March 21, 1997

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Mr. H. B. Barron
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: ISSUANCE OF AMENDMENTS - McGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M97755 AND M97756)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 173 to Facility Operating License NPF-9 and Amendment No. 155 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 13, 1997.

The amendments revise the TS so that the containment integrated leak rate Type A testing will now be performed consistent with the revised 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." No changes to implement Option B for the Type B and Type C tests were requested by the licensee at this time.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Victor Nerses, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 173 to NPF-9
2. Amendment No. 155 to NPF-17
3. Safety Evaluation

cc w/encls: See next page

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NAME	V.NERSES:cn	L.BERRY	R Bachmann	H.BERKOW
DATE	2/27/97	2/26/97	2/28/97	3/21/97
COPY	YES NO	YES NO	YES NO	YES NO

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 21, 1997

Mr. H. B. Barron
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: ISSUANCE OF AMENDMENTS - McGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M97755 AND M97756)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 173 to Facility Operating License NPF-9 and Amendment No. 155 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 13, 1997.

The amendments revise the TS so that the containment integrated leak rate Type A testing will now be performed consistent with the revised 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program." No changes to implement Option B for the Type B and Type C tests were requested by the licensee at this time.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Victor Nerses

Victor Nerses, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 173 to NPF-9
2. Amendment No. 155 to NPF-17
3. Safety Evaluation

cc w/encls: See next page

cc:
Mr. Paul R. Newton
Legal Department, (PB05E)
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242-0001

County Manager of Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Mr. J. E. Snyder
Regulatory Compliance Manager
Duke Power Company
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U. S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Mr. Peter R. Harden, IV
Account Sales Manager
Westinghouse Electric Corporation
Power Systems Field Sales
P. O. Box 7288
Charlotte, North Carolina 28241

Dr. John M. Barry
Mecklenberg County
Department of Environmental Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. G. A. Copp
Licensing - EC050
Duke Power Company
526 South Church Street
Charlotte, North Carolina 28242

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Mr. Dayne H. Brown, Director
Division of Radiation Protection
North Carolina Department of Environmental Health and Natural Resources
P. O. Box 27687
Raleigh, North Carolina 27611-7687

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
License No. NPF-9

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Facility Operating License No. NPF-9 filed by the Duke Power Company (licensee) dated January 13, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 173 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Technical Specification
Changes

Date of Issuance: March 21, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 173

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-9	3/4 6-9
3/4 6-10	3/4 6-10
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.30% by weight of the containment air per 24 hours at P_a, 14.8 psig,

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, and

c. A combined bypass leakage rate of less than 0.07 L_a for all penetrations identified as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or (b) the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) the combined bypass leakage rate exceeding 0.07 L_a, restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.07 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Type A containment leakage rates shall be demonstrated as required by 10 CFR 50.54(o) and Appendix J of 10 CFR 50, Option B, as modified by approved exemptions, and in accordance with the guidelines of Regulatory Guide 1.163, September, 1995.

SURVEILLANCE REQUIREMENTS (Continued)

a. Deleted

b. Deleted

c. The accuracy of each Type A test shall be verified by a supplemental test in accordance with Regulatory Guide 1.163, September, 1995.

d. Type B and C tests shall be conducted, in accordance with 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option A, with gas at P_a, 14.8 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks,

   2) Dual-ply bellows assemblies on containment penetrations between the containment building and the annulus, and

   3) Purge supply and exhaust isolation valves with resilient material seals.

   4) Type C tests performed on containment penetrations M372, M373 without draining the glycol-water mixture from the seats of their diaphragm valves (NF-228A, NF-233B, and NF-234A), if meeting a zero indicated leakage rate (not including instrument error) for the diaphragm valves. These tests may be used in lieu of tests which are otherwise required by Section III.C.2(a) of 10 CFR 50, Appendix J to use air or nitrogen as

SURVEILLANCE REQUIREMENTS (Continued)

the test medium. The above required test pressure (P_a) and test interval are not changed by this exception.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.9.3 or 4.6.1.9.4, as applicable;

f. The combined bypass leakage rate shall be determined to be less than 0.07 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 14.8 psig during each Type A test;

g. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3;

h. The space between each dual-ply bellows assembly on containment penetrations between the containment building and the annulus shall be vented to the annulus during Type A tests. Following completion of each Type A test, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3-5 psig to verify no detectable leakage or the dual-ply bellows assembly shall be subjected to a leak test with the pressure on the containment side of the dual-ply bellows assembly at P_a, 14.8 psig to verify the leakage to be within the limits of Specification 4.6.1.2f.;

i. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced Integrated Leakage Measurement System; and

j. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test (reference Specification 4.6.1.2) to verify no apparent changes in appearance of the surfaces or other abnormal degradation. If the Type A test is performed at 10-year intervals, two additional inspections shall be performed at approximately equal intervals during shutdowns between Type A tests. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73.

CONTAINMENT SYSTEMS

REACTOR BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the reactor building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

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

ACTION:

With the structural integrity of the reactor building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the reactor building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the reactor building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. If the Type A test is performed at 10-year intervals, two additional inspections shall be performed at approximately equal intervals during shutdowns between Type A tests. Any abnormal degradation of the reactor building detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72, and 50.73.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring Type A leakage rates is consistent with the requirements of Appendix J of 10 CFR 50, Option B. Type B and C tests are conducted in conformance with 10 CFR 50 Appendix J, Option A.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 1.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 15 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 14.5 psig. The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 14.8 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that: (1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions, and (2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 100°F for the lower compartment, 75°F for the upper compartment and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to 14.8 psig which is less than the containment design pressure of 15 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 15 psig in the event of a LOCA. A periodic visual inspection is sufficient to demonstrate this capability.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. NPF-17

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Facility Operating License No. NPF-17 filed by the Duke Power Company (licensee) dated January 13, 1997, applies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 155 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: March 21, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-9	3/4 6-9
3/4 6-10	3/4 6-10
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.30% by weight of the containment air per 24 hours at P_a, 14.8 psig,

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, and

c. A combined bypass leakage rate of less than 0.07 L_a for all penetrations identified as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or (b) the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) the combined bypass leakage rate exceeding 0.07 L_a, restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.07 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Type A containment leakage rates shall be demonstrated as required by 10 CFR 50.54(o) and Appendix J of 10 CFR 50, Option B, as modified by approved exemptions, and in accordance with the guidelines of Regulatory Guide 1.163, September, 1995.

a. Deleted

b. Deleted

c. The accuracy of each Type A test shall be verified by a supplemental test in accordance with Regulatory Guide 1.163, September, 1995.

d. Type B and C tests shall be conducted, in accordance with 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option A, with gas at P_a, 14.8 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks,

   2) Dual-ply bellows assemblies on containment penetrations between the containment building and the annulus, and

   3) Purge supply and exhaust isolation valves with resilient material seals.

   4) Type C tests performed on containment penetrations M372, M373 without draining the glycol-water mixture from the seats of their diaphragm valves (NF-228A, NF-233B, and NF-234A), if meeting a zero indicated leakage rate (not including instrument error) for the diaphragm valves. These tests may be used in lieu of tests which are otherwise required by Section III.C.2(a) of 10 CFR 50, Appendix J to use air or nitrogen as

SURVEILLANCE REQUIREMENTS (Continued)

the test medium. The above required test pressure (P_a) and test interval are not changed by this exception.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.9.3 or 4.6.1.9.4, as applicable;

f. The combined bypass leakage rate shall be determined to be less than 0.07 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 14.8 psig during each Type A test;

g. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3;

h. The space between each dual-ply bellows assembly on containment penetrations between the containment building and the annulus shall be vented to the annulus during Type A tests. Following completion of each Type A test, the space between each dual-ply bellows assembly shall be subjected to a low pressure test at 3-5 psig to verify no detectable leakage or the dual-ply bellows assembly shall be subjected to a leak test with the pressure on the containment side of the dual-ply bellows assembly at P_a, 14.8 psig to verify the leakage to be within the limits of Specification 4.6.1.2f.;

i. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced Integrated Leakage Measurement System; and

j. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test (reference Specification 4.6.1.2) to verify no apparent changes in appearance of the surfaces or other abnormal degradation. If the Type A test is performed at 10-year intervals, two additional inspections shall be performed at approximately equal intervals during shutdowns between Type A tests. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72 and 50.73.

CONTAINMENT SYSTEMS

REACTOR BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the reactor building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

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

ACTION:

With the structural integrity of the reactor building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the reactor building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the reactor building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. If the Type A test is performed at 10-year intervals, two additional inspections shall be performed at approximately equal intervals during shutdowns between Type A tests. Any abnormal degradation of the reactor building detected during the above required inspections shall be reported to the Commission pursuant to 10 CFR Sections 50.72, and 50.73.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring Type A leakage rates is consistent with the requirements of Appendix J of 10 CFR 50, Option B. Type B and C tests are conducted in conformance with 10 CFR 50 Appendix J, Option A.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 1.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 15 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 14.5 psig. The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 14.8 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that: (1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions, and (2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 100°F for the lower compartment, 75°F for the upper compartment and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to 14.8 psig which is less than the containment design pressure of 15 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 15 psig in the event of a LOCA. A periodic visual inspection is sufficient to demonstrate this capability.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated Jauary 13, 1997, Duke Power Company (DPC or the licensee) submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would permit implementation of 10 CFR Part 50, Appendix J, Option B, for the Type A containment integrated leak rate tests. The TS contain a reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Appendix J, Option B.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage rate postulated in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety, which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide (RG) 1.163, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions, which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the McGuire Nuclear Station, Units 1 and 2, TS.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed model TS to implement Option B. After some discussion, the staff and NEI agreed on a final TS, which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that a licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, a licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 <u>EVALUATION</u>

The proposed amendments would replace the prescriptive Type A containment leakage rate testing requirements of 10 CFR Part 50, Appendix J, with the performance-based requirements of the revised Option B of Appendix J. This requires changes to existing TS as discussed in detail below. Corresponding Bases were also modified.

The proposed amendments meet the particular requirement of Option B that the implementation document used by the licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS by including RG 1.163 in TS 4.6.1.2. This guide specifies a method acceptable to the NRC for complying with Option B.

TS 4.6.1.2 is modified to require that the containment leakage rates shall be demonstrated in accordance with a test schedule determined in conformance with Appendix J, Option B, using the methods and provisions of RG 1.163.

TS 3.6.1.2.a and its corresponding ACTION statement, TS 4.6.1.2.f, and TS 4.6.1.2.h have been modified by deleting the provisions to conduct Type A integrated leak rate testing (CILRT) at a reduced pressure since reduced pressure testing is not included in the revised Appendix J, Option B rule.

The Surveillance Requirements in TS 4.6.1.2.a and b. have been deleted because their provisions have been superseded by the modified requirements of Option B.

TS 4.6.1.2.c has been modified such that the accuracy of each Type A test shall be verified by a supplemental test in accordance with RG 1.163.

TS 4.6.1.2.d provides clarification that Type B and Type C tests wil continue to be performed according to the provisions of 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option A.

TS 4.6.1.6 and 4.6.1.7 have been modified such that the current frequency of conducting the visual inspections of the containment vessel and the reactor building of three times per 10-year interval will be maintained. This is consistent with the guidance of RG 1.163, Position 3, and is acceptable.

Appendix J, Option B permits a licensee to choose Type A; or Types B and C; or Types A, B, and C testing to be done on a performance basis. The licensee has elected to perform the Type A testing on a performance basis. The licensee has not proposed to perform the Type B and Type C tests in accordance with Option B at this time.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the model TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 6575 dated February 12, 1997). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: S.S. Kirslis

Date: March 21, 1997