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U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-369 AND 50-370 / RENEWED LICENSE NOS. NPF-9 AND NPF-17

SUBJECT: LICENSE AMENDMENT REQUEST (LAR) FOR ONE-TIME EXTENSION OF APPENDIX J TYPE A INTEGRATED LEAKAGE RATE TEST INTERVAL

In accordance with the provisions of 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) requests amendments, in the form of changes to the Technical Specifications (TS) to Renewed Facility Operating Licenses NPF-9 and NPF-17 for McGuire Nuclear Station (MNS) Units 1 and 2. The proposed LAR would allow for a one-time extension to the 10-year frequency of the MNS Units 1 and 2 containment leakage rate test (i.e., Integrated Leakage Rate Test (ILRT) or Type A test). This test is required by TS 5.5.2 "Containment Leakage Rate Testing Program." The proposed change would permit the existing ILRT frequency to be extended from 10 years to 10.5 years.

The proposed change would, based on current refueling outage (RFO) projected schedules, allow Duke Energy to minimize the impact of the ILRT on critical path outage activities by not having to perform the MNS Unit 1 ILRT prior to the expiration of the MNS Unit 1 10-year interval. Currently, the ILRT is to be performed approximately one year prior to the 10th year anniversary of the completion of the last ILRT (October 21, 2008). If granted, this revision would extend the period from 10 years to 10.5 years between successive tests. In terms of RFOs, this extension would move the performance of the next ILRT from the scheduled fall 2017 End of Cycle 25 RFO to the spring 2019 End of Cycle 26 RFO.

The proposed change would provide the same benefits for MNS Unit 2 outage activities. Currently, the MNS Unit 2 ILRT is to be performed approximately one year prior to the 10th year anniversary of the completion of the last Type A test (March 31, 2008). If granted, this revision would extend the period from 10 years to 10.5 years between successive tests. In terms of RFOs, this extension would move the performance of the next ILRT from the scheduled spring 2017 End of Cycle 24 RFO to the fall 2018 End of Cycle 25 RFO.

The proposed change would also correct a couple of typographical and administrative errors introduced into TS 5.5.2 by the implementation of two previous TS amendments.

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Enclosure 1 provides an evaluation of the proposed change. The marked-up Technical Specification, reflecting the proposed change in this submittal, is included in Attachment 1. Retyped (clean) TS pages are included in Attachment 2.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed change involves no significant hazards consideration. The bases for this determination are included in Enclosure 1.

This letter contains no new Regulatory Commitments and no revision to existing Regulatory Commitments.

In accordance with Duke Energy administrative procedures that implement the Quality Assurance Program Topical Report, this proposed change has been reviewed and approved by the Plant Operations Review Committee. A copy of this LAR is being sent to the State of North Carolina in accordance with 10 CFR 50.91 requirements.

Duke Energy requests approval of this LAR by December 19, 2016. Once approved, the amendment will be implemented within 60 days.

Should you have any questions concerning this letter or require additional information, please contact P. T. Vu of MNS Regulatory Affairs, at 980-875-4302.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 18, 2016.

Sincerely,

SD Capps

Steven D. Capps

Enclosure

1. Evaluation of the Proposed Change

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Evaluation of the Proposed Change

Subject: License Amendment Request - Revise Technical Specification Section 5.5.2 for One-Time Extension of Appendix J Type A Integrated Leakage Rate Test Interval

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Attachments:
1. Technical Specification Pages (Mark-up)
2. Technical Specification Pages (Retyped)

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Duke Energy requests an amendment to the McGuire Nuclear Station (MNS) Unit 1, Renewed Facility Operating License NPF-9, and Unit 2, Renewed Facility Operating License NPF-17, by incorporating the attached proposed change into the Unit 1 and Unit 2 Technical Specifications (TS). Specifically, the proposed change is a request to revise TS 5.5.2 "Containment Leakage Rate Testing Program" to allow one-time extension to the 10-year frequency of the MNS Unit 1 and Unit 2 containment leakage rate test (i.e., Integrated Leak Rate Test (ILRT) or Type A test). The proposed change would permit the existing ILRT frequency to be extended one time from 10 years to 10.5 years.

The proposed change would, based on current refueling outage (RFO) projected schedules, allow Duke Energy to minimize the impact of the ILRT on critical path outage activities by not having to perform the MNS Unit 1 ILRT prior to the expiration of the MNS Unit 1 10-year interval. Currently, the ILRT is to be performed approximately one year prior to the 10th year anniversary of the completion of the last ILRT (October 21, 2008). If granted, this revision would extend the period from 10 years to 10.5 years between successive tests. In terms of RFOs, this extension would move the performance of the next ILRT from the scheduled fall 2017 End of Cycle 25 RFO to the spring 2019 End of Cycle 26 RFO.

The proposed change would also provide the same benefits for MNS Unit 2 outage activities. Currently, the MNS Unit 2 ILRT is to be performed approximately one year prior to the 10th year anniversary of the completion of the last Type A test (March 31, 2008). If granted, this revision would extend the period from 10 years to 10.5 years between successive tests. In terms of RFOs, this extension would move the performance of the next ILRT from the scheduled spring 2017 End of Cycle 24 RFO to the fall 2018 End of Cycle 25 RFO.

The proposed change would also correct a couple of typographical and administrative errors introduced into TS 5.5.2 by the implementation of two previous TS amendments.

2.0 DETAILED DESCRIPTION

2.1 Current Containment Leakage Rate Testing Program

Current TS 5.5.2 specifies, "A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 27, 1993 (Unit 1) and August 20, 1993 (Unit 2) Type A test shall be performed no later than plant restart after the End Of Cycle 19 Refueling Outage (Unit 1) and August 19, 2008 (Unit 2); and

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

2.2 TS Change Description

The proposed change (in bold) will revise TS 5.5.2 to state, in part:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
>
> **a.** NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the **October 21, 2008** (Unit 1) and **March 31, 2008** (Unit 2) Type A test shall be performed no later than plant restart after the End of **Cycle 26** Refueling Outage (Unit 1) and **no later than plant restart after the End of Cycle 25 Refueling Outage** (Unit 2); and
>
> **b.** The containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR **3.6.1.1** Type A test, prior to initiating the Type A test.

The revised dates of October 21, 2008 and March 31, 2008 are the dates of last ILRTs completed for Unit 1 and Unit 2, respectively. The revised end of cycle numbers, Cycle 26 and Cycle 25, are the refueling outages when the next ILRTs will be performed for Unit 1 and Unit 2, respectively.

The proposed change would also correct a couple of typographical and administrative errors:

- NRC letter dated May 8, 2003 for TS Amendments 212/193 (Reference 14) had the correct designations, "a." and "b." for the two exceptions above. These amendments introduced the typographical error (SR 3.6.11) into the second exception above. It should have been SR 3.6.1.1. The licensee provided the correct SR number in the TS markup but wrong SR number in the retyped TS page to the NRC.

- NRC letter dated February 13, 2008 for TS Amendment 244 (Reference 25) introduced an administrative error into TS 5.5.2. This amendment inadvertently omitted the designation "a." for the first exception above. Without the "a." designation for the first exception, the designation for the second exception above was automatically changed by the word processing software to "a." during retyping of the TS page. The licensee provided the markup and retyped TS pages without the "a." designation to the NRC.

A markup of TS 5.5.2 is provided in Attachment 1. The retyped TS pages are provided in Attachment 2.

3.0 TECHNICAL EVALUATION

3.1 Justification for the Technical Specification Change

The requested corrections for the typographical and administrative errors are intended to restore them to the condition intended by their original amendment applications.

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not

exceed the allowable leakage values specified in the TS. 10 CFR 50, Appendix J, also ensures that periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating primary containment. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident (DBA). Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations; and (3) Type C tests, intended to measure containment isolation valve leakage rates. Types B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Types B and C testing.

In 1995, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based" in 10 CFR 50, Appendix J refers to both the performance history necessary to extend test intervals as well as to the criteria necessary to meet the requirements of Option B.

Also in 1995, RG 1.163 was issued. The RG endorsed NEI 94-01, Revision 0, (Reference 6) with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency for the containment Type A (ILRT) test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 7) and Electric Power Research Institute (EPRI) TR-104285 (Reference 8), both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small. In addition to the 10-year ILRT interval, provisions for extending the test interval an additional 15 months were considered in the establishment of the intervals allowed by RG 1.163 and NEI 94-01, but this "should be used only in cases where refueling schedules have been changed to accommodate other factors."

In 2008, NEI 94-01, Revision 2-A, (Reference 3) was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, subject to the limitations and conditions noted in Section 4.0 of the NRC Safety Evaluation Report (SER) on NEI 94-01. The NRC SER was included in the front matter of this NEI report. NEI 94-01, Revision 2-A includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163 (September 1995). It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk insights.

3.1.1 Current 10 CFR 50, Appendix J Requirements

Title 10 CFR Part 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance Based Requirements." MNS has implemented the requirements of 10 CFR Part 50, Appendix J, Option B for Types A, B and C testing.

RG 1.163, Section C.1, states that licensees intending to comply with 10 CFR Part 50, Appendix J, Option B, should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 Revision 0 (Reference 6) rather than using test intervals specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-1994. NEI 94-01 Revision 0, Section 11.0, refers to Section 9, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per ten years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage was less than 1.0 L_a (where L_a is the maximum allowable leakage rate at design pressure). Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Adoption of the Option B performance-based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Type A tests but did not alter the basic method by which Appendix J leakage testing is performed. The test frequency is based on an evaluation of the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. The allowed frequency for Type A testing as documented in NEI 94-01 Revision 0, is based, in part, upon a generic evaluation documented in NUREG-1493 (Reference 7). The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak tightness for differing types of containment types. NUREG-1493 concluded in Section 10.1.2 that reducing the frequency of Type A tests (ILRT) from the original three tests per ten years to one test per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Types B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements. Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, NUREG-1493 concluded that increasing the interval between ILRTs is possible with minimal impact on public risk.

NEI 94-01, Revision 0, Section 9.1, states the following concerning the extension of Type A test intervals:

> Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing given in this section may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

NEI 94-01, Revision 0, Section 9.2.2, states the following concerning Type A test intervals:

> If the test interval ends while primary containment integrity is either not required or it is required solely for shutdown activities, the test interval may be extended indefinitely. However, a successful Type A test shall be completed prior to entering the operating mode requiring primary containment integrity.

3.1.2 10 CFR 50, Appendix J, Option B Licensing History

SER dated September 4, 2002 – ML022540102 (Reference 12)

The Commission issued Amendment No. 207 to Facility Operating License No. NPF-9 and Amendment No. 188 to Facility Operating License NPF-17 for MNS, Units 1 and 2.

The amendments revised the TS to permit implementation of containment local leakage rate testing addressed by Title 10 of the Code of Federal Regulations, Part 50, Appendix J, Option B, and to reference Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B. In addition, the TS were revised regarding soap bubble testing and leak testing of containment purge valves with resilient seals for upper and lower compartments and instrument rooms.

SER dated March 12, 2003 - ML030760032 (Reference 13)

The Commission issued Amendment No. 211 to Facility Operating License No. NPF-9 and Amendment No. 192 to Facility Operating License NPF-17 for MNS, Units 1 and 2.

The amendments revised the TSs to allow a one-time change in the Appendix J, Type A containment ILRT interval from the currently required 10-year interval to a test interval of 15 years.

SER dated May 8, 2003 – ML031280431 (Reference 14)

The Commission issued Amendment No. 212 to Facility Operating License NPF-9 and Amendment No. 193 to Facility Operating License NPF-17 for MNS, Units 1 and 2.

The amendments revised the TS to (1) modify the Surveillance Requirement to be consistent with the design of the reactor building access openings, (2) modify the frequency of the Surveillance Requirement for visual inspections for the exposed interior and exterior surfaces of the reactor building, and (3) modify the administrative controls for the containment leakage rate testing program.

SER dated February 13, 2008 - ML073400670 (Reference 25)

The Commission issued Amendment No. 244 to Facility Operating License NPF-9 for MNS, Unit 1.

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 previous frequency).

3.1.3 Containment Building Description

Description of the Containment

The Containment consists of a Containment Vessel and a separate Reactor Building enclosing an annulus. The Containment Vessel is a freestanding welded steel structure with a vertical cylinder, hemispherical dome, and a flat circular base.

The containment vessel is a freestanding welded steel structure with a vertical cylinder, hemispherical dome and a flat base. The Containment shell is anchored to the Reactor Building foundation by means of anchor bolts around the circumference of the cylinder base. The base of the Containment is 1/4 inch liner plate encased in concrete and anchored to the Reactor Building foundation. The base liner plate functions only as a leak-tight membrane and is not designed for structural capabilities. The Containment Vessel has a nominal inside diameter of 115 feet, overall height of 171 feet 3 inches, nominal wall thickness of 0.75 inch, nominal dome thickness of 0.6875 inch, nominal bottom thickness of 0.25 inch, and net free volume of 1.2×10^6 cubic feet.

The ice condenser region is an insulated cold storage area contained within the annulus formed by the containment wall and the crane wall circumferentially over a 300-degree arc. This area is 74 feet high and is maintained at 15 to 27 °F by the glycol refrigeration system. Around 2 million pounds of borated ice is contained in an array of 1944 steel baskets (81 baskets per bay, 24 bays) resting on a lower support structure and positioned laterally by horizontal lattice frames installed at various elevations.

The Reactor Building is a reinforced concrete structure composed of a right cylinder with a shallow dome roof and flat circular foundation slab. The cylinder has an inside radius of 63 feet 6 inches and a wall thickness of 3 feet. The dome has an inside spherical radius of 87 feet and is 2 feet 3 inches thick. The foundation slab is 137 feet in diameter and 6 feet thick.

The Reactor Building houses the steel containment vessel and is designed to provide environmental as well as missile protection for the steel shell.

A five-foot annular space is provided between the steel containment vessel and the Reactor Building for control of the containment external temperatures and pressures. The annular space also provides a controlled air volume for filtering and provides access to penetrations for testing and inspection. Following a loss-of-coolant accident (LOCA), the annular space is kept at a slightly negative pressure to control and filter radioactive leakage, if any, from the containment vessel and penetrations.

Containment Penetrations

Several penetrations are required through the containment vessel for personnel and equipment access, fuel transfer and various piping systems. The containment penetrations are:

Equipment Hatch

The equipment hatch is composed of a 20-foot cylindrical sleeve in the containment shell and a dished head with mating, bolted flanges. The flanged joint has double compressible seals with an annular space for pressurization and testing.

Personnel Locks
Two personnel locks are provided for each unit. Each lock has double doors with an interlocking system to prevent both doors being opened simultaneously. Remote indication is provided to indicate the position of each door.

Double, inflatable seals are provided on each door. A top connection between the seals provides the capability for local leak rate testing as required. The use of double inflatable seals allows testing of the annulus space without the use of external strongbacks or other remote devices.

Fuel Transfer Penetration
A 20-inch fuel transfer penetration is provided for transfer of fuel to and from the fuel pool and the Containment fuel transfer canal. The fuel transfer penetration is provided with a double gasketed blind flange in the transfer canal and a gate valve in the fuel pool. Expansion bellows are provided to accommodate differential movement between the connecting buildings.

Spare Penetrations
Spare penetrations are provided to accommodate future piping and electrical penetrations. The spare penetrations consist of the penetration sleeve and head.

Penetration Sleeves
All penetration sleeves are preassembled and welded into containment vessel shell plates. Each shell plate having penetration sleeves is stress relieved prior to installation into the containment.

Purge Penetrations
The purge penetrations each have one interior and one exterior quick-acting tight-sealing isolation valves. During normal Plant operation, Modes 1-4, the containment purge and exhaust isolation valves are failed closed. In Modes 5, 6, and No Mode, these isolation valves will automatically actuate closed on containment high radiation signal.

Electrical Penetrations
Medium voltage electrical penetrations for reactor coolant pump power use sealed bushings for conductor seals. The assemblies incorporate dual seals along the axis of each conductor.

Low voltage power, control and instrumentation cables enter the containment vessel through penetration assemblies, which have been designed to provide two leak tight barriers in series with each conductor.

All electrical penetrations have been designed to maintain containment integrity for Design Basis Accident conditions including pressure, temperature and radiation. Double barriers permit testing of each assembly as required to verify that containment integrity is maintained.

Mechanical Penetrations
Mechanical penetrations are treated as fabricated piping assemblies meeting the requirements of ASME Section III, Subsections NC and which are assigned the same classification as the piping system that includes the assembly.

The process line and flued heads making up the pressure boundary are consistent with the system piping materials; fabrication, inspection and analysis requirements are as required by ASME Section III, Section NC.

Critical high temperature lines and selected engineered safety system and auxiliary lines (regardless of temperature) require the "Hot Penetration" assembly which features the exterior guard pipe for the purpose of returning any fluid leakage to the Containment and for protection of other penetrations in the building annular space. Other lines are treated as cold penetrations since a leak into the annular space would not cause a personnel hazard or damage other penetrations in the immediate area.

Overpressure Protection
Overpressure protection shall be provided where the potential exists to over pressurize containment penetration piping due to thermal expansion of the fluid trapped in the penetration piping. In other words, overpressure protection shall be provided to relieve the pressure buildup caused by the heatup of a trapped volume of incompressible fluid between two positively closing valves (due to containment temperature transient) back into containment where an open relief path exists. This open relief path could be the relief valve on any normally aligned component, or to the Reactor Coolant System (NC) itself.

3.1.4 <u>Plant-Specific Confirmatory Analysis</u>

The purpose of this analysis is to provide risk insights about extending the currently allowed containment Type A Integrated Leak Rate Test (ILRT) interval by 6 months for MNS Units 1 and 2. The extended test interval is a one-time 6-month increase over the currently approved 10-year test interval. This translates to an extended test interval of 10.5 years. The risk assessment followed the guidelines from the following:

- NEI 94-01, Revision 3-A,
- the methodology used in EPRI TR-104285 (Reference 8),
- the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 17),
- risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 4),
- the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval (Reference 18), and
- the methodology used in EPRI 1018243 (Revision 2-A of EPRI 1009325) (Reference 16).

The findings of the MNS risk assessment confirm the general findings of previous studies that the risk impact associated with extending the ILRT interval from 10 years to 10.5 years is "small." The MNS plant-specific results for extending ILRT interval from the current 10 years to 10.5 years are summarized below:

- Since the ILRT does not impact Core Damage Frequency (CDF), the relevant criterion is Large Early Release Frequency (LERF). The increase in LERF resulting from a change in the Type A ILRT test interval from three in 10 years to one in 10.5 years is very conservatively estimated to be "small."
- An additional assessment of the impact from external events was also performed. In this sensitivity case, the change in the total MNS LERF (including external events) was conservatively estimated to be "small."

- The change in Type A test frequency to one per 10.5 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.023 person-rem/year. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of < 1.0 person-rem per year, or < 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. Moreover, the risk impact when compared to other severe accident risks is "negligible."
- The increase in the conditional containment failure from the three in 10-year interval to one in 10.5-year interval is approximately 0.5%. EPRI Report No. 1009325, Revision 2-A, states that increases in CCFP of < 1.5 percentage points is very small. Therefore, this increase is judged to be "very small."

Therefore, increasing the ILRT interval to 10.5 years is considered to be insignificant since it represents a "small" change to the MNS risk profile.

The NRC, in NUREG-1493, has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B or Type C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test integrity of the containment structure.

The findings for MNS confirm these general findings on a plant-specific basis considering the severe accidents evaluated for MNS, the MNS containment failure modes, and the local population surrounding MNS.

The insights from this risk analysis support the deterministic analysis showing that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner of this license request.

3.2 Inspections

3.2.1 Primary Containment Coatings

Duke Energy complies with RG 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants.

The original coating materials and coating systems were specified by Engineering and applied by the Duke Power Construction Department to all structures within the containment and the Containment Vessel. The coating systems were qualified for radiation exposure, pressure, temperature, and water chemistry exposure during a DBA in accordance with ANSI N101.2.

Carboline coating materials are now used for maintenance of the existing coating systems and for any new applications. These coating systems are specified by Engineering and applied by the Duke Energy Maintenance Department. The Carboline coating materials have been qualified over the existing Mobil/Valspar coatings as a mixed system and as a new coating system for radiation exposure, pressure, temperature, and water chemistry exposure during a DBA in accordance with ANSI N101.2.

The elements of the McGuire Coatings Program are documented in Nuclear Operating Fleet Administrative Procedures. The McGuire Coatings Program includes periodic condition assessments of Service Level I coatings used inside containment. As localized areas of degraded coatings are identified, those areas are evaluated for repair or replacement, as necessary.

A maximum of 20,000 square feet of unqualified coatings inside Containment is considered to be a negligible fraction of the Containment interior surfaces.

Coatings inside the Containment are classified as either qualified or unqualified for the purposes of debris generation analysis, which encompasses all of the coating systems used (i.e., epoxies, alkyd enamels, and cold galvanizing products). Unqualified coatings in Containment are assumed to fail as particulates, in accordance with NEI 04-07 guidance. The total weight of unqualified coatings assumed to fail inside Containment is approximately 372.8 pounds, all of which is assumed to transport to the strainer. Qualified coatings are assumed to fail as particulates within a 5 D zone of influence (ZOI) based on the methodology outlined in WCAP-16568-P. The total weight of qualified coatings assumed to fail inside Containment is approximately 167.6 pounds, all of which is assumed to transport to the strainer.

3.2.2 Inservice Inspection Program for Containment – IWE

In accordance with the requirements of Paragraph 50.55a(g), and as modified and supplemented by paragraph 50.55a(b)(2)(ix) of 10 CFR Part 50, the Inservice inspection of Class MC metal containments at Units 1 and 2 of the MNS will be performed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, 2007 Edition with the 2008 Addenda (hereafter referred to as Section XI). The examinations will be performed to the extent practicable within the limitations of design, geometry and materials of construction of the component. Examinations were scheduled for the Third Inspection Interval in accordance with ASME Section XI IWE-2411.

Component/System Boundaries Subject to Inspection and Examination

The boundaries of Class MC non-exempt components and their supports are shown on drawings listed in Section 2 of the Unit Specific Containment ISI Plan.

> Note: Boundaries of piping penetration subassemblies (bellows) are also shown on drawings listed in Section 2 of the Unit Specific Containment ISI Plan. These boundaries include component parts classified as ASME Class NC and are indicated solely for convenience in performing examinations specified by the Owner and do not constitute the boundary for the Class MC containment vessel.

Components Exempted from Examination

Vessels, parts, and appurtenances that are outside the boundaries of the containment as defined in the Design Specifications are exempt. Mechanical process piping penetrations and subassemblies classified as ASME Code Class 2 (NC), including bellow assemblies (except surfaces of connecting welds to penetration sleeves) are exempt. However, examination of some piping penetration subassembly surfaces is included as Owner Specified Examination Categories and Requirements. Also exempted from examination are flanges and covers attached to exterior ends of containment penetration sleeves for those penetrations where flanges and covers are attached to the interior end of the penetration sleeve.

Embedded or inaccessible portions of containment vessels, parts, and appurtenances that met the requirements of the original Construction Code are exempt. Inaccessible portions of containment vessels, parts, and appurtenances are those with surfaces that cannot be visually examined by direct or remote methods on both sides, and which are embedded or otherwise obstructed from view by structures, components, or permanent plant equipment or materials.

Portions of containment vessels, parts, and appurtenances that become embedded or inaccessible as a result of vessel repair or replacement if the conditions of IWE-1232 and IWE-5220 are met are exempt.

Piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel are exempt. These components shall be examined in accordance with the rules of IWB or IWC, as appropriate to the classification defined by the Design Specifications.

Pressure Testing Class MC Components

Except as noted in IWE-5224, a pneumatic leakage test shall be performed following repair/replacement activities performed by welding or brazing, prior to returning the component to service.

There are no periodic system pressure testing requirements for Class MC Components in Subsection IWE.

Examination Boundaries

Section 2 of the Unit Specific Containment ISI Plan contains a listing of drawings that identify examination areas subject to IWE examination. These drawings also show Class MC component supports subject to examination, and include portions of Class NC bellows surfaces, which the Owner has elected to include in this program. Revisions to drawings are reviewed for additions/changes to the ISI boundaries. These additions/changes are incorporated into the Containment ISI Plan as necessary. The controlled drawings are maintained in accordance with applicable procedures and directives.

Inspection Interval and Inspection Periods

The Second and Third Inservice Inspection Intervals for Class MC metal containments and their supports are shown below. Note that these intervals may not coincide with Inservice Inspection Intervals for ASME Class 1, 2, and 3 systems and components. The term "EOC" used below is an abbreviation for "End of Cycle" and the associated number indicates the sequential refueling outage following initial operation of the unit.

Table 3.2.2-1, Second Containment Inservice Inspection Interval

	Unit 1 (Notes 1, 2)		
Start Date			**End Date**
07/15/2005	**07/15/2008**	**07/15/2011**	**08/31/2014**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 17)	Outage 3 (EOC 19)	Outage 5 (EOC 21)	
Outage 2 (EOC 18)	Outage 4 (EOC 20)	Outage 6 (EOC 22)	

(1) The schedule for Interval 2 is included in this discussion as it encompasses the previous Unit 1 ILRT.

(2) The start date for Interval 2 was delayed from 07/15/04 to 07/15/05, within the 12-month adjustment allowed by IWA-2430(d)(1). This adjustment was necessary because of the delay in obtaining NRC approval of Relief Request Serial #03-GO-010.

Table 3.2.2-2, Third Containment Inservice Inspection Interval

	Unit 1 (Notes 1, 2)		
Start Date			**End Date**
09/01/2014	**07/15/2017**	**07/15/2021**	**07/15/2024**
1st Period	2nd Period	3rd Period	
Outage 1 (EOC 23)	Outage 3 (EOC 25)	Outage 6 (EOC 28)	
Outage 2 (EOC 24)	Outage 4 (EOC 26)	Outage 7 (EOC 29)	
	Outage 5 (EOC 27)		

(1) The 2nd Interval end date was extended from 07/15/2014 to 8/31/2014 within the 12 months of 07/15/2014 and less than 11 years of 07/15/2004 as allowed by IWA-2430(d)(1), (Ref. ASME Code 1998 Edition with 2000 Addenda). Therefore, the start date for 3rd Interval is 09/01/2014.

(2) The end date for Interval 3 is adjusted to 10 years from the original end date 07/15/2014 of the 2nd Interval as allowed by IWA-2430(c)(3), (Ref. ASME Code 2007 Edition with 2008 Addenda).

Table 3.2.2-3, Second Containment Inservice Inspection Interval

		Unit 2 (Notes 1, 2)		
Start Date				**End Date**
07/15/2005	**07/15/2008**		**07/15/2011**	**08/31/2014**
1st Period		2nd Period	3rd Period	
Outage 1 (EOC 17)		Outage 3 (EOC 19)	Outage 5 (EOC 21)	
Outage 2 (EOC 18)		Outage 4 (EOC 20)	Outage 6 (EOC 22)	

(1) The schedule for Interval 2 is included in this discussion as it encompasses the previous Unit 2 ILRT.

(2) The start date for Interval 2 was delayed from 07/15/04 to 07/15/05, within the 12-month adjustment allowed by IWA-2430(d)(1). This adjustment was necessary because of the delay in obtaining NRC approval of Relief Request Serial #03-GO-010.

Table 3.2.2-4, Third Containment Inservice Inspection Interval

		Unit 2 (Notes 1, 2)		
Start Date				**End Date**
09/01/2014	**07/15/2017**		**07/15/2021**	**07/15/2024**
1st Period		2nd Period	3rd Period	
Outage 1 (EOC 23)		Outage 3 (EOC 25)	Outage 5 (EOC 27)	
Outage 2 (EOC 24)		Outage 4 (EOC 26)	Outage 6 (EOC 28)	

(1) The 2nd Interval end date was extended from 07/15/2014 to 8/31/2014 within the 12 months of 07/15/2014 and less than 11 years of 07/15/2004 as allowed by IWA-2430(d)(1), (Ref. ASME Code 1998 Edition with 2000 Addenda). Therefore, the start date for 3rd Interval is 09/01/2014.

(2) The end date for Interval 3 is adjusted to 10 years from the original end date 07/15/2014 of the 2nd Interval as allowed by IWA-2430(c)(3), (Ref. ASME Code 2007 Edition with 2008 Addenda).

Examination Categories and Requirements

The examination categories to be used are those listed in Table IWE-2500- I of Section XI. Specific examinations will be identified by an Item Number similar to those listed in Table IWE-2500-1 of Section XI plus an additional number to uniquely identify that examination (except as noted below). (Example: E01.011.001). Class MC Items to be inspected include the following:

Table 3.2.2-5, Category E-A: Containment Surfaces		
Table IWE-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
E1.10 Containment Vessel – Pressure Retaining Boundary		
E1.11	Accessible Surface Areas	General Visual, 100% each Inspection Period (See Note 1)
E1.12	Wetted Surfaces of Submerged Areas	(See Note 2)
E1.30	Moisture Barriers	General Visual, 100% each Inspection Period

(1) If this examination is to be credited towards satisfying the examinations required by 10 CFR 50, Appendix J, the examination shall be performed during the RFO in which a Type A test is to be performed, just prior to the start of the Type A test. Duke Energy Corporation intends to credit Item E1.11 visual exams towards satisfying the requirements of 10 CFR 50, Appendix J.

(2) These examinations are applicable only to BWR containments and are not applicable at MNS.

Table 3.2.2-6, Category E-C: Containment Surfaces Requiring Augmented Examination		
Table IWE-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
E4.10 Containment Surface Areas		
E4.11	Visible Surfaces	VT-1 Visual, 100% Each Period (Deferral Not Permissible)
E4.12	Surface Area Grid, Minimum Wall Thickness Location	Volumetric, Ultrasonic Thickness Measurement 100% Each Inspection Period (Deferral Not Permissible)

Table 3.2.2-7, Category E-G: Pressure Retaining Bolting		
Table IWE-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
E8.10	Bolted Connections	Visual VT-1, 100% End of Interval Deferral Permissible

Table 3.2.2-8, Category F-A: Supports		
Table IWF-2500-1 Item	**Part To Be Examined**	**Examination Requirement(s) Comments**
F1.40	Supports Other Than Piping Supports (Class MC Airlock Supports)	Visual VT-3, 100% End of Interval (See Notes)

(1) Although 10 CFR 50.55a does not include requirements for inservice or preservice examination of Class MC component supports; these supports shall be included within the Containment ISI Plan and shall be examined in accordance with Subsection IWF of the Code.

(2) The provisions of Table IWF-2500-1, footnote (3) may be used to minimize examination of multiple component supports on each airlock.

Owner Specified Examination Requirements

Owner specified examination requirements shall be performed as specified in this Plan. Specific examination listings and schedules are described in Part A, Section 6 of this Plan. Note that the term "Augmented Examinations" is not used in this plan to describe examinations that are above and beyond those required by the Code. An alternative term "Owner Specified Examinations" is used to alleviate confusion with Subsection IWE, Category E-C Augmented Examinations. Owner specified examinations may include those which are the result of regulatory commitments, those required solely by regulation, and may include other examinations deemed appropriate by the Owner for inclusion in this program.

Listing of Owner-Specified Examinations:

1. VT-3 Visual Examination of Fuel Transfer Tube penetration surfaces (including accessible surfaces of the Fuel Transfer Tube Penetration) on the exterior of the containment (Annulus side). These surfaces are not readily accessible because of lead shielding and locked access ports. These areas are posted as Very High Radiation (Grave Danger); however, these conditions exist only during fuel movement. These locations are identified because they are not routinely accessed for general visual

examination in accordance with the Containment ISI Program. A review of similar industry practices confirmed that many licensees consider access to these types of areas to be sufficiently difficult as to justify exemption from the General Visual Examination requirements of the ASME Code, Subsection IWE. The operating experience at MNS has not identified any specific concerns with these areas that would warrant changing the Containment ISI Program to require General Visual Examination of these areas. However, it is deemed appropriate to require that these surfaces receive a VT-3 Visual Examination once during each ten-year interval.

2. Ultrasonic Thickness Measurements on selected surfaces opposite the ice condenser areas once every ten-year interval. The number and extent of these examinations shall be determined by Engineering. At a minimum, some examinations should be performed every ten years opposite the Ice Condenser Floor Slab/containment vessel interface. Additional UT examinations may be warranted at locations near the Ice Condenser Top Deck Doors where condensation has been occasionally observed on the Annulus side of the vessel during scheduled examinations. If conditions are detected during the performance of these examinations, a determination shall be made as to whether the conditions warrant examination of the affected surfaces under Category E-C, Item E4.12.

3. VT-1 Visual Examination of selected surface areas along the embedment zone on the interior side of the Containment Vessel, near elevation 725+0 (Nom.), located behind containment vessel thermal insulation panels. The number and location of selected surface areas is based on engineering judgment of the Containment ISI Program Manager.

4. NRC Information Notice 2014-07 addresses NRC expectations for examination of containment leak chase channel systems. These items shall be added to the Containment ISI Plan as Elective Examinations to be performed 100% during each Inspection Interval (instead of the 100% each Inspection Period as indicated in IN 2014-07). Duke Energy shall elect to perform a VT-3 visual examination in accordance with procedure NDE-67. These examinations may be scheduled and performed as follows:

   - 100% of the containment interior concrete floors shall be examined during each inspection interval. Appropriately 1/3 of the floor surface areas shall be examined during each inspection period to determine the condition of all leak chase channel bronze caps installed in the floor within the examination area.
   - The condition of the bronze caps shall be considered acceptable if there is no evidence of damage or degradation that could result in possible leakage of water into the leakage channel systems. Evidence of boric acid in the vicinity of any bronze cap shall require evaluation by engineering to determine the acceptability of the item.
   - If, during the visual examination of the leak chase channel covers, it is determined that there is evidence of moisture intrusion or degradation of the cover to the extent that moisture intrusion could have occurred, then removal of the leak chase channel cover shall be necessary to assess the condition of the embedded containment liner plate within the leak chase channel. Examination of leak chase channel tubing and embedded containment liner plate using borescope would then be mandatory, and the observed conditions shall be considered reportable per 10 CFR 50.55a(b)(2)(ix)(A).

- The specific location Test No. of each bronze cap shall be identified on the examination record.

Owner-Specified Examination Categories and Requirements

Owner specified examinations are described in the following table.

Table 3.2.2-9, Owner Specified Examinations			
Item Subject to Owner Specified Examination	**Part to Be Examined**	**Examination Requirement(s)**	**Requirement / Comments**
E1.11	Accessible Surfaces of Process Piping Penetration Assemblies Classified as ASME Code Class NC as shown on in service inspection drawings. (See Note 1)	General Visual Examination Each Period (See Note 2)	None. These surfaces do not require examination in accordance with IWC or IWE. However, these surfaces form a part of the containment primary pressure boundary.
E4.12	Surface Area Grid, Minimum Wall Thickness Location	Volumetric (See Note 3)	None. (See Note 3)
F1.40	Class MC Component Supports	Visual, VT -3	None. However, Airlock supports at McGuire shall be treated as NF supports and shall be examined in accordance with IWF requirements at the Owner's discretion.
E1.30	Accessible Surfaces of 100% of Containment Leak Chase Channel Closures	General Visual Examination Each Inspection Period	See Note 4.

Notes:

1. Penetration subassemblies (bellows) are classified as Code Class NC at their welded connection to the first outboard circumferential weld on the containment vessel sleeve. However, the Containment ISI Plan drawings indicate that pressure retaining penetration subassembly surfaces are included within the scope of IWE Examination from the

containment sleeve circumferential weld to the bellows assembly seal weld at the flued head connection. These are Owner specified examinations and need not comply with applicable provisions of IWE or 10 CFR 50.55a(b)(2) and 10 CFR 50.55a(g)(4).

2. Visual Examination shall be performed on accessible surfaces of areas shown on the Containment ISI drawings, except that surfaces of subassemblies extending outside of the Reactor Shield Building need not be examined during general visual examinations specified for Item E1.11.

3. IWE-2500, Table IWE-2500-1, Category E-C, Item E4.12 examinations are performed to determine the minimum wall thickness location within each grid. The examinations specified herein as "Owner Specified" are in addition to those required by the Code and are performed within each grid subject to E4.12 examination when the minimum wall thickness measurement initially recorded lies within 4" of the centerline of any vertical or circumferential weld. These Owner specified examinations are not required by the Code, but have been added by the Owner to ensure that the minimum wall thickness measurements within each grid are not influenced by local plate thinning that typically can occur adjacent to welds. These examinations have been added to be consistent with procedure NDE-951, "Ultrasonic Thickness Measurement of Metallic Containment Structures".

4. A VT-3 visual examination in accordance with NDE-67, " Visual Examination (VT-1 and VT-3) of Metal and Concrete Containment," shall be specified to satisfy the General Visual Examination.

<u>Additional Program Requirements</u>

Additional programmatic requirements specified by 10 CFR 50.55a(b)(2)(ix) are described below:

10 CFR 50.55a(b)(2)(ix)(A)

For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA-6000:

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
- An evaluation of each area, and the result of the evaluation; and
- A description of necessary corrective actions.

10 CFR 50.55a(b)(2)(ix)(B)

When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the

conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

ASME Code 2007 Edition with 2008 Addenda has no Table IWA-2210-1. Duke Energy has elected to comply with the requirements of ASME Code 2007 Edition with 2008 Addenda, Subsection IWA for performance of VT-1 and VT-3 visual examinations on metal containment.

10 CFR 50.55a(b)(2)(ix)(J)

In general, a repair/replacement activity such as replacing a large containment penetration, cutting a large construction opening in the containment pressure boundary to replace steam generators, reactor vessel heads, pressurizers, or other major equipment; or other similar modification is considered a major containment modification. When applying IWE-5000 to Class MC pressure-retaining components, any major containment modification or repair/replacement, must be followed by a Type A test to provide assurance of both containment structural integrity and leak-tight integrity prior to returning to service, in accordance with 10 CFR 50, Appendix J, Option A or Option B on which the applicant's or licensee's Containment Leak-Rate Testing Program is based. When applying IWE-5000, if a Type A, B, or C Test is performed, the test pressure and acceptance standard for the test must be in accordance with 10 CFR 50, Appendix J.

This condition modifies the requirements of IWE-5000 for major containment modification or repair/replacement activities. These activities must be followed by a 10 CFR 50 Appendix J, type A test. This eliminates the bubble test-vacuum box technique detailed in IWE-5224 following major containment repair/replacement activities.

<u>Accessible Surface Areas</u>

The provisions of IWE-1231(a)(3) shall be met. This paragraph describes how the total accessible surface area is to be determined to meet this provision.

The total surface area of the containment (in square feet) was computed for the First Interval Containment ISI Plan. Based on the data and computations documented during ISI Interval 1, it is clear that the requirements of IWE-1231(a) have been met during Interval 1 and the examination surface areas were the same during Interval 2. Because the examination surface areas are the same during Interval 3, these computations need not be repeated in the Third Interval Containment ISI Plan. It is also clear that, unless major modifications are made to the Containment Vessel, the minimum required areas will remain accessible for visual examination from at least one side of the vessel.

<u>Inaccessible Surface Area</u>

The inaccessible surface area is equal to the total surface area obstructed within the areas identified above such that direct or remote visual examination of that area is not possible from both the inside and outside surface. An area need not be considered inaccessible if access permits examination from at least one side of the vessel. However, the location and extent of all surface areas where access is not possible from one side shall be documented. The First and Second Intervals Containment ISI Plan did not identify a significant number of inaccessible surface areas, and additional inaccessible surface areas may be documented during the inspection intervals. If additional inaccessible areas are identified during the third ISI Interval, the ISI Plan shall be updated to document this data and shall be revised as needed to

demonstrate continued compliance with IWE-1231(a)(3). Data on the location and extent of inaccessible surface areas shall be of sufficient detail as to allow these areas to be identified and located on applicable inservice inspection drawings referenced in this Plan.

Requests for Relief from ASME Code and Regulatory Requirements

Currently there are no relief requests for the third interval plan.

3.2.3 License Renewal

The Containment Inservice Inspection Plan - IWE is credited in the joint Catawba and McGuire License Renewal Application (LRA) with managing loss of material due to corrosion of steel surfaces. The Containment Inservice Inspection Plan - IWE implements the requirements of ASME Code Section XI Subsections IWE. The purpose of ASME Subsection IWE examination is to identify and correct degradation of the accessible steel surfaces of the containment liner prior to the loss of the essentially leak tight barrier.

Section 3.5 of the LRA identified the Containment ISI Plan - IWE as an aging management program for Reactor Building containment steel components. The discussion of the program and objective evidence associated with the effectiveness of the program were provided in LRA Appendix B.3.7 of the Catawba and McGuire application.

The Containment Leak Rate Testing Program is credited in the Catawba and McGuire LRA with managing loss of material of steel components of the Reactor Building Containment and cracking of penetration bellows. The purpose of the Containment Leak Rate Testing Program is to assure that leakage through the containment and systems and components penetrating containment shall not exceed allowable leakage rate values specified in the Technical Specifications or associated bases and periodic surveillances of containment penetrations and isolation valves are performed.

Section 3.5 of the LRA identified the Containment Leak Rate Testing Program as an aging management program for steel components of the Reactor Building Containment. The discussion of the program and objective evidence associated with the effectiveness of the program were provided in LRA Appendix B.3.8 of the Catawba and McGuire application..

3.2.4 Integrated Leakage Rate Testing (ILRT) History

Previous Type A tests confirmed that the MNS reactor containment structure has leakage well under acceptance limits and represents minimal risk to increased leakage. Continued Type B and Type C testing for direct communication with containment atmosphere minimize this risk. Also, the Inservice Inspection (IWE/IWL) program and maintenance rule monitoring provide confidence in containment integrity.

To date, five operational Type A tests have been performed on MNS Unit 1 and four on MNS Unit 2. There is considerable margin between these Type A test results and the TS 5.5.2 limit of 0.75 L_a (0.225 % Weight per Day), where L_a is equal to 0.3% by weight of the containment air per day at the peak accident pressure. These test results demonstrate that MNS has low leakage Containments.

Table 3.2.4-1, MNS Unit 1 ILRT Test Results	
Test Date	**Leakage weight % per day**
April 1983	0.1446
August 1986	0.1533
May 1990	0.1965
May 1993	0.1482
October 2008	0.1065

Table 3.2.4-2, MNS Unit 2 ILRT Test Results	
Test Date	**Leakage weight % per day**
May 1986	0.0837
August 1989	0.1138
August 1993	0.1469
March 2008	0.1242

3.3 Containment Leakage Rate Testing Program, Type B and Type C Testing

MNS Type B and C testing program currently requires testing of electrical penetrations, airlocks, hatches, flanges, bellows, and containment isolation valves in accordance with 10 CFR Part 50, Appendix J, Option B. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. Per TS 5.5.2, the allowable maximum pathway total for Type B and C leakage is 0.6 L_a where L_a equals 140,379 sccm.

As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of all potential Containment leakage paths. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

A review of the Type B and Type C test results from 2008 through 2014 for MNS Unit 1 and from 2009 through 2015 for MNS Unit 2 has shown an exceptional amount of margin between the actual As-Found (AF) and As-left (AL) outage summations and the regulatory requirements as described below:

- The As-Found minimum pathway leak rate average for MNS Unit 1 shows an average of 5.97% of L_a with a high of 7.884%L_a.
- The As-Left maximum pathway leak rate average for MNS Unit 1 shows an average of 7.09% of L_a with a high of 8.416% L_a.

- The As-Found minimum pathway leak rate average for MNS Unit 2 shows an average of 1.40% of L_a with a high of 2.603% L_a.

- The As-Left maximum pathway leak rate average for MNS Unit 2 shows an average of 2.03% of L_a with a high of 3.188% L_a.

Tables 3.3-1 and 3.3-2 provide LLRT data trend summaries for MNS since 2008 for Unit 1 and 2009 Unit 2 . This summary shows that there have been no As-Found failures that resulted in exceeding the Technical Specification 5.5.2 limit of 0.6 L_a and demonstrates a history of successful tests.

The following demonstrates a history of satisfactory Type B and Type C tested component performance.

Table 3.3-1, MNS Unit 1 Type B and C LLRT Trend Summary

RFO	2008 M1EOC19 (sccm)	2010 M1EOC20 (sccm)	2011 M1EOC21 (sccm)	2013 M1EOC22 (sccm)	2014 M1EOC23 (sccm)
As-Found Pathway	6049.2	8338.2	11068	7589	8871.4
Percent of La	4.309	5.940	7.884	5.406	6.320
As-Left Pathway	9277	11155.8	11814	10196	7313.6
Percent of La	6.608	7.947	8.416	7.263	5.210

Table 3.3-2, MNS Unit 2 Type B and C LLRT Trend Summary					
RFO	**2009 M2EOC19 (sccm)**	**2011 M2EOC20 (sccm)**	**2012 M2EOC21 (sccm)**	**2014 M2EOC22 (sccm)**	**2015 M2EOC23 (sccm)**
As-Found Pathway	1357.3	1793.94	1420.3	3654.71	1612.3
Percentage of La	0.967	1.278	1.012	2.604	1.149
As-Left Pathway	4475.21	3208.69	1383	2526.24	2649.7
Percentage of La	3.188	2.268	0.985	1.800	1.888

3.4 NRC Information Notices (INs)

3.4.1 IN 2010-12, "Containment Liner Corrosion"

This Information Notice was issued to alert plant operators to three events that occurred where the steel liner of the containment building was corroded and degraded. At Beaver Valley and Brunswick plants material had been found in the concrete which trapped moisture against the liner plate and corroded the steel. In one case, it was material intentionally placed in the building, and in the other case, it was foreign material which had inadvertently been left in the form when the wall was poured. The result in both cases was that the material trapped moisture against the steel liner plate leading to corrosion. In the third case, Salem, an insulating material placed between the concrete floor and the steel liner plate adsorbed moisture and led to corrosion of the liner plate.

Discussion:

All the referenced examples are from sites with Concrete Primary Containments with steel liners. In a Concrete Containment, the liner, in addition to helping provide a leak tight barrier, acts as the inner "form" when the concrete is poured into a containment structure. In these cases, the liner and associated sleeves and penetrations are in direct contact with the concrete, or are in contact with intermediate materials installed during construction that communicate between the liner and concrete.

Duke Energy conducted a focused self-assessment of Containment Integrity in 2008. The assessment reviewed the Containment ISI (ASME BPV Code Section XI) Program, the Boric Acid Corrosion Program, the 10 CFR 50 Appendix J Program, and the Coatings Program, as well as Operating Experience Data for three Duke Energy Nuclear sites, Oconee, McGuire and Catawba. The McGuire specific concerns identified in the report were tracked in the Corrective Action Program.

The assessment found that overall the Programs were satisfying the purposes for which they had been established; however, there were some risks identified in the assessment that could justify changes to the programs. The most significant risks to the McGuire plant were listed as the Unit 2 Fuel Transfer Tube Area and the Thermal Insulation Panels. After this assessment was complete, but before all the Corrective Actions and the assessment were closed, the NRC issued IN 2010-12. Due to the related nature of the issues, the response to the IN was included into a new Corrective Action, which contained the assessment and the responses. The new Corrective Action was to process an Engineering Change Request, which led to an Engineering Change to provide details for an easy access inspection port through the Thermal Barrier on the Steel Containment Vessel. This EC supported the needed ISI inspection where the access to the surface of the containment plate is not accessible.

No further actions were recommended based on evaluation of the OE provided in NRC Information Notice 2010-12.

3.4.2 IN 2014-07, Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner

The containment basemat metallic shell and liner plate seam welds of pressurized water reactors are embedded in a 3-foot to 4-foot thick concrete floor during construction and are typically covered by a leak-chase channel system that incorporates pressurizing test connections. This system allows for pressure testing of the seam welds for leak-tightness during construction and also in service, as required. A typical basemat shell or liner weld leak-chase channel system consists of steel channel sections that are fillet welded continuously over the entire bottom shell or liner seam welds and subdivided into zones, each zone with a test connection.

Each test connection consists of a small carbon or stainless steel tube (less than 1-inch diameter) that penetrates through the back of the channel and is seal-welded to the channel steel. The tube extends up through the concrete floor slab to a small steel access (junction) box embedded in the floor slab. The steel tube, which may be encased in a pipe, projects up through the bottom of the access box with a threaded coupling connection welded to the top of the tube, allowing for pressurization of the leak-chase channel. After the initial tests, steel threaded plugs or caps are installed in the test tap to seal the leak-chase volume. Gasketed cover plates or countersunk plugs are attached to the top of the access box flush with the containment floor. In some cases, the leak-chase channels with plugged test connections may extend vertically along the circumference of the cylindrical containment shell or liner to a certain height above the floor.

Discussion:

During a 2EOC23 containment walkdown in response to IN 2014-07, four Steel Containment Vessel Base Liner Plate Leak Chase Test Channel enclosure locations, #45, #89, #144 and #61, in the concrete floor were discovered to be concrete-type filled and missing the bronze covers (caps) designed to enclose the test ports. The sealed as-found condition of these four test channel enclosures indicates that no leakage into the test channels below has occurred.

During a 1EOC23 containment walkdown in response to IN 2014-07, one location, #86, was found without a bronze cover during 1EOC23 (AR 1686595). Similar to the Unit 2 as found condition, the test port was sealed with a hard, solid material obstructing the tube inner

diameter. There was a limited extent of condition performed for other test channel enclosures in the area (approximately 10%) with no adverse findings.

An open test port could permit moisture/foreign material intrusion to the embedded containment base liner plate surfaces within the test channels. The sealed as-found condition of the above listed test channel enclosures indicated that no leakage into the test channels below had occurred.

Specified Safety Function

The function of the cover and seal on the test channel enclosures is to protect the steel containment vessel (SCV) from moisture intrusion. The following Technical Specifications are applicable to the SCV system:

- ITS 3.6.1, Containment
- ITS 5.5.2, Containment Leak Rate Testing Program

The function of the Steel Containment Vessel (SCV) Structure is to provide an essentially leak-tight barrier against the release of radioactivity and energy that may result from a Design Basis Event. The steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment operable, limits the leakage of fission product radioactivity from the containment to the environment.

Testing of the Base Liner Plate welds using the test channel ports was performed during original construction; afterwards, the test channel enclosures and test ports were sealed and covered to protect the liner plate from moisture and foreign material intrusion. Foreign material intrusion alone does not introduce a degradation mechanism to the SCV. Continued containment integrity is currently monitored via the Containment Leak Rate Testing Program.

Unit 2 Evaluation

There is reasonable assurance that moisture intrusion affecting the integrity of the SCV has not occurred, as noted below:

- For the four Unit 2 enclosures found without covers, the leak chase channel test tubing is sufficiently sealed by concrete and/or grout and additional coatings over two of the covers; therefore, there is no credible evidence that moisture intrusion has occurred into the embedded portions of the containment liner plate within the affected leak chase test channels since original construction. Efforts were made to expose the 1/2 inch test channel tubing for further interrogation, however the materials sealing the test channel port were unable to be removed without causing significant damage to the leak chase enclosure.

- For the remaining enclosures with covers installed per the design drawings, the design of the leak chase enclosure cover is such that once torqued into place the cover with the seal provides an adequate barrier to moisture intrusion. The sealant material is compressed between a metal sleeve and the bronze cover. On many of the enclosures, the epoxy floor coating covered the joint interface between the bronze cover and enclosure sleeve which provides an additional barrier.

- Any legacy moisture/foreign material intrusion into the test channel from original construction would have become anoxic, and thus any degradation would be self-limiting.

- ILRT data shows containment is sufficiently sealed and adequate margin exists. The McGuire Unit 2 ILRT mass point leakage history confirms there is not degradation of the steel containment liner.

  There remains a large margin between the latest 2008 ILRT results at 0.1242% per day, and the limit of 0.225% per day. Note that the bronze caps are not removed when performing ILRT testing.

- Industry operating experience shows similar or more significant findings, such as missing plugs, completely open ports, evidence of boric acid, and channels filled with water; none of which resulted in degradation of the base liner plate that challenged operability.

Based on the discussion above, there is reasonable assurance that the McGuire Unit 2 SCV Base Liner Plate is not degraded; and therefore, the SCV is capable of performing its specified safety function. This evaluation is also applicable and bounding to McGuire Unit 1.

<u>Extent of Condition</u>

Unit 2
All 142 test enclosure locations have been dispositioned (identified or deemed inaccessible) and the four locations with missing bronze covers were addressed herein. It is planned to perform qualified visual inspection of all accessible bronze covers during the 2EOC24 (Spring 2017) refueling outage per the guidance in NRC IN 2014-07. This inspection is intended to verify that the bronze covers remain adequately sealed. In response to NRC IN 2014-07, these inspections have been added to the ISI plan.

Unit 1
One location, #86, was found without a bronze cover during 1EOC23. Similar to the Unit 2 as found condition, the test port was sealed with a hard solid material obstructing the tube inner diameter. There was a limited extent of condition performed for other test channel enclosures in the area (approximately 10%) with no adverse findings. Unit 1 ILRT data was reviewed with large margin between the latest 2008 ILRT results at 0.1065% per day, and the limit of 0.225% per day. There is reasonable assurance that all Unit 1 test channel enclosures are covered and/or are otherwise providing an adequate moisture barrier function. An investigation to identify all Unit 1 test channel enclosures is planned for the 1EOC24 (Fall 2017) refueling outage. This evaluation is applicable and bounding to Unit 1.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/ Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. 10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR Part 50, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." Appendix J

specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing, and reporting requirements for each type of test.

RG 1.163 was developed to endorse NEI 94-01, Revision 0 with certain modifications and additions.

The adoption of the Option B performance-based containment leakage rate testing for Type A, Type B, and Type C testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR Part 50, Appendix J, the test frequency is based upon an evaluation that reviewed "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage.

EPRI TR-1009325, Revision 2, provided a risk impact assessment for optimized ILRT intervals up to 15 years, utilizing current industry performance data and risk informed guidance. NEI 94-01, Revision 2-A, Section 9.2.3.1 states that Type A ILRT intervals of up to 15 years are allowed by this guideline. The Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, EPRI Report 1018243 (Formerly TR-1009325, Revision 2) indicates that, in general, the risk impact associated with ILRT interval extensions for intervals up to 15 years is small. However, plant-specific confirmatory analyses are required.

The NRC staff reviewed NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2. For NEI TR 94-01, Revision 2, the NRC staff determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J. This guidance includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. The NRC staff found that the Type A testing methodology as described in ANSI/ ANS-56.8-2002, and the modified testing frequencies recommended by NEI TR 94- 01, Revision 2, serves to ensure continued leakage integrity of the containment structure. Type B and Type C testing ensures that individual penetrations are essentially leak tight.

In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

4.2 Precedent

This license amendment request is similar in nature to the following license amendments previously approved by the NRC to extend the Type A test frequency:

- December 29, 1994 (ML011080782), for Nine Mile Point Nuclear Station Unit 1,
- June 2, 2003 (ML031320686), for Vermont Yankee Nuclear Power Station,
- July 20, 2009 (ML091540158) for Arkansas Nuclear One, Unit No. 2,
- August 23, 2010 (ML102090137) for Palisades Nuclear Plant.
- October 1, 2012 (ML12250A339) for Oconee Nuclear Station Unit 1.
- August 5, 2013 (ML13193A329) for Oconee Nuclear Station Units 2 and 3.

4.3 Significant Hazards Consideration

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below. There is no significant hazards consideration involved with the proposed corrections to typographical and administrative errors introduced by previous license amendments.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment to the Technical Specifications (TS) involves the extension of the McGuire Nuclear Station (MNS) Type A containment integrated leak rate test interval to 10.5 years. The current Type A test interval of 120 months (10 years) would be extended on a one-time basis to no longer than 10.5 years from the last Type A test. This extension is bounded by the 15 month extension, permissible only for non-routine emergent conditions, allowed in accordance with NEI 94-01 revision 0. The proposed extension also does not change the test method or procedure. The containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. The containment and the testing requirements invoked to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. The change in dose risk for changing the Type A test frequency from 10 years to 10.5 years, measured, as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.023 person-rem/year. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of ≤ 1.0 person-rem per year, or ≤ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

   As documented in NUREG-1493, Performance-Based Containment Leak-Test Program, Type B and C tests have identified a very large percentage of containment leakage paths, and the percentage of containment leakage paths that are detected only by Type A testing is very small. The MNS Type A test history supports this conclusion.

   The integrity of the containment is subject to two types of failure mechanisms that can be categorized as: (1) activity based, and; (2) time based as previously discussed. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as configuration management and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the containment combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and TS requirements serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by a Type A test. Based on the above, the proposed extensions do not significantly increase the consequences of an accident previously evaluated.

Therefore, the proposed change does not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed amendment to the TS involves the extension of the MNS Type A containment integrated leak rate test interval from 10 years to 10.5 years. The current Type A test interval of 120 months (10 years) would be extended on a one-time basis to 10.5 years from the last Type A test. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed amendment to TS 5.5.2 involves the extension of the MNS Type A containment integrated leak rate test interval to 10.5 years. The current Type A test interval of 120 months (10 years) would be extended on a one-time basis to no longer than 10.5 years from the last Type A test. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the TS Containment Leak Rate Testing Program exist to ensure that the degree of containment structural integrity and leak tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by TS is maintained.

   The proposed change involves only the extension of the interval between Type A containment leak rate tests for MNS. The proposed surveillance interval extension is bounded by the 15-year ILRT interval currently authorized within NEI 94-01, Revisions 2-A and 3-A. Industry experience supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, and TS serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Type A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the approval of this proposed change, since these are not affected by changes to the Type A test intervals.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

RG 1.163 was developed to endorse NEI 94-01, Revision 0 with certain modifications and additions.

The adoption of the Option B performance-based containment leakage rate testing for Type A testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, the test frequency is based upon an evaluation that review "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

Based on the previous ILRT tests conducted at MNS, it is concluded that the extension of the containment ILRT interval from 10 to 10.5 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR Part 50, Appendix J and the overlapping inspection activities performed as part of the following MNS inspection programs:

- Containment Inservice Inspection Program (IWE/IWL)
- Periodic Condition Assessments of Service Level I Coatings
- Containment Structural Integrity Inspection

This conclusion is supplemented by a plant-specific confirmatory analysis provided in Section 3.1.4 of this submittal. The findings of the assessment confirm the general findings of previous studies, on a plant-specific basis, that extending the ILRT interval from ten to 10.5 years is considered to be insignificant since it represents a "small" change to the MNS risk profile.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program, September 1995.

2. NEI 94-01, Revision 3-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 2012.

3. NEI 94-01, Revision 2-A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, October 2008.

4. Regulatory Guide 1.174, Revision 2, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, May 2011.

5. Regulatory Guide 1.200, Revision 2, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, March 2009.

6. NEI 94-01, Revision 0, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, July 1995.

7. NUREG-1493, Performance-Based Containment Leak-Test Program, January 1995.

8. EPRI TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, August 1994.

9. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI), Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (TAC No. MC9663), dated June 25, 2008.

10. ML15106A627, Letter from R. Hill (ASME) to J. Lubinski (NRC) dated April 13, 2015. ASME Code, Section XI Actions to Address Requirements for Examination of Containment Leak-Chase Channels.

11. Letter from S. Bahadur (NRC) to B. Bradley (NEI), Final Safety Evaluation of Nuclear Energy Institute (NEI) Report 94-01, Revision 3, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J (TAC No. ME2164), dated June 8, 2012.

12. ML022540102, Letter from R. Martin (NRC) to H. Barron (Duke), Issuance of Amendment No. 207 to Facility Operating License NPF-9 and Amendment No. 188 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2 (TAC Nos. MB3565 and MB3566) dated September 4, 2002.

13. ML030760032, Letter from R. Martin (NRC) to D. Jamil (Duke), Issuance of Amendments RE: One-time Change in the Appendix J, Type A Containment Integrated Leakage Rate Test Interval- McGuire Nuclear Station, Units 1 and 2 (TAC Nos. MB5307 and MB5308) dated March 12, 2003.

14. ML031280431, Letter from R. Martin (NRC) to D. Jamil (Duke), McGuire Nuclear Station, Units 1 And 2 Re: Issuance Of Amendments (TAC NOS. MB6500 and MB6501) dated May 8, 2003.

15. ML14261A051, Letter from J. Lubinski (NRC) to K. Ennis (ASME) dated March 3, 2015. NRC Information Notice 2014-07 Regarding Inspection of Containment Leak-Chase Channels

16. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325, EPRI, Palo Alto, CA. 1018243, October 2008.

17. Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Rev. 4, Developed for NEI by EPRI and Data Systems and Solutions, November 2001.

18. Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC (Document Control Desk), Docket No. 50-317, dated March 27, 2002.

19. ML102090137, Letter from M. Chawla (NRC) to Vice President, Operations (Entergy) dated August 23, 2010. Palisades Nuclear Plant - Issuance of Amendment Re: One-Time Extension to the Integrated Leak Rate Test Interval (TAC NO. ME2122).

20. ML13193A329, Letter from J. Boska (NRC) to S. Batson (Duke) dated August 5, 2013. Oconee Nuclear Station, Units 2 And 3, Issuance of Amendments Regarding Extension of the Reactor Building Integrated Leak Rate Test (TAC NOS. ME9777 AND ME9778).

21. ML12250A339, Letter from J. Boska (NRC) to P. Gillespie (Duke) dated October 1, 2012. Oconee Nuclear Station, Unit 1, Issuance of Amendment Regarding Extension of the Reactor Building Integrated Leak Rate Test (TAC NO. ME8407).

22. ML011080782, Letter from D. Brinkman (NRC) to R. Sylvia (Niagara Mohawk) dated December 29, 1994. Issuance of Amendment For Nine Mile Point Nuclear Station Unit No. 1 (TAC NO. M90278).

23. ML031320686, Letter from R. Pulsifer (NRC) to J. Thayer (Entergy) dated June 2, 2003. Vermont Yankee Nuclear Power Station - Issuance of Amendment Re: One-Time Extension of Appendix J Type A Integrated Leakage Rate Test Interval (TAC NO. MB6507).

24. ML091540158, Letter from N. Kalyanam (NRC) to Vice President, Operations (Entergy) dated July 20, 2009. Arkansas Nuclear One, Unit No.2 - Issuance of Amendment Re: One-Time Extension to 10-Year Frequency of Integrated Leak Rate Test (TAC NO. MD9502).

25. ML073400670, Letter from J. Stang (NRC) to G. Peterson (McGuire) dated February 13, 2008. McGuire Nuclear Station, Unit 1 Issuance Of Amendment Regarding Extension Of Appendix J, Type A Integrated Leakage Rate Test Interval (TAC No. MD4654).

Attachment 1

Technical Specification Pages

(Mark-up)

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Plant Manager or Radiation Protection Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions: March 31, 2008 October 21, 2008

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the ~~May 27, 1993~~ (Unit 1) and ~~August 20, 1993~~ (Unit 2) Type A test shall be performed no later than plant restart after the End Of Cycle ~~19~~ 26 Refueling Outage (Unit 1) and ~~August 19, 2008~~ no later than plant restart after the End Of Cycle 25 Refueling Outage (Unit 2), and

5.5 Programs and Manuals (continued)

b. ~~a.~~ The containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR ~~3.6.11~~ 3.6.1.1 Type A test, prior to initiating the Type A test.

~~5.5.2 Containment Leakage Rate Testing Program (continued)~~

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

Leakage Rate acceptance criteria are:

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

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

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing frequencies required by 10CFR50, Appendix J.

5.5.3 Primary Coolant Sources Outside Containment

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

a. Preventive maintenance and periodic visual inspection requirements; and

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

5.5.4 Deleted

Attachment 2

Technical Specification Pages

(Retyped)

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the Plant Manager or Radiation Protection Manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Containment Leakage Rate Testing Program

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

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the October 21, 2008 (Unit 1) and March 31, 2008 (Unit 2) Type A test shall be performed no later than plant restart after the End Of Cycle 26 Refueling Outage (Unit 1) and no later than plant restart after the End Of Cycle 25 Refueling Outage (Unit 2), and

5.5 Programs and Manuals (continued)

b. The containment visual examinations required by Regulatory Position C.3 shall be conducted 3 times every 10 years, including during each shutdown for SR 3.6.1.1 Type A test, prior to initiating the Type A test.

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

Leakage Rate acceptance criteria are:

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

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

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing frequencies required by 10CFR50, Appendix J.

5.5.3 Primary Coolant Sources Outside Containment

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

a. Preventive maintenance and periodic visual inspection requirements; and

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

5.5.4 Deleted