February 13, 2008

Mr. G. R. Peterson
Vice President
McGuire Nuclear Station
Duke Power Company LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: MCGUIRE NUCLEAR STATION, UNIT 1 ISSUANCE OF AMENDMENT REGARDING EXTENSION OF APPENDIX J, TYPE A INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MD4654)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 244 to Renewed Facility Operating License NPF-9 for the McGuire Nuclear Station, Unit 1 (McGuire Unit 1). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 21, 2007, as supplemented August 9, 2007.

The amendment revises administrative TS 5.5.2, “Containment Leak Rate Testing Program,” from the currently approved 15-year interval (since the last McGuire Unit 1 Type A test) to a frequency encompassing the end of the McGuire Unit 1 End of Cycle (EOC) 19 refueling outage (approximately 6 months beyond the present frequency).

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,

***/RA/***

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-369

Enclosures:
1. Amendment No. 244 to NPF-9
2. Safety Evaluation

cc w/encls: See next page

February 13, 2008

Mr. G. R. Peterson
Vice President
McGuire Nuclear Station
Duke Power Company LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: MCGUIRE NUCLEAR STATION, UNIT 1 ISSUANCE OF AMENDMENT REGARDING EXTENSION OF APPENDIX J, TYPE A INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MD4654)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 244 to Renewed Facility Operating License NPF-9 for the McGuire Nuclear Station, Unit 1 (McGuire Unit 1). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 21, 2007, as supplemented August 9, 2007.

The amendment revises administrative TS 5.5.2, “Containment Leak Rate Testing Program,” from the currently approved 15-year interval (since the last McGuire Unit 1 Type A test) to a frequency encompassing the end of the McGuire Unit 1 End of Cycle (EOC) 19 refueling outage (approximately 6 months beyond the present frequency).

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,

***/RA/***

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-369

Enclosures:
1. Amendment No. 244 to NPF-9
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION: Public
LPL2-1 R/F
RidsNrrDorlLpl2-1
RidsRgn2MailCenter (JMoorman)
RidsNrrLAMO'Brien (hard copy)
RDenning, NRR
KManoly, NRR
RidsAcrsAcnwMailCenter
GHill (2 hard copies)
RidsNrrDirsItsb (TKobetz)
RidsNrrDe (ZXi)
RidsNrrDss (BLee)
GThomas, NRR
RidsOgcRp
RidsNrrDorlDpr
RidsNrrPMJStang
RidsNrrDe (HAshar)
RidsNrrDss (RPalla)
MRubin, NRR

ADAMS Accession No.: ML073400670 Pkg. No. ML073400621 Tech Spec No.: ML080460565

*SE transmitted by memo dated

| OFFICE | NRR/LPL2-1/PM | NRR/LPL2-1/LA | OGC | NRR/EMCB/BC | NRR/APLA/BC | NRR/SCVB/BC | NRR/LPL2-1/BC (A) |
|---|---|---|---|---|---|---|---|
| NAME | JStang | MO'Brien | SUttal | KManoly* | MRubin* | RDenning* | MWong |
| DATE | 02/04/08 | 02/04/08 | 1/20/08 | 09/21/07 | 06/26/07 | 06/26/07 | 02/04/08 |

OFFICIAL RECORD COPY

DUKE POWER COMPANY LLC

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 244
Renewed 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), Renewed Facility Operating License No. NPF-9, filed by the Duke Power Company LLC (licensee), dated February 21, 2007, as supplemented August 9, 2007, 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 Renewed Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 244, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

Melanie C. Wong, Acting Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-9
and the Technical Specifications

Date of Issuance: February 13, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 244

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| License Page | License Page |
| NPF page 3 | NPF page 3 |
| TS Page | TS Page |
| 5.5-1 | 5.5-1 |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 244 TO RENEWED FACILITY OPERATING LICENSE NPF-9

DUKE POWER COMPANY LLC

MCGUIRE NUCLEAR STATION, UNIT 1

DOCKET NO. 50-369

## 1.0 INTRODUCTION

By application dated February 21, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070650355), as supplemented by letter dated August 9, 2007 (ADAMS Accession No. ML072330118), Duke Power Company LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the McGuire Nuclear Station, Unit 1 (McGuire 1).

The proposed change would revise administrative TS 5.5.2, “Containment Leak Rate Testing Program,” from the currently approved 15-year interval (since the last McGuire 1 Type A test) to a frequency encompassing the end of the McGuire 1 End of Cycle (EOC) 19 refueling outage (approximately 6 months beyond the present frequency).

## 2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. McGuire 1 TS 5.5.2, “Containment Leakage Rate Testing Program,” requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at McGuire 1 have been successful, so the current interval requirement would normally be 10 years. However, by letter dated May 29, 2002 (ADAMS ML No. ML021580235), as supplemented by letters dated September 25, 2002 (ADAMS ML No. ML022750482), and November 12, 2002

(ADAMS ML No. ML023330411), and January 8, 2003 (ADAMS ML No. ML030210432), and January 29, 2003 (ADAMS ML No. ML030420407), the licensee requested a one-time extension of the test interval to 15 years. On March 12, 2003, the U.S. Nuclear Regulatory Commission (NRC) staff granted this request in License Amendment No. 211. The licensee is requesting a change to TS 5.5.2 which would alter their exception from the guidelines of RG 1.163 and NEI 94-01, Revision 0, by adding approximately 6 more months to the 5-year extension already in place, for a total interval of approximately 15 years and 6 months. Specifically, the exception states that the first Type A test performed after the May 27, 1993, Type A test shall be performed no later than plant restart after the end of cycle 19 refueling outage. McGuire 1 refueling outage 19 is scheduled in the fall of 2008.

The local leakage rate tests (LLRTs) (Type B and Type C tests), including their schedules, are not affected by this request.

## 3.0 TECHNICAL EVALUATION

### 3.1 General

McGuire 1 is a Westinghouse pressurized-water reactor with an ice-condenser primary containment structure. The containment consists of a free-standing cylindrical steel structure enclosed by a separate reinforced-concrete reactor shield building. The containment pressure boundary consists of the steel wall, containment access penetrations, and penetrations for process piping and electrical wiring. The overall integrity of the containment structure is verified by a Type A integrated leak rate test (ILRT) and the integrity of the penetrations and isolation valves are verified by Type B and Type C LLRTs as required by 10 CFR Part 50, Appendix J. These tests are performed to verify the essentially leak-tight characteristics of the containment structure at the design-basis accident pressure. The licensee’s Containment Leakage Testing Program currently implements the performance-based Option B of the 10 CFR Part 50, Appendix J. The most recent two Type A tests at McGuire 1 (May 1993 and May 1990) have been successful, so the current interval requirement, per NEI 94-01, would normally be 10 years. However, in March 2003, the NRC approved License Amendment No. 211 following the licensee’s request for a one-time extension of the ILRT interval from 10 years to 15 years. The NRC staff approval of the ILRT interval extension to 15 years is documented in the safety evaluation (SE) for License Amendment No. 211. The ILRT extension was justified by historical performance of the containment, based on results from the licensee’s primary containment leakage testing program and inservice inspection (ISI) program, supported by a risk-informed analysis.

In the February 21, 2007, application, the licensee requested a one-time extension of the containment Type A test interval by an additional 6 months beyond the current 15 years to enable the next ILRT to be performed prior to startup from the 1EOC19 refueling outage scheduled for fall 2008. The licensee justified the request based on the results of previous ILRTs, containment inspection programs, and a revised risk-informed analysis to reflect the extended interval.

### 3.2 Containment ISI Program and Structural Integrity Considerations

The licensee provided a tabulation of the test results of all previous ILRTs performed on McGuire 1. These results were previously reported in the licensee’s application for License Amendment No. 211. The as-found and as-left leakage from the last ILRT, conducted in May 1993, was 0.1482 weight-percent per day. The licensee stated that the previous ILRT test results confirmed that the containment structure leakage is acceptable, with considerable margin, compared to the TS acceptance criterion of 0.30 percent of primary containment air weight per day (1.0 La).

The leakage rate testing requirements of 10 CFR 50, Appendix J, Option B (ILRT and LLRT) and the containment inservice inspection (ISI) requirements mandated by 10 CFR 50.55a together help ensure the continued leak-tight and structural integrity of the containment during its service life. Therefore, the NRC staff's review of the application for License Amendment No. 211 concentrated on the ISI program for management of containment degradation. Since the 6-month test interval extension requested by the licensee is relatively small, the same general methodology is used in this evaluation to assess the current condition of structural and leak-tight integrity of the McGuire 1 containment. The licensee stated that the McGuire 1 steel containment vessel is examined in accordance with the requirements of an the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (Code), Section XI Program, the nuclear generation department coating program, and the containment leakage rate test program.

The ASME Section XI Program requires that the steel containment vessel be examined in accordance with the requirements of the ASME Code, Section XI, Subsection IWE, and associated modifications and limitations imposed by 10 CFR 50.55a(b)(2). These examinations have been performed using the 1992 Edition with the 1992 Addenda during the first inservice inspection interval, and the 1998 Edition with the 2000 Addenda during the second inservice inspection interval (which commenced July 15, 2005).

The nuclear generation department coating program requires a visual examination to be performed to assess and document the condition of nuclear safety-related protective coatings located inside primary containment during each refueling outage. The interior surfaces of the containment vessel are included in the scope of this examination.

The containment leakage rate test program controls leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, and as modified by approved amendments. As part of this program, the accessible interior and exterior surfaces of the primary containment are visually examined for structural deterioration which may affect the containment leak-tight integrity at a frequency of 3 times every 10 years, including during each shutdown for a Type A test prior to initiating the Type A test. These visual examination requirements are satisfied by performing the ASME Code, Section XI, Subsection IWE, IWE-2500, Table IWE-2500-1, Examination Category E-A, Item E1.11 examinations in accordance with the McGuire 1 containment structural integrity inspection procedure.

In addition to the programs identified above, McGuire 1 Technical Specification 3.6.16 requires that the structural integrity of the reactor shield building be verified by performing a visual inspection of the exposed interior and exterior surfaces 3 times every 10 years, coinciding with visual examination of the steel containment vessel. Because the reactor shield building acts as a secondary containment and protects the primary containment from the effects of weather, these visual examinations help to assure the integrity of the primary containment.

The licensee stated that examination results from the above programs have revealed no significant degradation that could affect either the containment vessel structural integrity or leak-tightness. The licensee summarized results, of the containment ISI program examinations completed on McGuire 1 to date since March 12, 2003, when License Amendment No. 211 was granted that extended the ILRT interval from 10 years to15 years. These containment inspection results are discussed in the following paragraphs.

During refueling outage 1EOC16 (2004), the following containment inspections were performed: General visual examination on accessible surface areas of the containment vessel in accordance with Subsection IWE (1992 Edition with the 1992 Addenda), Table IWE-2500-1, Examination Category E-A, Item E1.11; VT-3 visual examinations on selected moisture barriers in accordance with Table IWE-2500-1, Examination Category E-D, Item E5.30; general visual examination of accessible interior and exterior surfaces of the Reactor Shield Building; and the coatings program visual examinations. The licensee noted that although the inspections revealed no unacceptable or significant degradation, some conditions such as moisture barrier degradation, leaked boric acid crystals from components, and protective coatings degradation were observed at a number of locations on the containment vessel and noted in the licensee's corrective action program. In a request for additional information (RAI), the Nuclear Regulatory Commission (NRC) staff requested the licensee to provide information about the specific locations of the associated conditions on containment surface areas and the corrective actions that have been taken or will be taken to prevent potential further degradation at these locations. The licensee provided responses to the RAI questions by letter dated August 9, 2007.

In the response to RAI 1, with regard to moisture barrier indications, the licensee provided details of seven locations at the embedment zone interface on the exterior side of the containment vessel where missing or degraded moisture barrier materials were observed and documented in the corrective action program. The licensee stated that these indications were corrected by restoring the associated sealant materials during refueling outage 1EOC17 (2005). With regard to indications where boric acid crystals were observed, the licensee provided detailed information of (i) areas on the interior surface of the containment vessel shell with rust and boron stains; (ii) leakage from two pumps that caused boric acid crystals on the annulus floor and base of the containment vessel and on stiffening ring on exterior of the containment vessel; and (iii) leakage from a valve that caused borated water to contact the containment vessel at the embedment zone on the exterior side. For each of these indications, the licensee stated that the condition was documented in the corrective action program, the affected areas were cleaned, recoated (where needed), reinspected and found to be acceptable. The conditions in item (i) above were also recommended for continued monitoring during future examinations. With regard to the indications of coatings degradation, the licensee provided detailed information of locations on the containment vessel and its stiffening ring/welds where conditions of minor rusting and corrosion were observed. In one location, staining on the floor indicated that borated water might have migrated to the containment vessel shell surfaces behind the labyrinth door (access to Fuel Transfer Tube area on exterior of containment vessel). The area behind the labyrinth door was inspected and a very thin film of boron was observed on the floor. There were no signs of corrosion or any other problems noted. For each of these indications, the conditions were documented in the examination record and as part of the corrective action program, the affected areas were cleaned and recoated during 1EOC17. The licensee also provided information of conditions observed on the equipment hatch cover and barrel. These included areas of (i) flaking paint and minor corrosion at the bottom of the barrel at the base of the cover; (ii) corrosion at the bottom of the barrel (portion that extends through reactor shield building); and (iii) minor dings in coatings at various locations along the interior of the barrel and on the exterior of the hatch cover. Corrective actions for these documented indications were completed during refueling outage 1EOC17. The staff finds that the licensee provided detailed information of the locations and corrective actions taken for the conditions noted in RAI 1 and the corrective actions taken are adequate to address the respective condition; and, therefore, the response to the RAI 1 is acceptable.

During refueling outage 1EOC17 (2005), the licensee stated that the following examinations were performed: (i) VT-1 visual examinations on 35 percent of all pressure retaining bolted connections with no unacceptable conditions observed; (ii) VT-3 visual examinations on the containment lower airlock barrel supports with acceptable results; and (iii) coating program visual examinations which revealed areas where corrective coatings maintenance was warranted, but no significant degradation on the containment vessel surfaces.

Since the summary of refueling outage 1EOC17 (2005) examinations in the licensee's application dated February 21, 2007, did not make any mention about the conditions observed and previously discussed with regard to the moisture barrier, boric acid crystals and coatings degradation during refueling outage 1EOC16 (2004), in RAI 2 the staff requested the licensee to describe the findings and compare locations and degradations levels in 2005 with those in 2004. The licensee responded that a general visual examination was not performed during refueling outage 1EOC17. However, a general visual examination was recently completed during refueling outage 1EOC18 in accordance with the ASME Code, Section XI, Subsection IWE (1998 Edition through the 2000 Addenda), IWE-2500, Table IWE-2500-1, Examination Category E-A, Item E1.11 in 2007. The visual examination results were consistent with the results reported during previous examinations, and included a description of areas requiring coatings maintenance, minor corrective actions to restore some moisture barrier conditions, and cleaning of surfaces exposed to staining. The licensee provided a detailed description of the most significant results of this general visual examination. This included a comparison of results of examination performed on the indications identified in 1EOC16 (2004) and discussed in the response to RAI-1 and is summarized below.

The licensee stated that no indications were recorded on the previously corrected moisture barrier indications. With regard to the boric acid crystal indications, boric acid crystals were again observed at the two pump locations. These areas were cleaned to the extent needed to perform an examination of the containment vessel and embedment zone moisture barrier, and the condition of these items were found to be acceptable at these locations. The other two locations where boric acid crystals were previously observed were clean, dry, with no evidence of boron and acceptable. With regard to the coatings degradation indications observed during 1EOC16 and corrective maintenance performed, conditions were unchanged from the previous examination. However, some areas continue to warrant corrective coatings maintenance but the condition of the containment vessel was acceptable in these areas.

In response to RAI 2, the licensee also provided a detailed description of significant examination results observed during the visual examination performed during refueling outage 1EOC18 for items other than those identified in the previous paragraph. These included indications of missing moisture barrier, delaminated topcoat, flaking paint, and light rust on the internal vessel of the containment surface. These conditions were entered into the corrective action program for future coatings maintenance. The moisture barrier condition was corrected. In an area with indications of boron residue, flaking paint, and light rusting on the containment vessel interior surface, the boron was removed, the area cleaned and containment coatings maintenance was performed during 1EOC18. The licensee stated that the condition of the containment vessel was acceptable at each of these locations.

In the response to RAI 2, the licensee further noted a condition where boron, debris, rust, and staining were observed on the containment vessel shell interior surfaces and surfaces of penetrations Mk. M-302 and M-278 within the ECCS sump area at azimuth 184 degrees and azimuth 176 degrees. These areas had previously been considered inaccessible for general visual examination because of debris screens which prevented access for visual examination. However, ECCS sump modifications conducted during 1EOC18 removed these screens, providing access to perform these visual examinations. Boron identified at this location was entered into the licensee's corrective action program. The boron, debris, and corrosion was removed, a VT-1 examination was performed on surfaces in this area, including containment vessel surfaces below the concrete floor interface (some concrete was removed to perform this examination), and ultrasonic thickness measurements were performed to confirm the containment vessel shell thickness in these areas. The ultrasonic thickness measurements revealed no significant wall thickness loss, and the affected surfaces were restored, corrective coatings maintenance performed, and moisture barrier material was installed at the concrete/containment vessel interface to seal this embedment zone from any future moisture intrusion. Additional action is planned to include this area in the inservice inspection plan for examination in accordance with IWE-2500, Table IWE-2500-1, Examination Category E-C, Item E4.11 during the next inspection period. The licensee found the condition of the containment vessel to be acceptable in this area. The licensee described another condition where the bottom portion of the Equipment Hatch Cover mating flange surfaces were observed to have areas with minor surface imperfections. These imperfections, including some minor scratches, dings, gouges, and residue have caused problems with sealing the equipment hatch. The affected surfaces were blasted, cleaned, and refaced to improve the sealing surface condition. The Equipment Hatch was subsequently retested with very low leakage, confirming the adequacy of the corrective action.

The NRC staff finds that, in the response to RAI 2, the licensee provided comprehensive comparative information of results of the general visual examination performed during 1EOC18 for the conditions observed during 1EOC16. The licensee also provided further detailed information of other significant findings during from the examinations performed during 1EOC18. The staff finds that the licensee's ISI program implementation adequately monitored locations where corrective actions were previously taken to prevent or correct further degradation at these locations and also identify and manage any new indications. Therefore, the response to RAI 2 is acceptable.

The licensee noted the testing frequency for penetrations using seals and gaskets to assure containment leak tight integrity is not affected by this requested extension to the Type A test interval from 15 years to approximately 15.5 years. The licensee stated that in addition, although McGuire is only required to test the bellows following an ILRT, a more conservative approach (test plan) has been implemented. McGuire has developed and is presently utilizing a supplementary testing program that tests one-third of the bellows each outage. Under this program all bellows have now been tested. In RAI 3, the staff requested the licensee to provide additional information about the above mentioned more conservative test plan for bellows. The licensee responded to RAI 3 as stated below:

> Currently, containment penetration bellows leak testing is required to be performed after an ILRT but no longer than 10 year test interval (TS 3.6.1).

> As a result of the test plan, between-the-plies leak testing of all containment penetration bellows was performed since the one immediately after the last ILRT for Unit 1 (May 1993). Additionally, as required, all penetration bellows found with detectable between-the-plies leakage (more than the minimum instrument error of 2 sccm at reduced pressure - 4 psig) were subjected to full pressure (15 psig) leak test from the containment direction during 1EOC16 (April 2004) to satisfy the 10 year surveillance.
>
> All bellows with detectable leakage are tested during each refueling outage by between-the-plies leak testing or were tested in 1EOC18 by internal pressurization using temporary boundaries to full containment pressure tests in cases where between-the plies leakage cannot be performed. These tests demonstrate that these bellows are not degrading.
>
> The results of the bellows leakage testing ensures that containment leakage is within acceptance criteria and provides confidence that no degradation is occurring that would challenge meeting 1O CFR 50 Appendix J leakage requirements.

The information provided describes the licensee's implementation of a conservative test plan for testing of bellows that ensure that the penetration bellows are not degrading. The NRC staff finds the licensee's response to RAI 3 acceptable.

Since inaccessible areas of the containment are an area of concern with regard to ensuring leak-tight integrity, in RAI 4, the NRC staff requested the licensee describe programs, if any, that are used to monitor the inaccessible, un-inspectable, or embedded areas of the containment, such as steel shell on the back of the ice baskets and to discuss the findings (if any) from these programs. The licensee provided the following response.

1. Surfaces of the containment vessel that are embedded in concrete are inaccessible for examination from either side, and Duke Energy Corporation {the Licensee] does not perform any visual, surface, or volumetric examinations on these surfaces. However, testing is performed on ground water samples to monitor the susceptibility of the embedded containment liner plate exterior surfaces to corrosion as described below:

   A groundwater sample collection penetration is installed in the floor of the incore instrumentation room to permit periodic collection and testing of groundwater directly beneath the containment embedded liner plate at this location. Samples are collected every 5 years and are analyzed to determine whether the groundwater is corrosive. During 1EOC18, a sample was collected from this penetration, was analyzed, and was found to meet established acceptance criteria.

2. During our Containment Inservice Inspection Interval 1 (September 9, 1998 through September 9, 2006), a number of surface areas were examined in accordance with the ASME Code, Section XI, Subsection IWE, IWE-2500, Table IWE-2500-1, Examination Category E-C (1992 Edition with the 1992 Addenda, as modified by approved relief requests). These locations and the basis for examining these areas was identified in our letter to the NRC, dated September 25, 2002. The results of augmented examinations performed during the first inspection interval revealed no unacceptable conditions or detectable wall thickness loss on any area examined. During the Containment Inservice Inspection Interval 2 (starting on July 15, 2005),

most of the areas that were subject to examination under Examination Category E-C during the first interval were no longer included in the Inservice Inspection plan because the requirement of IWE-2420(c) (1998 Edition through the 2000 Addenda) had been satisfied. The remaining areas subject to examination under Examination Category E-C are those that had not yet been examined during at least two consecutive inspection periods, as required by IWE-2420(c). All of the Examination Category E-C examinations scheduled for the current inspection interval were completed during refueling outage 1EOC18, and the examination results revealed no unacceptable conditions or detectable wall thickness loss on any item examined.

Containment vessel interior surfaces located behind the ice condensers are not examined in accordance with the ASME Code, Section XI, Subsection IWE, IWE 2500, Table IWE-2500-1, Examination Category E-C. The reason for this is as follows:

a. Operating experience at McGuire has shown that containment vessel surface areas at greater risk of degradation include those where cork expansion joint material has been installed between floor slabs of interior structures and the containment vessel shell. As a result of this operating experience, the cork expansion joint material was removed from between the ice condenser floor slabs and the containment vessel. As such, these surfaces are no longer deemed to be at greater risk that warrants examination in accordance with IWE-2500, Table IWE-2500-1, Examination Category E-C. It should be noted that the cork expansion joint material at these locations was removed prior to the start of the Containment Inservice Inspection Interval 1.

3. Portions of containment penetrations are not accessible for visual examination, and local leak-rate testing performed in accordance with 10 CFR 50, Appendix J is another program that is used to confirm the leak-tight integrity of these containment penetrations.

The licensee's response above to RAI 4 indicates that the licensee has considered and implemented measures that indirectly monitor or test degradation of the inaccessible, uninspectable, or embedded areas of the containment.

The discussion in the above paragraphs demonstrate that the licensee has adequate containment ISI programs in place that have been effectively implemented to examine, monitor and take appropriate corrective actions, as necessary, to ensure the leak-tight and structural integrity of the McGuire 1 containment. Based on the containment ISI program results, the previous ILRT results, and the SE for License Amendment No. 211, the NRC staff finds that the structural and leak-tight integrity of the McGuire 1 containment is sound and there is reasonable assurance that the containment integrity will be maintained at least until verified by the next Type A test proposed to be conducted during the 1EOC19 (fall 2008) refueling outage.

3.3 Risk Anaylsis

Based on the analyses provided by the licensee, the risk impacts and risk comparisons for the proposed change are essentially unchanged from those reported in License Amendment No. 211 and the staff conclusions remain valid. Specifically, the increase in the total integrated plant risk is small and supportive of the proposed change, the increase in the test interval results in only a small change in large early release frequency consistent with the acceptance guidelines of RG 1.174, and the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable

## 4.0 SUMMARY

The NRC staff finds that the structural and leak-tight integrity of the McGuire 1 containment is sound and there is reasonable assurance that the integrity will be maintained at least until verified by the next Type A test proposed to be conducted during the 1EOC19 (fall 2008) refueling outage. Therefore, granting a one-time 6-month extension to the current 15-year interval for performing the ILRT, as proposed by the licensee is acceptable.

## 5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission’s regulations in 10 CFR 50.92(c), "Issuance of amendment," state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the 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.

The NRC staff has reviewed the licensee’s analysis against the standards of 10 CFR 50.92(c). The licensee’s analysis is presented below.

> First Standard
>
> The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed extension to the Type A testing intervals cannot increase the probability of an accident previously evaluated since extension of the intervals is not a physical plant modification that could alter the probability of accident occurrence, nor is it an activity or modification by itself that could lead to equipment failure or accident initiation. The proposed extension to the Type A testing intervals does not result in a significant increase in the consequences of an accident as documented in NUREG-1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and Type C tests. It concludes that reducing the Type A testing frequency to once per twenty years leads to an imperceptible increase in risk.

McGuire provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. Prior Type A tests for McGuire Unit 1 identified containment leakage within acceptance criteria, indicating a very leak tight containment. Inspections required by the ASME Code are also performed in order to identify indications of containment degradation that could affect leak tightness. Separately, Type B and Type C testing, required by TS, identify any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the Type A test intervals will not represent a significant increase in the consequences of an accident.

Second Standard

The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revisions to the McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of fifteen years, based on past performance, would be extended on a one-time basis to approximately fifteen and a half years from the last Type A test. The proposed extension to the Type A test interval does not create the possibility of a new or different type of accident since there are no physical changes being made to the plants and there are no changes to the operation of the plants that could introduce a new failure mode.

Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. The proposed revisions to the McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of fifteen years, based on past performance, would be extended on a one-time basis to approximately fifteen and a half years from the last Type A test. The proposed extension to Type A test intervals will not significantly reduce the margin of safety. The NUREG-1493 generic study of the effects of extending containment leakage testing intervals found that a twenty-year interval resulted in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal effect on this risk, since 95 percent of the Type A detectable leakage paths would already be detected by Type B and Type C testing. Similar proposed changes have been previously reviewed and approved by the NRC, and they are applicable to McGuire.

The NRC staff has reviewed the licensee's analysis and, based on this review, has concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that the proposed amendment involves no significant hazards consideration.

## 6.0 STATE CONSULTATION

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.

## 7.0 ENVIRONMENTAL CONSIDERATION

The amendment relates to changes in recordkeeping, reporting, or administrative procedures or requirements. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 74357). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environment impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 8.0 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 Contributors: G. Thomas
H, Ashar
B. Lee
R. Palla

Date: February 13, 2008

McGuire Nuclear Station, Units 1 & 2

cc:

Vice President
McGuire Nuclear Station
Duke Power Company, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

Associate General Counsel and Managing Attorney
Duke Energy Carolinas, LLC
526 South Church Street - EC07H
Charlotte, North Carolina 28202

County Manager of Mecklenburg County
720 E. Fourth St.
Charlotte, NC 28202

Regulatory Compliance Manager
Duke Energy Corporation
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, NC 28078

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, NC 28078

Mecklenburg County
Department of Environmental Protection
700 N. Tryon St
Charlotte, NC 28202

Vice President
Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road, 12th Floor
Charlotte, NC 28210

NCEM REP Program Manager
4713 Mail Service Center
Raleigh, NC 27699-4713

Assistant Attorney General
NC Department of Justice
P.O. Box 629
Raleigh, NC 27602

Manager
Nuclear Regulatory Issues & Industry Affairs
Duke Energy Corporation
526 S. Church St.
Mail Stop EC05P
Charlotte, NC 28202

Division of Radiation Protection
NC Dept of Environment, Health & Natural Resources
3825 Barrett Dr.
Raleigh, NC 27609-7721

Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, SC 29745

Group Vice President, Nuclear Generation & Chief Nuclear Officer
P.O. Box 1006-EC07H
Charlotte, NC 28201-1006

Senior Counsel
Duke Energy Carolinas, LLC
526 South Church Street - EC07H
Charlotte, NC 28202