Duke Power
A Duke Energy Company

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April 2, 2003

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

Subject: Duke Energy Corporation
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Proposed Technical Specifications and Bases Amendment
3.6.16.1 and 3.6.16.3; and Administrative Controls 5.5.2
Reactor Building Integrity

Enclosed as Attachment 1 to this letter, please find McGuire Technical Specification pages 3.6.16-1, 3.6.16-2, 5.5-1, 5.5-2, and McGuire Technical Specification Bases pages B3.6.16-1, B3.6.16-2, and B3.6.16-3 which were affected by our proposed request of September 30, 2002. Since our September 30, 2002 request, License Amendment 211/192, approved by the NRC Staff on March 12, 2003, revised Section 5.5.2, "Containment Leak Rate Testing Program," by adding an exception regarding the conduct of subsequent Unit 1 and 2 Type A testing. The approval of this license amendment request has necessitated the revision of pages 5.5-1 and 5.5-2 from those included with our September 30, 2002 submittal to include Amendment 221/192. No additional changes are made.

Please note that the revision number of the McGuire Technical Specification Bases pages should be Revision 40.

Please address any additional inquiries in this matter to J. A. Effinger at (704) 382-8688.

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Yours truly,

W. R. Mc Collum, Jr.
Senior Vice President
Nuclear Support

W. R. Mc Collum, Jr., being duly sworn, affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

W R McCollum Jr.
W. R. Mc Collum, Jr., Senior Vice President

Subscribed and sworn to me: April 2, 2003
Date

Mary P. Nelms, Notary Public

My commission expires: JAN 22, 2006

SEAL

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ATTACHMENT 1

McGUIRE UNITS 1 AND 2 TECHNICAL SPECIFICATIONS
AND
TECHNICAL SPECIFICATION BASES

REPRINTED VERSION

Remove Page	Insert Page
3.6.16-1	3.6.16-1
3.6.16-2	3.6.16-2
B3.6.16-1	B3.6.16-1
B3.6.16-2	B3.6.16-2
B3.6.16-3	B3.6.16-3
5.5-1	5.5-1
5.5-2	5.5-2

3.6 CONTAINMENT SYSTEMS

3.6.16 Reactor Building

LCO 3.6.16 The reactor building shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	24 hours
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 3.6.16.1 Verify the door in each access opening is closed, except when the access opening is being used for normal transit entry and exit.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.16.2	Verify each Annulus Ventilation System train produces a pressure equal to or more negative than -0.5 inch water gauge in the annulus within 22 seconds after a start signal and -3.5 inches water gauge after 48 seconds. Verifying that upon reaching a negative pressure of -3.5 inches water gauge in the annulus, the system switches into its recirculation mode of operation and that the time required for the annulus pressure to increase to -0.5 inch water gauge is ≥ 278 seconds.	18 months on a STAGGERED TEST BASIS
SR 3.6.16.3	Verify reactor building structural integrity by performing a visual inspection of the exposed interior and exterior surfaces of the reactor building.	3 times every 10 years, coinciding with containment visual examinations required by SR 3.6.1.1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.16 Reactor Building

BASES

BACKGROUND

The reactor building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the reactor building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Annulus Ventilation System (AVS) establishes a negative pressure in the annulus between the reactor building and the steel containment vessel under post accident conditions. Filters in the system then control the release of radioactive contaminants to the environment. The reactor building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the AVS. To ensure the retention of containment leakage within the reactor building:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

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

APPLICABLE SAFETY ANALYSES

The design basis for reactor building OPERABILITY is a LOCA. Maintaining reactor building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The reactor building satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Reactor building OPERABILITY must be maintained to ensure proper operation of the AVS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

BASES

APPLICABILITY

Maintaining reactor building OPERABILITY prevents leakage of radioactive material from the reactor building. Radioactive material may enter the reactor building from the containment following a LOCA. Therefore, reactor building OPERABILITY is required in MODES 1, 2, 3, and 4 when a steam line break, LOCA, or rod ejection accident could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, reactor building OPERABILITY is not required in MODE 5 or 6.

ACTIONS

A.1

In the event reactor building OPERABILITY is not maintained, reactor building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

B.1 and B.2

If the reactor building 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 3.6.16.1

Maintaining reactor building OPERABILITY requires maintaining the door in each access opening closed, except when the access opening is being used for normal transit entry and exit. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.16.2

The ability of a AVS train to produce the required negative pressure within the required times provides assurance that the building is adequately sealed. The negative pressure prevents leakage from the building, since outside air will be drawn in by the low pressure. The negative pressure must be established within the time limit to ensure that no significant quantity of radioactive material leaks from the reactor building prior to developing the negative pressure.

The AVS trains are tested every 18 months on a STAGGERED TEST BASIS to ensure that in addition to the requirements of LCO 3.6.10, "Annulus Ventilation System," either AVS train will perform this test. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage.

SR 3.6.16.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the reactor building. The Frequency is based on engineering judgment and is the same as that for containment visual inspections performed in accordance with SR 3.6.1.1.

REFERENCES

1. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

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, as modified by the following exceptions:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 27, 1993 (Unit 1) and August 20, 1993 (Unit 2) Type A test shall be performed no later than May 26, 2008 (Unit 1) and August 19, 2008 (Unit 2), and

5.5 Programs and Manuals (continued)

b. The containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR 3.6.11 Type A test, prior to initiating the Type A test.

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