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

March 27, 2017

Mr. Steven D. Capps
Vice President
McGuire Nuclear Station
Duke Energy Carolinas, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING CHANGE TO TECHNICAL SPECIFICATION 3.6.14, "DIVIDER BARRIER INTEGRITY" (CAC NOS. MF8058 AND MF8059)

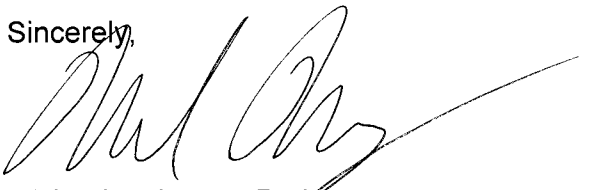
Dear Mr. Capps:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 294 to Renewed Facility Operating License No. NPF-9 and Amendment No. 273 to Renewed Facility Operating License No. NPF-17 for the McGuire Nuclear Station, Units 1 and 2, respectively. The amendments are in response to your application dated June 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16193A656), as supplemented by letter dated December 8, 2016 (ADAMS Accession No. ML16351A197).

The amendments modify Technical Specification 3.6.14, "Divider Barrier Integrity," to revise Conditions A and D to allow either one steam generator enclosure hatch or pressurizer enclosure hatch to be open (or inoperable) for a duration up to 48 hours.

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

Sincerely,



Michael Mahoney, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 294 to NPF-9
2. Amendment No. 273 to NPF-17
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 294
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 Duke Energy Carolinas, LLC (licensee), dated June 30, 2016, as supplemented by letter dated December 8, 2016, 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. 294, are hereby incorporated into this renewed operating 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: March 27, 2017

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-370

MCGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 273
Renewed 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), Renewed Facility Operating License No. NPF-17, filed by the Duke Energy Carolinas, LLC (the licensee), dated June 30, 2016, as supplemented by letter dated December 8, 2016, 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-17 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 273, are hereby incorporated into this renewed operating 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-17
and Technical Specifications

Date of Issuance: March 27, 2017

ATTACHMENT

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

LICENSE AMENDMENT NO. 294

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

LICENSE AMENDMENT NO. 273

RENEWED FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| NPF-9, page 3 | NPF-9, page 3 |
| NPF-17, page 3 | NPF-17, page 3 |

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| 3.6.14-1 | 3.6.14-1 |

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts, 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as my be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and,

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such by product material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or thereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3469 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 294 are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than March 3, 2023, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59, and otherwise complies with the requirements in that section.

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts, 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as my be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and,

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such by product material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or thereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3469 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 273 are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than March 3, 2023, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59, and otherwise complies with the requirements in that section.

3.6 CONTAINMENT SYSTEMS

3.6.14 Divider Barrier Integrity

LCO 3.6.14 Divider barrier integrity shall be maintained.

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

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. ----------NOTE---------- For this action, separate Condition entry is allowed for each personnel access door or equipment hatch. ---------- One or more personnel access doors or equipment hatches (other than one pressurizer or one steam generator enclosure hatch addressed by Condition D) open or inoperable, other than for personnel transit entry. | A.1 Restore personnel access doors and equipment hatches to OPERABLE status and closed positions. | 1 hour |
| B. Divider barrier seal inoperable. | B.1 Restore seal to OPERABLE status. | 1 hour |
| C. Required Action and associated Completion Time not met. | C.1 Be in MODE 3.<br>AND<br>C.2 Be in MODE 5. | 6 hours<br>36 hours |
| D. One pressurizer or one steam generator enclosure hatch open or inoperable. | D.1 Restore affected hatch to OPERABLE status and closed position. | 48 hours |

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO

# AMENDMENT NO. 294 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-9

# AND

# AMENDMENT NO. 273 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-17

# DUKE ENERGY CAROLINAS, LLC

# MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

# DOCKET NOS. 50-369 AND 50-370

## 1.0 INTRODUCTION

By letter to the United States Nuclear Regulatory Commision (NRC, the Commission) dated June 30, 2016 as supplemented by letter dated December 8, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16193A656 and ML16351A197, respectively), Duke Energy Carolinas, LLC (Duke Energy) submitted an application requesting a change to the Technical Specifications (TSs) for McGuire Nuclear Station (MNS), Units 1 and 2. The supplement dated December 8, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 3, 2017 (82 FRN 158).

The proposed License Amendment Request (LAR) would revise TS 3.6.14, "Divider Barrier Integrity," Conditions A and D to allow one steam generator (SG) enclosure hatch or pressurizer enclosure hatch to be open for a duration up to 48 hours (from 6 hours). The proposed LAR would modify Required Action D.1 to delete reference to the "pressurize enclosure" and generically apply action to restore "affected" hatch to operable status and closed position.

The licensee also requested to change the Completion Time of Condition D to allow either one SG enclosure hatch or pressurizer enclosure hatch to be open (or inoperable) for a duration up to 48 hours. The request is made to facilitate potential inspections and maintenance and to enhance personnel and radiation safety.

## 2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR) Section 50.36, "Technical specifications" states that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions of operation (LCOs); (3) SRs [surveillance requirements]; (4) design features; and (5) administrative controls. Paragraph 10 CFR 50.36(c)(2)(i) states that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility and that when a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

The NRC staff's acceptance criteria is based on the following General Design Criterias (GDC) in Chapter 10 of the *Federal Code of Regulations* (CFR), Section 50, Appendix A:

- "Criterion 16 – Containment design," as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded.

- "Criterion 38 – Containment heat removal," as it relates to the Containment Heat Removal System (CHRS) function to rapidly reduce the containment pressure and temperature following any loss-of-coolant accident (LOCA) and maintain them at acceptably low levels.

- "Criterion 50 – Containment design basis," as it relates to the reactor containment structure and associated Heat Removal System(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions.

## 3.0 TECHNICAL EVALUATION

MNS, Units 1 and 2, are Westinghouse 4-loop designs with ice condenser containments.

The ice condenser containment consists of three parts; the upper compartment, the lower compartment, and the ice condenser compartment. A high-energy line break (HELB) inside containment, such as a LOCA, would occur in the lower compartment. The resulting high pressure would be relieved into the ice condenser compartment and then into the upper compartment. Air and steam is transferred through the ice condenser compartment, and then enters the upper compartment. The ice acts to keep the peak containment pressure below design pressure during an accident, such as a LOCA.

If a pressurizer enclosure hatch or SG enclosure hatch is open during a HELB inside containment, steam would bypass the ice condenser compartment and enter directly to the upper compartment from the lower compartment. This steam bypass could lead to a higher containment pressure compared to the pressure developed with pressurizer and SG enclosure hatches closed.

### 3.1 Licensee's Proposed Technical Specification Changes

Currently TS 3.6.14, Condition D allows for one pressurizer enclosure hatch to be open or inoperable for an allowed outage time of six hours. The six-hour duration in Condition D is an

increase from the original allowed outage time of one hour, which was approved in a license amendment that was issued by the NRC on April 5, 2005 (ADAMS Accession No. ML050340223).

Current TS LCO 3.6.14, CONDITION A states:

> A. One or more personnel access doors or equipment hatches (other than one pressurizer enclosure hatch addresses by Condition D) open or inoperable, other than for personnel transit entry.

Revised TS LCO 3.6.14, CONDITION A would state:

> A. One or more personnel access doors or equipment hatches (other than one pressurizer or one steam generator enclosure hatch addressed by Condition D) open or inoperable, other than for personnel transit entry.

Current TS LCO 3.6.14, CONDITION D, REQUIRED ACTION D.1, AND COMPLETION TIME states:

| | | | |
|---|---|---|---|
| D. One pressurizer enclosure hatch open or inoperable. | D.1 | Restore pressurizer enclosure hatch to OPERABLE status and closed position | 6 hours |

Revised TS LCO 3.6.14, CONDITION D, REQUIRED ACTION D.1, AND COMPLETION TIME would state:

| | | | |
|---|---|---|---|
| D. One pressurizer or one steam generator enclosure hatch open or inoperable. | D.1 | Restore affected hatch to OPERABLE status and closed position | 48 hours |

The licensee stated that the changes are requested to faciltitate unplanned entries into a pressurizer or a SG enclosure. The licensee stated some reasons for unplanned entries are as follows:

- Suspected instrument tubing leak affecting pressurizer level indication
- Suspected leakage of pressurizer safety valve(s) or Power-Operated Relief Valve (PORV)
- Malfunctioning hydrogen igniters
- PORV block valve indication.

## 3.2 NRC Staff Evaluation

### 3.2.1 Pressurizer Enclosure Hatch

There is one pressurizer enclosure hatch plug in MNS, Unit 1, and three pressurizer enclosure hatch plugs in MNS, Unit 2. These pressurizer enclosure hatches have been evaluated

previously in accordance with NRC's NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," for lifting with the polar crane located inside upper containment for removal of the hatch plugs. In its letter dated June 30, 2016, the licensee stated that there is no loss of required safe shutdown function due to a postulated drop of the largest pressurizer enclosure hatch plug.

3.2.2 SG Enclosure Hatch

The SG enclosure hatch cover is a 30-inch nominal pipe size blind flange. The hatch is used for personnel access into a SG cavity area for repairs or troubleshooting activities, when necessary. A device called a davit arm is used to secure the SG enclosure hatch when removed and the hatch is less than 300 pounds and not considered a heavy load, per NUREG-0612. In its letter dated June 30, 2016, the licensee stated there is no safety-related equipment directly under the SG enclosure hatch and in the event the hatch fell, no damage is expected.

3.2.3 Containment Analysis

The MNS, Units 1 and 2, Updated Final Safety Analysis Report (UFSAR), Section 6.2.1.1.3.1, presents results from analyzed effects of divider barrier deck leakages for bypass areas up to 50 square feet ($ft^2$). The assumed bypass leakage area prior to the LAR was 5 $ft^2$ and the request incorporates an additional 7.5 $ft^2$ bypass area. The additional 7.5 $ft^2$ bypass area conservatively bounds the opening of either a pressurizer enclosure hatch or a SG enclosure hatch.

The NRC staff issued a request for additional information (RAI) on October 19, 2016 (ADAMS Accession No. ML16293A901) to determine the impact on the long term peak containment pressure due to the additional 7.5 $ft^2$ bypass area. The licensee's response to the RAI dated December 8, 2016 (ADAMS Accession No. ML16351A197) included the containment pressure and sump temperature results of a RELAP5 computer code for Mass and Energy (M&E) release analysis and Generation of Thermal Hydraulic Information for Containments (GOTHIC) code for containment response analysis with the additional bypass area. The NRC staff has approved the licensee's RELAP5/GOTHIC methodology in a letter dated February 29, 2000 (ADAMS Accession No. ML003781034, not publically available).

The transient evaluated was the limiting large-break LOCA with the break located in the cold-leg discharge piping with minimum Emergency Core Cooling System (ECCS) flows. The results showed that the containment pressure response Analysis-Of-Record (AOR) is bounding. The sump temperature profile showed an insignificant difference from the AOR for the post-recirculation sump temperature and, therefore, would have no impact on the Net Positive Suction Head (NPSH) analysis of the pumps that draw water from the sump in the recirculation mode.

The long term peak containment pressure is 13.87 pounds per square inch gage (psig) in the AOR (UFSAR Section 6.2.1.1.3) and 13.45 psig for the containment response analysis with the additional 7.5 $ft^2$ bypass area. According to USFSAR Section 6.2.1.1.2, the containment design pressure is 15 psig. For the first five to six minutes of the transient, the peak containment pressure is higher in the upper containment with the additional bypass area because there is more steam in the upper containment at this time than in the AOR, however, the peak containment pressure occurs later. As the transient progresses, ice melts in the ice condenser and steam is added to the upper containment. The licensee stated, in part, in their December 8,

2016 letter, "After about one hour, the impact of the additional divider deck bypass area has diminished to the point where the containment pressures and upper containment temperatures are virtually the same." The long-term peak containment pressure occurs in the transient after the one-hour mark, which is bounded by the AOR.

Additionally, the NRC requested information regarding any changes that may occur to the containment spray initiation time due to the increased bypass flow. In the licencees December 8, 2016 response, they explained that there were no changes to the containment spray initiation time. The licensee stated that they initiate one train of containment spray manually after the refueling water storage tank low level has been reached for LOCA scenarios. The licensee stated that the initiation time is not impacted because the suction source for the spray flow is the containment sump and the draindown rate of the refueling water storage tank is not impacted by a change in bypass area.

In UFSAR Section 6.2.3.1.1, it is stated that the M&E release methodology is the Westinghouse Commercial Atomic Power (WCAP)-10325-P-A, "Mass and Energy Release Model for Containment Design" dated April 1979. The SATAN computer code in this methodology is used for the blowdown M&E release analysis during a LOCA which is an input to determine the subcompartment pressure and differential pressure responses during the blowdown phase using the Transient Mass Distribution (TMD) code. Westinghouse has issued Nuclear Safety Advisory Letters (NSALs)-06-6, -11-5, and -14-2, and InfoGram (IG)-14-1 reporting errors in the WCAP-10325-P-A methodology.

The NRC staff inquired whether corrections for the M&E analysis should be made due to the results of NSALs -06-6, -11-5, and -14-2 and IG-14-1. The licensee's December 8, 2016 letter, stated, in part:

> The results from the SATAN code are used in the analyses of record (UFSAR Section 6.2.1.3.1) to calculate the short-term (or blowdown) mass and energy release during a LOCA. These mass and energy release calculations are then used as input to determine the subcompartment pressures and differential pressure responses using the TMD code, as discussed in UFSAR Section 6.2.1.2.3. The TMD results presented in the McGuire UFSAR (Tables 6-2, 6-3, 6-44 and 6-45) show that the peak subcompartment pressures and differential pressures are reached between 0.009 and 3.0 seconds after the postulated break. Therefore, any issues identified with the SATAN-V code would not impact the mass and energy results within the first few seconds of the transient and would therefore have no impact on the subcompartment pressure and differential pressure results. The subcompartment pressure and differential pressure results are largely driven by the forced movement of the air/steam mixture after the postulated LOCA in the lower containment volume and/or the subcompartments being analyzed.
>
> The issues raised by Westinghouse NSALs -06-6, -11-5, and -14-2 as well as InfoGram IG 14-1 can all be characterized as having no impact on the first few seconds of the analyzed transients. These issues include the volumetric heat capacities of stainless steel, the flow values, purge volumes, and [Auxiliary Feedwater] AFW flows during the reflood and post-reflood periods, reactor vessel metal mass, secondary side pressures, and amounts of stored energy within the Reactor Coolant System metal mass. Since all of these factors either involve conduction of energy into the Reactor Coolant System, or are associated with

periods of the transient much further out than the first few seconds, it is concluded that there is no impact on the TMD results of transients where the peak subcompartment or differential pressure is reached within the first 3 seconds.

3.3 NRC Staff Conclusion

The NRC staff concludes that the proposed change meets the requirements of 10 CFR Part 50 Appendix A, "Criterion 16 - Containment design," as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded.

The NRC staff concludes that the proposed change meets the requirements of 10 CFR Part 50 Appendix A, "Criterion 38, Containment heat removal," as it relates to the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

The NRC staff also concludes that the proposed change meets the requirements of 10 CFR Part 50 Appendix A, "Criterion 50, Containment design basis," as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions.

The NRC staff concludes that with the proposed changes in the TS, containment integrity is maintained because the peak containment pressure during a LOCA remains below the containment design pressure and the CHRS will perform its required safety function because the proposed change does not affect NPSH of the pumps that draw water from the sump during the recirculation phase of containment cooling.

Therefore, the NRC staff concludes that the revised TS 3.6.14, Conditions A and D, and Required Action D.1 and associated extended Completion Time to 48 hours will continue to maintain reasonable assurance that the containment peak pressure and the long-term containment peak pressure remain acceptable. The NRC staff concludes that the increased time a single SG enclosure hatch or pressurizer enclosure hatch is open (or inoperable) and the revised TSs continue to meet the requirements of 10 CFR 50.36(c)(2)(i) and is, thefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments on March 3, 2017. The NRC confirmed on March 6, 2017, that the State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. 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 this finding (82 FRN 158: January 3, 2017). 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.

## 6.0 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) there is reasonable assurance that 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: D. Woodyatt, NRR

Date: March 27, 2017.

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING CHANGE TO TECHNICAL SPECIFICATION 3.6.14, "DIVIDER BARRIER INTEGRITY" (CAC NOS. MF8058 AND MF8059)

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**ADAMS Accession No.: AMD: ML17060A481; BWI: ML17060A492** *by memorandum

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| DATE | 3/8/17 | 3/8/17 | 1/27/17 | 3/9/17 |
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