Duke Energy

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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Proposed Technical Specifications Amendment
Technical Specification Bases 3.0 (Surveillance Requirement (SR) Applicability), Technical Specifications and Bases 3.6.1 (Containment), 3.6.2 (Containment Air Locks), 3.6.3 (Containment Isolation Valves), Technical Specification 5.5.2 (Containment Leakage Rate Testing Program)
Implementation of 10 CFR 50, Appendix J, Option B for Type B and C Testing

Pursuant to 10 CFR 50.90, Duke Energy Corporation is requesting an amendment to the McGuire Nuclear Station Facility Operating License and Technical Specifications (TS). This amendment will allow implementation of 10 CFR 50, Appendix J, Option B, which governs performance-based containment leakage testing requirements, for Type B and C testing. McGuire had previously implemented 10 CFR 50, Appendix J, Option B requirements for Type A testing. In addition to the changes associated with the adoption of 10 CFR 50, Appendix J, Option B, McGuire is also proposing the following changes:

1. TS 3.6.3 is being modified to delete the requirement for conducting soap bubble tests of welded penetrations during Type A tests which are not individually Type B or Type C testable.

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2. TS 3.6.3 is also being modified to delete a separate requirement for leak testing containment purge lower and upper compartment and instrument room valves with resilient seals. These valves will be covered by the overall Containment Leakage Rate Testing Program.

The contents of this amendment request package are as follows:

Attachment 1 provides a marked copy of the affected TS and Bases pages for McGuire, showing the proposed changes. Attachment 2 contains reprinted pages of the affected TS and Bases pages. Attachment 3 provides a description of the proposed changes and technical justification. Pursuant to 10 CFR 50.92, Attachment 4 documents the determination that the amendment contains No Significant Hazards Considerations. Pursuant to 10 CFR 51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

This license amendment request is consistent with the guidance contained in Technical Specification Task Force (TSTF)-52, Revision 3. Revision 3 to TSTF-52 has been reviewed and approved by the NRC.

For scheduling and/or personnel dose concerns, McGuire may elect not to perform as-found testing on selected components. These components will not be placed on extended test intervals until they meet the as-found criteria referenced in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, dated July 26, 1995. As-found testing will not be required for components that remain on the nominal test intervals, with one exception. An as-found test will be required prior to work being done on a component to correct a condition expected to adversely affect its leak tightness. In cases when as-found testing does not need to be performed, the as-left test results will be recorded as the as-found test results. Similar exceptions to as-found testing have been previously approved by the NRC for other plants.

Implementation of this amendment to the McGuire Facility Operating License and TS will impact the McGuire Updated Final Safety Analysis Report (UFSAR). The affected UFSAR section is 6.2.4, "Containment Isolation System". Necessary UFSAR changes will be submitted in accordance with 10 CFR 50.71(e).

This license amendment request is almost identical to the Catawba submittal dated March 1, 2001 (TAC NOS. MB1383 and MB1384). The only significant difference is the deletion of the separate requirement for leak testing of the containment purge valves as stipulated in Surveillance Requirement 3.6.3.6. Sections that are different from the Catawba submittal are noted by the phrase "deviation from Catawba submittal" in the marked-up pages of the amendment request.

Duke is requesting NRC review and approval of this amendment request by February 20, 2002, so that this amendment may be implemented in conjunction with the Unit 2 End-of-Cycle 14 Refueling Outage. Duke has determined that the standard 30-day implementation period will be sufficient for this amendment.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been previously reviewed and approved by the McGuire Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the appropriate State of North Carolina official.

Inquiries on this matter should be directed to Norman T. Simms at (704) 875-4685.

Sincerely,

HB Barron

H B Barron

Attachments

cc: w/attachments

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H.B. Barron, being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this amendment to the McGuire Nuclear Station Facility Operating Licenses Numbers NPF-9 and NPF-17 and Technical Specifications; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

H B Barron

H.B. Barron, Site Vice President

Subscribed and sworn to me: 12/7/01
Date

Deborah G. Thrap, Deborah G. Thrap
Notary Public

My commission expires: 4/6/2002
Date

bxc: w/attachments

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ATTACHMENT 1

MARKED-UP TECHNICAL SPECIFICATIONS AND BASES PAGES FOR McGUIRE

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.1 ---------------NOTE (5)--------------- (1) The space between each dual ply bellows assembly on penetrations between the containment building and annulus shall be vented to the annulus during Type A tests. * See next page --- Perform required visual examinations and ~~Type A~~ 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

(continued)

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SURVEILLANCE	FREQUENCY
~~SR 3.6.1.2 ---NOTE---~~ ~~1.~~ 2. Following each Type A test, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3 to 5 psig to verify no detectable leakage, or the assembly shall be subjected to a leak test with the pressure on the containment side of the assembly at P_a. ~~2.~~ 3. Type C tests on penetrations M372 and M373 may be performed without draining the glycol-water mixture from the seats of their diaphragm valves if meeting a zero indicated leakage rate (not including instrument error). --- ~~Perform required Type B and C leakage rate testing, except for containment air lock testing and valves with resilient seals, 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~~

ACTIONS (continued)
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	------NOTE------ 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.~~2~~ 1. ------ 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. ~~The acceptance criteria for air lock testing are:~~ ~~a. Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.~~ ~~b. For each door, leakage rate is $< 0.01 L_a$ when tested at $\geq$ 14.8 psig.~~	~~------NOTE------ SR 3.0.2 is not applicable ------~~ In accordance with ~~10 CFR 50, Appendix J, Option A, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program.
SR 3.6.2.2	Perform a pressure test on each inflatable air lock door seal and verify door seal leakage is < 15 sccm.	6 months
SR 3.6.2.3	Verify only one door in the air lock can be opened at a time.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	---NOTE--- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --- Verify each containment isolation manual valve and blind flange that is located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for containment purge lower and upper compartment and Instrument room valves with resilient seals. Deviation From Catawba's Submittal	~~184 days~~ ~~AND~~ ~~within 92 days after opening the valve~~ In accordance with the Containment Leakage Rate Testing Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.8	~~------------------------------NOTE------------------------------ Penetrations not individually testable shall be determined to have no visible leakage when tested with soap bubbles. --~~	~~---------NOTE------- SR 3.0.2 is not applicable. -------------------------~~
	Verify the combined leakage rate for all reactor building bypass leakage paths is $\leq$ 0.07 L_a when pressurized to $\geq$ P_a, 14.8 psig.	In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions, for Type B and C testable penetrations~~ the Containment Leakage Rate Testing Program ~~AND During SR 3.6.1.1 Type A tests for penetrations not individually testable~~

Deviation From Catawba's Submittal

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 be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, ~~for Type A testing,~~ as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

(continued)

The containment design pressure is 15 psig.

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 14.8 psig. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.3% of containment air weight per day.

Leakage Rate acceptance criteria are:

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

and $< 0.6 L_a$ for Type B and Type C tests

b. Airlock testing acceptance criteria for the overall airlock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$. For each door, the leakage rate is $\leq 0.01\ L_a$ when tested at ≥ 14.8 psig.

~~The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.~~

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.

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection, Chemical and Volume Control, Nuclear Sampling, RHR, Boron Recycle, Refueling Water, Liquid Waste, and Waste Gas. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.4 Deleted

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

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.

Insert "A"

~~Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SRs include a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.~~

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

INSERT "A"

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 requirements of regulations. The TS cannot in and of themselves extend a test interval specified in regulations.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete reactor building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain the radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident (LOCA). Additionally, the containment vessel and reactor building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a flat circular base. It is completely enclosed by a reinforced concrete reactor building. An annular space exists between the walls and domes of the steel containment vessel and the concrete reactor building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The reactor building provides shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and Nuclear Steam Supply System.

The inner steel containment 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. ~~SR 3.6.1.2 leakage rate requirements comply with 10 CFR 50, Appendix J, Option A (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:

BASES

BACKGROUND (continued)

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 secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

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

c. All equipment hatches are closed and sealed; and

d. The sealing mechanism associated with a penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY 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 DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs 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 0.3% 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 containment leakage rate at the calculated peak containment internal 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 0.3% per day in the safety analysis at P_a = 14.8 psig (Ref. 3). Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

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

BASES

LCO

Containment Leakage Rate Testing Program

the applicable leakage limits must be met.

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 combined Type B and C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a.~~

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.

Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and reactor building bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA 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 3.9.4, "Containment Penetrations."

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 OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within

BASES

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and ~~Type A~~ leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet specific leakage limits for the air lock, secondary containment bypass leakage path, and purge valve with resilient seals (as specified in LCO 3.6.2 and LCO 3.6.3) does not invalidate the acceptability of the overall containment leakage determinations unless the specific leakage contribution to overall Type A, B, and C leakage causes one of these overall leakage limits to be exceeded. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be ~~≤ 0.75 L_a~~ for overall Type A 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 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 ≤ 0.75 La for Option B

Note 1

three Notes.

The Surveillance is modified by ~~a Note which~~ requires that the space between each dual-ply bellows assembly on containment penetrations between the containment building and the annulus be vented to the annulus during each Type A test.

~~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. Failure to meet specific leakage limits for the air lock, secondary containment bypass leakage path, and purge valve with resilient seals as specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of the overall containment leakage determinations unless the specific leakage contribution to Type A, B and C leakage causes one of these overall~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~leakage limits to be exceeded. 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.~~

~~The Surveillance is modified by two Notes.~~ Note ~~1~~ 2 requires that following each Type A test, the space between each dual-ply bellows assembly be subjected to a low pressure leak test with no detectable leakage. Otherwise, the assembly must be tested with the containment side of the bellows assembly pressurized to P_a and meet the requirements of SR 3.6.3.8 (bypass leakage requirements). Note ~~2~~ 3 allows penetrations M372 and M373 to be tested without draining the glycol-water mixture from the associated diaphragm valves (NF-228A, NF-233B, and NF-234A) as long as no leakage is indicated. This test may be used in lieu of ~~Section III.C.2(a) of~~ 10 CFR 50, Appendix J, Option ~~A which requires air or nitrogen as the test medium.~~ B as defined in ANSI/ANS 56.8-1994 Section 3.3.5 (Test Medium). The required test pressure and interval are not changed.

Deviation From Catawba's Submittal

All test leakage rates shall be calculated using observed data converted to absolute values. Error analysis shall also be performed to select a balanced integrated leakage measurement system.

REFERENCES

1. 10 CFR 50, Appendix J, Option B

2. UFSAR, Chapter 15.

3. UFSAR, Section 6.2.

4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

, Option B

0.3%

APPLICABLE SAFETY ANALYSES (continued)

design basis LOCA.

defined in 10 CFR 50, Appendix J (Ref. 1), as L_a = ~~0.20%~~ of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 14.8 psig following a ~~DBA~~. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. 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 kept 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, a DBA 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 3.9.4, "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. It is preferred that the air lock be accessed from inside primary 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

BASES

ACTIONS (continued)

status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant 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.~~ 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 acceptance criteria were established during initial air lock and containment OPERABILITY testing. 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 ~~Appendix J, Option A (Ref. 1), as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

~~The frequency required by 10 CFR 50 Appendix J, Option A, includes leak testing each door seal within 72 hours of closing or every 72 hours when entries are being made more frequently. The seal annulus leakage must be < 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the pressure between the seals ≥ P_a. Overall airlock leakage tests are conducted at P_a every 6 months. The overall air lock leakage rate must also be verified prior to establishing containment OPERABILITY. If the periodic 6-month test required Appendix J, Option A, is current, the seal leakage test may be substituted for the full pressure test provided no maintenance has been performed on an air lock. Whenever maintenance has been performed on an air lock, the requirements of paragraph III.D.2(b)(ii) of Appendix J, Option A must still be met. This is an exemption from 10 CFR 50, Appendix J, Option A.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria ~~of~~ which are applicable to SR 3.6.1.~~2~~. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate. 1.

SR 3.6.2.2

Door seals must be tested every 6 months to verify the integrity of the inflatable door seal. The measured leakage rate must be less than 15 standard cubic centimeters per minute (sccm) per door seal when the seal is inflated to approximately 85 psig. This ensures that the seals will remain inflated for at least 7 days should the instrument air supply to the seals be lost. The Frequency of testing ~~is consistent with the overall airlock leakage tests required every 6 months by 10 CFR 50, Appendix J, Option A (Ref. 1).~~ has been demonstrated to be acceptable through operating experience.

SR 3.6.2.3

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 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 and 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 plant outage, and the potential for loss of containment OPERABILITY if the surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the interlock.

BASES

REFERENCES

1. 10 CFR 50, Appendix J~~.~~, Option B
2. UFSAR, Section 6.2.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is specified in the UFSAR and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

B

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option A is required to ensure OPERABILITY. The measured leakage rate for containment purge lower compartment and instrument room valves must be $\leq 0.05\ L_a$ when pressurized to P_a. The measured leakage rate for containment purge upper compartment valves must be $\leq 0.01\ L_a$ when pressurized to P_a. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment) ~~a Frequency of 184 days was established.~~

INSERT "B"

Deviation from Catawba Submittal

~~The containment purge upper compartment valves may be used during normal operation, therefore, in addition to the 184 day Frequency, this SR must be performed every 92 days after opening the valves. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened. The containment purge lower compartment valves and instrument room valves remain closed during normal operation and this SR is only performed every 184 days for these valves.~~

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. The isolation signals involved are Phase A, Phase B, and Safety Injection. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant

Insert "B"

, these valves will not be placed on the maximum extended test interval, but tested on the nominal test interval in accordance with the Containment Leakage Rate Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all reactor building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. ~~The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). Penetrations which are not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized during SR 3.6.1.1 Type A testing.~~ The Frequency is required by ~~10 CFR 50, Appendix J, Option A, as modified by approved exemptions (and therefore, the Frequency extensions of SR 3.0.2 may not be applied), since the testing is an Appendix J, Type B or C test.~~ the Containment Leakage Rate Testing Program This SR simply imposes additional acceptance criteria.

Bypass leakage is considered part of L_a.

REFERENCES

1. UFSAR, Section 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Section 6.2.
4. Generic Issue B-24.
5. UFSAR, Section 6.2.4.2

ATTACHMENT 2

REPRINTED TECHNICAL SPECIFICATIONS AND BASES FOR McGUIRE

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	-------------------------------NOTES------------------------------ 1. The space between each dual ply bellows assembly on penetrations between the containment building and annulus shall be vented to the annulus during Type A tests. 2. Following each Type A test, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3 to 5 psig to verify no detectable leakage, or the assembly shall be subjected to a leak test with the pressure on the containment side of the assembly at P_a. 3. Type C tests on penetrations M372 and M373 may be performed without draining the glycol-water mixture from the seats of their diaphragm valves if meeting a zero indicated leakage rate (not including instrument error). --	
	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

ACTIONS (continued)
SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-------------------------------NOTE------------------------------ 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	Perform a pressure test on each inflatable air lock door seal and verify door seal leakage is < 15 sccm.	6 months
SR 3.6.2.3	Verify only one door in the air lock can be opened at a time.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	-------------------------------NOTE------------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --- Verify each containment isolation manual valve and blind flange that is located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for containment purge lower and upper compartment and Instrument room valves with resilient seals.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.8	Verify the combined leakage rate for all reactor building bypass leakage paths is $\leq 0.07\ L_a$ when pressurized to $\geq P_a$, 14.8 psig.	In accordance with the Containment Leakage Rate Testing Program

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 be established to implement 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-Test Program," dated September 1995.

(continued)

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 14.8 psig. The containment design pressure is 15 psig. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.3% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.75\ L_a$ for Type A tests and $< 0.6\ L_a$ for Type B and Type C tests.

b. Airlock testing acceptance criteria for the overall airlock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$. For each door, the leakage rate is $\leq 0.01\ L_a$ when tested at $\geq$ 14.8 psig.

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 10CFR50, Appendix J.

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Containment Spray, Safety Injection, Chemical and Volume Control, Nuclear Sampling, RHR, Boron Recycle, Refueling Water, Liquid Waste, and Waste Gas. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.4 Deleted

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

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 requirements of regulations. The TS cannot in and of themselves extend a test interval specified in 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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete reactor building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain the radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident (LOCA). Additionally, the containment vessel and reactor building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a flat circular base. It is completely enclosed by a reinforced concrete reactor building. An annular space exists between the walls and domes of the steel containment vessel and the concrete reactor building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The reactor building provides shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and Nuclear Steam Supply System.

The inner steel containment 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:

BACKGROUND (continued)

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 secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

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

c. All equipment hatches are closed and sealed; and

d. The sealing mechanism associated with a penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY 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 DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs 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 0.3% 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 containment leakage rate at the calculated peak containment internal 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 0.3% per day in the safety analysis at P_a = 14.8 psig (Ref. 3). Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

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

BASES

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.

Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and reactor building bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of $1.0\ L_a$.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA 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 3.9.4, "Containment Penetrations."

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 OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within

BASES

ACTIONS (continued)

36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant 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. Failure to meet specific leakage limits for the air lock, secondary containment bypass leakage path, and purge valve with resilient seals (as specified in LCO 3.6.2 and LCO 3.6.3) does not invalidate the acceptability of the overall containment leakage determinations unless the specific leakage contribution to overall Type A, B, and C leakage causes one of these overall leakage limits to be exceeded. 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 $\leq 0.75\ L_a$ for Option B for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ 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.

The Surveillance is modified by three Notes.

Note 1 requires that the space between each dual-ply bellows assembly on containment penetrations between the containment building and the annulus be vented to the annulus during each Type A test.

Note 2 requires that following each Type A test, the space between each dual-ply bellows assembly be subjected to a low pressure leak test with no detectable leakage. Otherwise, the assembly must be tested with the containment side of the bellows assembly pressurized to P_a and meet the requirements of SR 3.6.3.8 (bypass leakage requirements).

Note 3 allows penetrations M372 and M373 to be tested without draining the glycol-water mixture from the associated diaphragm valves (NF-228A, NF-233B, and NF-234A) as long as no leakage is indicated. This test may be used in lieu of 10 CFR 50, Appendix J, Option B as defined

BASES

SURVEILLANCE REQUIREMENTS (continued)

in ANSI/ANS56.8-1994 Section 3.3.5 (Test Medium). The required test pressure and interval are not changed.

All test leakage rates shall be calculated using observed data converted to absolute values. Error analysis shall also be performed to select a balanced integrated leakage measurement system.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Chapter 15.
3. UFSAR, Section 6.2.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.3% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 14.8 psig following a design basis LOCA.. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. 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 kept 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, a DBA 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 3.9.4, "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. It is preferred that the air lock be accessed from inside primary 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

BASES

ACTIONS (continued)

status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant 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 acceptance criteria were established during initial air lock and containment OPERABILITY testing. 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 a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which are 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

Door seals must be tested every 6 months to verify the integrity of the inflatable door seal. The measured leakage rate must be less than 15 standard cubic centimeters per minute (sccm) per door seal when the seal is inflated to approximately 85 psig. This ensures that the seals will remain inflated for at least 7 days should the instrument air supply to the

BASES

SURVEILLANCE REQUIREMENTS (continued)

seals be lost. The Frequency of testing has been demonstrated to be acceptable through operating experience.

SR 3.6.2.3

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 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 and 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 plant outage, and the potential for loss of containment OPERABILITY if the surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the interlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 6.2.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

SURVEILLANCE REQUIREMENTS (continued)

OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time is specified in the UFSAR and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. The measured leakage rate for containment purge lower compartment and instrument room valves must be $\leq 0.05\ L_a$ when pressurized to P_a. The measured leakage rate for containment purge upper compartment valves must be $\leq 0.01\ L_a$ when pressurized to P_a. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), these valves will not be placed on the maximum extended test interval, but tested on the nominal test interval in accordance with the Containment Leakage Rate Testing Program.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. The isolation signals involved are Phase A, Phase B, and Safety Injection. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.8

This SR ensures that the combined leakage rate of all reactor building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The Frequency is required by the Containment Leakage Rate Testing Program. This SR simply imposes additional acceptance criteria.

Bypass leakage is considered part of L_a.

REFERENCES

1. UFSAR, Section 15.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Section 6.2.
4. Generic Issue B-24.
5. UFSAR, Section 6.2.4.2

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

BACKGROUND INFORMATION

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. McGuire 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 the Local Leakage Rate Test (LLRT) to once per five year test interval versus the current two year interval. The LLRT extended test interval is based upon satisfactory performance of two "As Found" tests (test performance prior to any maintenance on the component).

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

Due to the large number of individual changes that are being made to the TS and Bases, this amendment request describes each proposed TS/TS Bases change separately. The appropriate technical justification for each proposed change is provided with the description of each proposed change for ease of reference.

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

TS 3.6.1

McGuire 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 is presently conducted according to the requirements of 10 CFR 50, Appendix J, Option A. The TS therefore have split these testing requirements into two distinct SRs, one for Type A testing and one for Types B and C testing. This proposed amendment combines all requirements for Types A, B, and C testing into one SR 3.6.1.1, which references the Containment Leakage Rate Testing Program. The frequency note in SR 3.6.1.2 concerning SR 3.0.2 not being applicable is also being deleted, as this note is not used when Option B is employed.

Compliance with the requirements of 10 CFR 50, Appendix J, is still assured. The Containment Leakage Rate Testing Program will contain specifics concerning McGuire 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-1431, "Standard Technical Specifications, Westinghouse 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.

TS 3.6.2

McGuire 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. The frequency note in SR

3.6.2.1 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.

The relocation of the acceptance criteria concerning air lock door testing to the TS Administrative Controls is consistent with NUREG-1431. This change is administrative in nature and 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-1431, and Option B of 10 CFR 50, Appendix J, the proposed changes do not adversely affect existing plant safety margins.

TS 3.6.3.6

McGuire TS SR 3.6.3.6 delineates the requirements for periodic leakage rate testing for containment purge lower and upper compartment and instrument room valves with resilient seals. These valves are presently tested once per 184 days and within 92 days after opening the valve. Plant configuration and operating procedures do not permit these valves to be open during Modes 1 through 4. These valves are considered passive within the Inservice Testing

Program; thus they can not be opened during Modes 1 through 4. Therefore, the previous requirement to leak test within 92 days after opening no longer applies.

Regulatory Guide 1.163 limits the testing frequency of these valves to 30 months as specified by ANSI/ANS 56.8-1994, with consideration given to operating experience and safety significance. Operating experience and leakage rate test data illustrate that these valves meet and exceed the performance criteria of 10 CFR 50, Appendix J, Option B.

Considering the operating experience, frequency of operation, modes of operation and leakage rate test data, these valves will be tested in accordance with Option B, but limited to the 30 month test interval.

TS 3.6.3.8

McGuire TS SR 3.6.3.8 delineates the requirements for verifying the combined leakage rate for all reactor building bypass leakage paths. The reference to 10 CFR 50, Appendix J associated with Type B and C testable penetrations has been replaced by a reference to the Containment Leakage Rate Testing Program. The frequency note in SR 3.6.3.8 concerning SR 3.0.2 not being applicable is also being deleted, as this note is not used when Option B is employed.

McGuire is proposing an additional change to SR 3.6.3.8 that is unrelated to the adoption of 10 CFR 50, Appendix J, Option B. This change deletes the surveillance note concerning soap bubble testing for penetrations that are not individually testable. This change also deletes the surveillance frequency discussion pertaining to testable and non-testable penetrations. At McGuire, the requirement for soap bubble testing applies to penetrations surrounded by guard piping or penetrations welded to the containment liner itself. For most configurations, the outside containment end of the guard pipe is seal welded to the process pipe. The rationale for conducting soap bubble testing of these types of penetrations was based on language contained in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports For Nuclear Power Plants, LWR Edition." Branch Technical Position (BTP) CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," (Revision 2 dated July 1981) states that welded joints on penetrations (e.g., guard pipes) which pass through-both the primary and secondary containment barriers should be considered potential bypass leakage paths around the leakage collection and filtration systems of the secondary containment. The BTP goes on to state that provisions should be made to permit preoperational and periodic leakage rate testing in a manner similar to the Type B or C tests of Appendix J to 10 CFR Part 50 for each bypass leakage path described in the BTP. An acceptable alternative for local leakage rate testing for welded joints would be to conduct a soap bubble test of the welds concurrently with the integrated (Type A) leakage test of the primary containment required by Appendix J. Any detectable leakage determined in this manner would require repair of the joint.

On November 14, 2000, Duke (Catawba Nuclear Station) and NRC personnel participated in a conference call concerning this issue. The NRC stated during this conference call that the statement concerning welded joints on containment penetrations in BTP CSB 6-3 was no longer the officially held position within the NRC. The NRC acknowledged that the BTP was outdated in certain respects and was in need of revision. In particular, Item 5c on welded joints needs to be deleted and Item 7 needs revision to delete the reference to soap bubble testing for welded joints. The NRC indicated that requiring such a test of individual welds would be tantamount to requiring local leak rate testing on individual sections of the steel containment vessel itself. This issue was documented in a November 21, 2000 letter from Gary R. Peterson, Duke Energy Corporation, to the NRC.

McGuire is therefore proposing these additional changes to SR 3.6.3.8 to delete the requirement for soap bubble testing on the grounds that the basis for the testing requirement has been superceded and that the test provides no additional assurance concerning the integrity of the welds themselves.

McGuire is also proposing to place "≥" in front of "P_a, 14.8 psig" to be consistent with TSTF-52.

TS 5.5.2

McGuire TS 5.5.2 (Containment Leakage Rate Testing Program) is modified by this proposed amendment to delete the reference to Type A testing and to incorporate the acceptance criteria for Types B and C testing and air lock testing. Note that the inequality sign associated with the acceptance criteria for each air lock door has been changed to "≤" as opposed to the "<" sign contained in SR 3.6.2.1, Item B. This change is made for consistency with TSTF-52. In addition, a sentence is added concerning the containment design pressure of 15 psig to conform to TSTF-52. Also, under Item a, the inequality sign associated with the Type A test acceptance criteria has been changed from "<" to "≤" to be consistent with TSTF-52. Finally, the sentence pertaining to SR 3.0.2 not being applicable is deleted and replaced with the statement, "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J." This change was made to conform to TSTF-52 and is purely editorial (i.e., the effect of the two statements is equivalent).

The Containment Leakage Rate Testing Program is defined in accordance with the requirements of 10 CFR 50.54(o) and Option B to 10 CFR 50, Appendix J as modified by currently approved exemptions. The program is in accordance with the guidelines contained within Regulatory Guide 1.163.

The changes include the incorporation of leakage rate acceptance criteria for Type B and Type C tests. Leakage rate acceptance criteria for primary containment air locks are also incorporated. No changes were made to these acceptance criteria in their incorporation within the TS Administrative Controls; therefore, existing safety margins remain unaffected by these changes.

<u>TS Bases 3.0</u>

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 deleted wording in SR 3.0.2 was originally adopted during McGuire's conversion to the Improved TS and reflected the then-current NRC model for implementation of Option B enclosed in a letter from C.I. Grimes, NRC, to D.J. Modeen, NEI, dated November 2, 1995. This guidance has been, superceded by TSTF-52.

TS Bases 3.6.1

The "Background" and "Applicable Safety Analyses" sections have been editorially revised to be consistent with TSTF-52. Changes to the "LCO" section have been made consistent with the changes to the TS themselves. The Bases for SR 3.6.1.1 are revised consistent with the changes to SR 3.6.1.1 and the deletion of SR 3.6.1.2 and are consistent with TSTF-52. Finally, the "References" section is modified (Reference 1) to reflect Option B, consistent with TSTF-52.

TS Bases 3.6.2

The "Applicable Safety Analyses" section is revised to include the appropriate reference to Option B and makes an editorial revision in accordance with TSTF-52. The Bases for SR 3.6.2.1 are revised consistent with the changes proposed to the SR itself. The Bases for SR 3.6.2.2 are also being revised to state that the SR frequency has been appropriately demonstrated through operating experience. The 6-month frequency for this SR was originally contained in the old McGuire TS (pre-Improved TS version) and was intended to be consistent with the 6-month frequency required for the overall airlock leakage test required by Option A. Any change to this 6-month frequency would have to be justified separately from the adoption of 10 CFR 50, Appendix J, Option B. McGuire is therefore not proposing a change to the 6-month frequency for testing the door seals at this time; hence, the existing Bases discussion would be misleading after Option B is adopted and is being editorially revised. Also, the L_a value of 0.30% is being revised to 0.3% to be consistent with TSTF-52. Finally, the "References" section is revised (Reference 1) to add a reference to Option B.

TS Bases 3.6.3

The Bases for SR 3.6.3.6 and SR 3.6.3.8 are revised to be consistent with changes to the TS and are consistent with TSTF-52. Additionally, the SR 3.6.3.8 Bases are revised to delete the reference to soap bubble testing, consistent with the changes proposed to SR 3.6.3.8 itself.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. Involve a significant reduction in a margin of safety.

First Standard

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. 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, the proposed amendment does not increase the consequences of any accident previously evaluated.

Second Standard

The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. 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 McGuire's 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 safety 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.

Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. 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. Similar proposed changes have been previously reviewed and approved by the NRC, and they are applicable to McGuire.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

ATTACHMENT 5

ENVIROMENTAL ANALYSIS

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

This amendment to the McGuire TS allows for the implementation of 10 CFR 50, Appendix J, Option B for Types B and C testing. Implementation of this amendment will have no adverse impact upon the McGuire units; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the McGuire TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.