D.2 (continued)

12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of ~~10 CFR 50, Appendix J, Option A (Ref. 1), as modified by approved exemptions~~. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by ~~10 CFR 50, Appendix J, Option A, as modified by approved exemptions~~. ~~Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. Either a full air lock leak test or a leak test of the outer air lock door seal performed within 7 days of initial opening, and during periods of frequent use, at least once every 17 30 days, is an acceptable method of complying with 10 CFR 50, Appendix J requirements (References 5 and 6).~~

Containment Leakage Rate Testing Program

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of an accident. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.~~2~~1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry or exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, and the potential loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. 10 CFR 50, Appendix J, Option ~~A and~~ B.
2. UFSAR, Section 15.14.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36.
5. Duke Power Company letter from William O. Parker, Jr. to Harold R. Denton (NRC) dated July 24, 1981.
6. NRC Letter from Philip C. Wagner to William O. Parker, Jr., dated November 6, 1981, Issuance of Amendment 104, 104 and 101 to Licenses DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station Units Nos 1, 2 and 3.

ATTACHMENT 2

TECHNICAL SPECIFICATIONS AND TECHNICAL SPECIFICATIONS BASES

REPRINTED PAGES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment airlock testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	18 months

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Station Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

Test Program," dated September 1995. Containment system visual examinations required by Regulatory Guide 1.163, Regulatory Position C.3 shall be performed as follows:

1. Accessible concrete surfaces and post-tensioning system component surfaces of the concrete containment shall be visually examined prior to initiating SR 3.6.1.1 Type A test. These visual examinations, or any portion thereof, shall be performed no earlier than 90 days prior to the start of refueling outages in which Type A tests will be performed. The validity of these visual examinations will be evaluated should any event or condition capable of affecting the integrity of the containment system occur between the completion of the visual examinations and the Type A test.

2. Accessible interior and exterior surfaces of metallic pressure retaining components of the containment system shall be visually examined at least three times every ten years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig. The containment design pressure is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of the containment air weight per day.

Leakage rate acceptance criterion is:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..."basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the reactor building (RB) structure, its steel liner, and the penetrations of this liner and structure. The containment is designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the containment provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The containment design includes ungrouted tendons where the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The reinforced concrete structure is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. The equipment hatch is closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting accident without exceeding the design leakage rate.

The accidents that result in a challenge to containment from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the accident analyses, it is assumed that the containment is OPERABLE such that, for the accidents involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.20% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable leakage rate at the calculated maximum peak containment pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis at P_a = 59.0 psig (Ref. 3).

The containment satisfies Criterion 3 of the 10 CFR 50.36 (Ref. 4).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met. Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

BASES (continued)

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If the Required Action and associated Completion Time is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for Option B for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are as described in Specification 5.5.7, "Pre-stressed Concrete Containment Tendon Surveillance Program."

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Sections 15.13 and 15.14.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36.
5. UFSAR Chapter 18, Table 18-1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks, also known as the personnel hatch and the emergency hatch, form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following an accident in containment. As such, closure of a single door supports containment OPERABILITY. Each of the outer doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). Each personnel air lock door is provided with limit switches that provide control room indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the containment leakage rate within limit in the event of an accident. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The accident that results in a release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.20% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated

BASES

APPLICABLE SAFETY ANALYSES (continued)

maximum peak containment pressure (P_a) following a design bases LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4).

LCO

Each containment air lock forms part of the containment pressure boundary. As a part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from an accident. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are normally closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, an accident could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. An inoperable inner door can be accessed from inside containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is

BASES

ACTIONS

D.1 and D.2 (continued)

12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of an accident. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry or exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage, and the potential loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during use of the air lock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 15.14.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36.
5. Duke Power Company letter from William O. Parker, Jr. to Harold R. Denton (NRC) dated July 24, 1981.
6. NRC Letter from Philip C. Wagner to William O. Parker, Jr., dated November 6, 1981, Issuance of Amendment 104, 104 and 101 to Licenses DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station Units Nos 1, 2 and 3.