Table of Contents

List of Tables

18.0 Aging Management Programs and Activities

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18.1 Introduction

Duke Energy Corporation prepared an Application for Renewed Operating Licenses of McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2 (Application) [Reference 1]. The application, including information provided in additional correspondence, provides sufficient information for the NRC to complete their technical and environmental reviews and provides the basis for the NRC to make the findings required by §54.29 (Final Safety Evaluation Report – Final SER) [Reference 2]. Pursuant to the requirements of §54.21(d), the UFSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by §54.21 (a) and (c), respectively.

As an aid to the reader, Table 18-1 provides a summary listing of the programs, activities and time-limited aging analyses (TLAA) (topics) required for license renewal. The first column of Table 18-1 provides a listing of these topics. The second column of Table 18-1 indicates where the topic is located in the Application. This is an historical reference. The third column of Table 18-1 identifies where the description of the Program, Activity, or TLAA is located in either the McGuire UFSAR or in the McGuire Improved Technical Specifications (ITS).

Section 18.2 contains summary descriptions of the aging management programs and periodic inspections that are ongoing through the duration of the operating licenses of McGuire Nuclear Station.

Station documents will be established, implemented, and maintained to cover the aging management programs and activities described in Chapter 18.

The Problem Investigation Process (PIP) (NSD-208) provides a structured approach for a formal corrective action program which facilitates the prioritization, evaluation, and correction of conditions adverse to quality, as defined by 10 CFR Part 50, Appendix B. This same corrective action program is credited for systems, structures, and components whose aging will be managed by the aging management programs and activities described herein.

18.1.1 References

1. M. S. Tuckman (Duke) letter dated June 13, 2001, to Document Control Desk (NRC), Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, Docket Nos. 50-369, 50-370, 50-413, and 50-414.
2. NUREG-1772, Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, Docket Nos. 50-369, 50-370, 50-413, and 50-414, March 2003.

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18.2 Aging Management Programs and Activities

18.2.1 Alloy 600 Aging Management Review

The original Alloy 600 Aging Management Review was proposed during the license renewal review process for McGuire Nuclear Station, which was completed with the issuance of renewed operating license on December 5, 2003. This program description is being revised to reflect requirements imposed and commitments made subsequent to issuance of the renewed operating license. Unless otherwise noted, the intent of the original Alloy 600 Aging Management Review is met by the more comprehensive Alloy 600 Aging Management Program.

The purpose of the Alloy 600 Aging Management Review is to ensure that nickel-based alloy locations are adequately inspected. The program will facilitate the general oversight and management of cracking due to primary water stress corrosion cracking (PWSCC).

Consideration of industry operating experience is part of the Alloy 600 Aging Management Program. The NRC staff has issued several generic communications regarding degradation of Alloy 600. These communications imposed requirements that cover specific components in the Reactor Coolant System. The NRC Staff has approved ASME Code Cases N-722-1, N-729-1, and N-770-1 with conditions specified in 10 CFR Part 50.55a. ASME Code Case N-722-1 requires additional visual examinations of components fabricated with Alloy 600/82/182 materials. ASME Code Case N-729-1 requires additional surface and/or volumetric examinations of the reactor vessel head. ASME Code Case N-770-1 provides the inspections requirements for mitigated and unmitigated Class 1 PWR piping and vessel nozzle butt welds fabricated from Alloy 600/82/182 materials. ASME Code Case N-770-1 with the conditions specified in 10 CFR Part 50.55a replaces the NEI-03-08 mandatory document, MRP-139 Rev. 1 Primary System Piping Butt Weld Inspection and Evaluation Guideline. Additionally, ongoing inspections will continue of the Alloy 600 susceptible materials located on the pressurizer in accordance with NRC Bulletin 2004-01. The subsequent sections describe the general requirements of the Alloy 600 Aging Management Program and the specific inspections performed for the reactor vessel head and the pressurizer.

The Alloy 600 Aging Management Program will be updated as necessary to reflect any new or revised commitments made by Duke Energy in response to industry operating experience or NRC generic communications related to Alloy 600.

Deleted Per 2012 Update.

18.2.1.1 Susceptibility Ranking

As part of the Alloy 600 Aging Management Program all of the Alloy 600/82/182 and Alloy 690/52/152 locations have been identified and ranked based on each location's susceptibility to PWSCC. The Alloy 600 susceptibility ranking is a qualitative ranking determined by temperature, type of weld in the component, post weld heat treatment, and industry operation experience. The susceptibility ranking is doucmented in calculation DPC-1201.01-00-0009. The Alloy 600 susceptibility ranking was initially created to ensure nickel based alloy locations were adequately inspected through the Inservice Inspection, Steam Generator, and Reactor Vessel Internals Programs. However, the Alloy 600 susceptibility ranking was never used independently to augment the ISI program or any other inspection program. The Alloy 600 susceptibility ranking in conjunction with mandatory inspection and evaluation guidelines issued by NEI, ASME Code Cases made mandatory by the NRC, and other NRC bulletins were all applied to ensure Alloy 600 degradation was monitored appropriately.

One of the commitments made as part of the original Alloy 600 Aging Management Review was for Duke to submit to the NRC the results of the susceptibility ranking for pressurizer surge and spray nozzle

thermal sleeves attachement welds prior to the extended period of operation. Duke understands that the staff will review these results and may request additional information to gain an understanding of the results.

For McGuire, the results for the pressurizer surge and spray nozzle thermal sleeves attachment welds will be submitted to the NRC following issuance of renewed operating licenses for McGuire Nuclear Station and prior to June 12, 2021 (the end of the initial license of McGuire Unit 1).

18.2.1.2 Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetration Inspection

Scope – The scope of the Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection includes the control rod drive mechanism nozzles, head vent penetrations and carbon steel head surface of each reactor vessel as described in ASME Section XI Code Case N-729-1 subject to the conditions in paragraphs (g)(6)(ii)(D)(2) through (6) of 10CFR Part 50.55a. These penetrations include 78 Control Rod Drive Mechanism (CRDM) type penetrations, and one head vent penetration.

The four auxiliary head adaptors (AHAs) are located on the outer portion of the reactor vessel head at 0°, 90°, 180°, and 270°. The AHAs shall be volumetrically inspected on a 7-year inspection frequency in accordance with ASME Code Case N-770-1 and the conditions specified in 10 CFR Part 50.55a dated June 21, 2011.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection monitors cracking of nickel based alloy nozzles with partial penetration welds in the reactor vessel closure head and associated borated water leakage onto the closure head carbon steel surface.

The AHA inspection monitors cracking of nickel based alloy nozzles with full penetration butt welds in the reactor vessel closure head and associated borated water leakage onto the closure head carbon steel surface.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending below, The Control Rod Drive Mecchanism and Other Vessel Closure Penetration Inspection will detect cracking of nickel based alloy reactor vessel head penetrations prior to loss of component intended function.

Monitoring & Trending – The Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection will inspect all Reactor Pressure Vessel (RPV) head pressure-retaining partial-penetration weld nozzles and the RPV head surface. This program will consist of both visual and volumetric examinations.

The McGuire RPV heads are composed of PWSCC-susceptible materials. The following is a brief summary of the inspections required by ASME Section XI Code Case N-729-1 subject to the conditions set forth in 10CFR Part 50.55a, paragraphs (g)(6)(ii)(D)(2) through (6) for heads with UNS N06600 nozzles and UNS N06082 or UNS W86182 partial-penetration welds:

- A bare metal visual examination of the entire outer RPV head surface including 360º around each penetration nozzle each refueling outage. If the Effective Degradation Years (EDY) are less than 8 and no flaws are unacceptable for continued service, the reexamination frequency may be extended to every 3rd refueling outage or 5 calendar years, whichever is less, provided an IWA-2212 VT-2 visual examination of the head is performed under the insulation through multiple access points in outages that the VE is not completed.
- A volumetric and/or surface examination of all partial-penetration weld nozzles, every 8 calendar years or before the Reinspection Years (RIY) is equal to 2.25, whichever is less. These examinations

should cover essentially 100% of the required volume or equivalent surfaces of the nozzle tube, as identified by Figure 2 of ASME Code Case N-729-1. A demonstrated volumetric or surface leak path assessment through all J-groove welds shall be performed.

The AHAs contain two Alloy 600/82/182 full penetration butt welds per AHA and operate at the cold leg operating temperature, which places these components within scope of ASME Code Case N-770-1. The AHAs shall be categorized as Inspection Item B as specified in ASME Code Case N-770-1 and shall receive volumetric examinations every second inspection period not to exceed 7 years.

Acceptance Criteria – The visual and volumetric/surface examinations will use acceptance criteria set forth in ASME Code Case N-729-1 subject to the conditions in 10CFR Part 50.55a.

Corrective Action & Confirmation Process – For the bare metal visual inspection, if leakage is detected, the source of leakage and leakpath will be determined and repairs completed. Specific corrective actions and confirmation are implemented in accordance with the Corrective Action Program and NSD 322, "Boric Acid Corrosion Control Program".

Any indications detected during the volumetric examinations of the AHAs and cannot be justified for continued service by analysis shall be repaired in accordance with ASME Section XI. Indications which can be justified for continued service shall be managed by the station's Corrective Action Program and in accordance with ASME Code Case N-770-1 subject to the conditions specified in paragraphs (g)(6)(ii)(F)(2) through (g)(6)(ii)(F)(10).

For the volumetric examination, indications detected during volumetric examination which can not be justified for continued service by analysis will be repaired in accordance with ASME Section XI. Flaws which can be justified for continued service will be managed by the station Corrective Action Program and in accordance with ASME Code Case N-729-1 subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of 10CFR Part 50.55a.

Administrative Controls – Inspections will be controlled by site specific procedures. Engineering evaluations are performed in accordance with the station Corrective Action Program.

a. Prior to the September 10, 2008 10 CFR Part 50 Rule Change, RPV head examinations were dictated by First Revised NRC Order EA-03-009.

b. Per 10 CFR 50.55a (g)(6)(ii)(D)(1), all licensees of PWR's shall augment their inservice inspection program with ASME Code Case N-729-1 subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6). This augmented inservice inspection program should be implemented by December 31, 2008. Once a licensee implements this requirement, the First Revised NRC Order EA-03-009 (Reference 14) no longer applies and is deemed to be withdrawn.

c. Licensees of existing, operating pressurized-water reactors as July 21, 2011 shall implement the requirements of ASME Code Case N-770-1, subject to conditions (g)(6)(ii)(F)(2) through (g)(6)(ii)(F)(10) of 10 CFR Part 50.55a by the first refueling outage after August 22, 2011.

18.2.1.3 Pressurizer Inspection

Scope – The scope of the Pressurizer Inspection includes pressurizer connections that utilize Alloy 600 wrought or Alloy 82/182 weld materials and the manway diaphragm plate seal weld. This inspection ensures that commitments made in response to NRC Bulletin 2004-01 are satisfied (References 15, 16 and 17).

Preventive Actions – Perform structural weld overlays on Alloy 600 wrought or Alloy 82/182 weld materials using materials that are highly resistant to PWSCC.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending below, the Pressurizer inspection will detect cracking of pressurizer connections containing Alloy 600/82/182 materials and the pressurizer manway diaphragm plate seal weld prior to loss of component intended function.

Monitoring and Trending –

The following inspections will be performed each refueling outage:

1. A bare metal visual inspection around 100% of each pressurizer connections that utilize Alloy 600 wrought or Allow 82/182 weld materials (except those penetrations subject to volumetric or surface ISI during that RFO) and
2. A bare metal visual inspection of the gap between the manway cover and pressurizer manway for evidence of manway diaphragm plate seal weld leakage.

Additionally, perform volumetric examination of any structural weld overlays per ASME Section XI and Code Case N-504-2 requirements.

Acceptance Criteria – Any boron detected on the outside of the vessel due to leakage is unacceptable.

Corrective Action and Confirmation Process – Evidence of leakage will be addressed in accordance with the Boric Acid Corrosion Control Program, including evaluation by engineering to determine extent of condition and applicability to other locations. The station Corrective Action Program will be utilized to evaluate the need for additional NDE methods and increased inspection scopes, including like locations and other Duke units (Reference 16).

If circumferential cracking is observed in either the pressure boundary or non-pressure boundary portions of any locations covered under the scope of NRC Bulletin 2004-01, Duke will develop plans to perform an adequate extent-of-condition evaluation and Duke will discuss those plans with cognizant NRC technical staff prior to restarting the affected unit (Reference 17).

Administrative Controls – Inspections results will be documented according to applicable procedures. Engineering evaluations are performed in accordance with the Duke Corrective Action Program and Boric Acid Corrosion Control Program.

18.2.2 Borated Water Systems Stainless Steel Inspection

Scope – The scope of the Borated Water Systems Stainless Steel Inspection is stainless steel components exposed to an alternate wetting and drying borated water environment in the following McGuire systems:

- Containment Spray
- Refueling Water

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameters inspected by the Borated Water Systems Stainless Steel Inspection are pipe wall thickness, as a measure of loss of material, and evidence of cracking.

Detection of Aging Effects – The Borated Water Systems Stainless Steel Inspection is a one-time inspection that will detect the presence and extent of loss of material or cracking of stainless steel components.

Monitoring & Trending – The Borated Water Systems Stainless Steel Inspection will inspect stainless steel components, welds, and heat affected zones, as applicable, in the Containment Spray System in the area of the internal air/water interface. The borated water environment found downstream of valves NS-

12, 15, 29, 32, 38, and 43 in the Containment Spray System at McGuire is stagnant and isolated from the remainder of the system, and therefore, not controlled by the Chemistry Control Program. Water from the refueling water storage tank is introduced during valve testing with level in the piping reaching the same elevation as the tank. Since the pipe is open to containment, evaporation occurs and concentration of contaminants could occur at the air/water interface. This concentration of contaminants could lead to loss of material or cracking. Therefore, a one-time inspection around this water line is warranted.

One of twelve possible locations at McGuire will be inspected using a volumetric technique. If no parameters are known that would distinguish the susceptible locations, one of the twelve available at McGuire will be examined based on accessibility and radiological concerns. The results of this inspection are considered to be bounding, will serve as a leading indicator and can be applied to the specific stainless steel components exposed to an alternate wetting and drying borated water environment in the Refueling Water System.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

No actions are taken as part of this activity to trend inspection results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The acceptance criteria for the Borated Water Systems Stainless Steel Inspection is no unacceptable loss of material or cracking that could result in a loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, then the aging management review is complete and no further action is required. If engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The Borated Water Systems Stainless Steel Inspection will be implemented in accordance with controlled plant procedures.

18.2.3 Bottom-Mounted Instrumentation Thimble Tube Inspection Program

Scope – The scope of the Bottom Mounted Instrumentation Thimble Tube Inspection Program includes all accessible thimble tubes installed in each reactor vessel.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The Bottom Mounted Instrumentation Thimble Tube Inspection Program monitors tube wall degradation of the BMI thimble tubes. Failure of the thimble tubes would result in a breach of the reactor coolant pressure boundary; however, this breach is isolatable via the thimble cutoff valve.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending below, the Bottom Mounted Instrumentation Thimble Tube Inspection Program will detect loss of material due to wear prior to loss of component intended function.

Monitoring & Trending – Inspection of the BMI thimble tubes is performed using eddy current testing. All accessible thimble tubes are inspected. The frequency of examination is based on an analysis of the data obtained using wear rate relationships that are predicted based on Westinghouse research that is presented in WCAP-12866, Bottom Mounted Instrumentation Flux Thimble Wear [Reference 2]. These wear rates, as well as the results of the eddy current examinations, are documented in site specific calculations. The eddy current results are trended and inspections are planned prior to the refueling outage in which thimble tube wear is predicted to exceeding the Acceptance Criteria, below. This ensures that the thimble tubes continue to perform their pressure boundary function.

Acceptance Criteria – The acceptance criterion for the BMI thimble tubes is 80% through wall (thimble tube wall thickness is not less than 20% of initial wall thickness). This acceptance criterion was developed by Westinghouse in WCAP 12866, "Bottom Mounted Instrumentation Flux Thimble Wear," and reported to the NRC by Duke [Reference 1.]

Corrective Action & Confirmation Process – Thimble tubes that are predicted to exceed the acceptance criterion may be capped or repositioned. Specific corrective actions and confirmatory actions are implemented in accordance with the corrective action program.

Administrative Controls – Data are collected and evaluated using written procedures. The data are evaluated and the timing for the next inspection is determined using engineering calculations using methodology based on the information Westinghouse developed in WCAP-12866 [Reference 2].

18.2.4 Chemistry Control Program

The purpose of the Chemistry Control Program is to manage loss of material and/or cracking of components exposed to borated water, closed cooling water, fuel oil, and treated water environments. This program manages the relevant conditions that lead to the onset and propagation of loss of material, cracking, and fouling which could lead to a loss of structure or component intended functions. Relevant conditions are specific parameters such as halogens, dissolved oxygen, conductivity, biological activity, and corrosion inhibitor concentrations that could lead to loss of material and/or cracking if not properly controlled.

The Chemistry Control Program contains system specific acceptance criteria that are based on the guidance provided in EPRI PWR Primary Water Chemistry Guidelines, EPRI PWR Secondary Water Chemistry Guidelines, and EPRI Closed Cooling Water Chemistry Guideline.

18.2.5 Containment Inservice Inspection Plan - IWE

The Containment Inservice Inspection Plan – IWE was developed to implement applicable requirements of 10 CFR 50.55a. Section 50.55a(g)(4) requires that throughout the service life of nuclear power plants, components which are classified as either Class MC or Class CC pressure retaining components and their integral attachments must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME Code and Addenda that are incorporated by reference in §50.55a(b). Furthermore, §50.55a(g)(4)(v)(A) requires that metal containment pressure retaining components and their integral attachments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC. These requirements are subject to the limitation listed in paragraph (b)(2)(vi) and the modifications listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of §50.55a, to the extent practical within the limitations of design, geometry and materials of construction of the components [Reference 3].

18.2.6 Deleted Per 2005 Update

18.2.7 Crane Inspection Program

Scope – The scope of the Crane Inspection Program includes seismically restrained cranes.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The Crane Inspection Program inspects the crane rails and girders for loss of material.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending, the Crane Inspection Program will detect loss of material due to corrosion prior to loss of structure or component intended function.

Monitoring & Trending – The Crane Inspection Program detects aging effects through visual examination of the crane rails and girders. No actions are taken as part of this program to trend inspection or test results. Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:

- ANSI B30.2.0, "Overhead and Gantry Cranes," American National Standard, Section 2-2, Safety Standards for Cableways, Cranes, Derricks, Hoists, Hooks, Jacks and Slings, The American Society of Mechanical Engineers, New York.
- ANSI B30.16, Overhead Hoists (Underhung), The American Society of Mechanical Engineers, New York.
- 29 CFR Chapter XVII, 1910.179, Occupational Safety and Health Administration, Overhead and Gantry Cranes.

Acceptance Criteria – The acceptance criterion is no unacceptable visual indication of loss of material. The acceptance criterion is specified in the crane and hoist inspection procedures.

Corrective Actions & Confirmation Process – Structures and components that do not meet the acceptance criteria are evaluated by engineering for continued service and repaired as required. Structures and components which are deemed unacceptable are documented under the corrective action program. Specific corrective actions and confirmatory actions are implemented in accordance with the corrective action program.

Administrative Controls – The Crane Inspection Program is implemented by plant procedures and through the work management system using model work orders.

18.2.8 Fire Protection Program

Elements of the Fire Protection Program that serve to manage aging are implemented in accordance with Selected Licensee Commitments (See Table 18-1).

18.2.8.1 Sprinkler Branch Lines

The integrity of the sprinkler branch lines is assured by sprinkler flow tests performed by procedure every 18 months. Additionally, fouling of sprinkler branch lines that do not receive flow during periodic testing will be managed. Since these lines do not receive flow, it is believed that they are less susceptible to fouling than the lines that receive flow during testing. To validate this belief, branch lines of a few representative sprinkler systems will be volumetrically examined. Subsequent examinations for the period of extended operation will be determined based on the initial examination results. For McGuire,

this volumetric examination will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

Additionally, a sample of sprinklers are either inspected or replaced at 50 years of operation in accordance with NFPA 25.

18.2.8.2 Main Fire Pump Strainer

The Main Fire Pump Strainer Inspection will identify any loss of material of each main fire pump strainer. The raw water flow could result in loss of material. The acceptance criteria for the Main Fire Pump Strainer Inspection is no unacceptable loss of material that could result in a loss of component intended function(s) as determined by engineering. For McGuire, the initial Main Fire Pump Strainer Inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

18.2.8.3 Jockey Pump Strainer

The Jockey Pump Strainer Inspection will identify any loss of material of each jockey pump strainer basket. The raw water flow could result in loss of material. The acceptance criteria for the Jockey Pump Strainer Inspection is no unacceptable loss of material that could result in a loss of component intended function(s) as determined by engineering. For McGuire, the initial Jockey Pump Strainer Inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

18.2.8.4 Tank and Connected Piping

The purpose of the Tank and Connected Piping Internal Inspection is to manage loss of material of the internal surfaces of the carbon steel fire protection system pressure maintenance accumulator tank and connecting piping and valves supplying high-pressure air. The internal carbon steel surfaces of the tank are coated with an epoxy coating. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel tank. This preventive maintenance activity inspects the internal coating of the fire protection system pressure maintenance accumulator tank to check the condition of the coating to identify coating failures and the condition of the connecting piping supplying high-pressure air to identify loss of material. The Tank and Connected Piping Internal Inspection is a condition monitoring activity. The initial Tank and Connected Piping Internal Inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

18.2.8.5 Turbine Building Manual Hose Stations

For the period of extended operation associated with license renewal, all of the hose stations in the Turbine Building within the scope of license renewal will be periodically tested as follows: Every three (3) years, open each hose station valve partially to verify no flow blockage. For McGuire, the Turbine Building Manual Hose Station Flow Test will be implemented following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

18.2.9 Flood Barrier Inspection

The Flood Barrier Inspection manages cracking and change in material properties of the elastomeric flood seals to ensure that safety-related equipment is protected from floods and flooding flow paths such that no equipment safety-related intended functions or station safe shutdown capability are adversely impacted. This activity includes periodic visual inspections of the flood seals to identify degradation that could

result in loss of the intended function of the flood seals. The Flood Barrier Inspection is a condition monitoring program.

18.2.10 Flow Accelerated Corrosion Program

Scope – For license renewal, the Flow Accelerated Corrosion Program, which focuses inspections on piping, is credited for managing loss of material due to flow accelerated corrosion of carbon steel piping, valves, and cavitating venturies within the susceptible regions of the following systems:

- Auxiliary Steam
- Boron Recycle
- Feedwater
- Liquid Waste Recycle
- Liquid Waste Monitor and Disposal
- Turbine Exhaust

The only portions of Boron Recycle, Liquid Waste Recycle, and Liquid Waste Monitor and Disposal within the scope of license renewal that are susceptible to flow accelerated corrosion are supply lines from Auxiliary Steam.

Preventive Actions – Component replacement with a non-susceptible material is initiated as part of the Flow Accelerated Corrosion Program. Opportunities to replace components are evaluated when related modifications are being performed on a susceptible location or when economic benefit is realized.

Parameters Monitored or Inspected – Loss of material due to flow accelerated corrosion of carbon steel components is detected by inspection of susceptible component locations. The Flow Accelerated Corrosion Program inspections focus on piping. These inspections provide symptomatic evidence of loss of material due to flow accelerated corrosion of other components within the susceptible piping runs. Inspection methods include volumetric examinations using ultrasonic testing and radiography to measure component wall thickness. Visual examinations are also employed when access to interior surfaces is allowed by component design.

Detection of Aging Effects – In accordance with the information provided in Monitoring & Trending, the Flow Accelerated Corrosion Program will detect loss of material due to flow accelerated corrosion prior to loss of component intended function.

Monitoring & Trending – The program is consistent with the basic guidelines or recommendations provided by EPRI document NSAC-202L [Reference 5]. Component wall thickness is measured using volumetric examinations such as ultrasonic testing and radiography. Visual examinations are also employed when access to interior surfaces is allowed by component design. Component wall thickness acceptability is judged in accordance with the McGuire component design code of record.

Defined inspection locations exist in the several systems within the scope of license renewal. Auxiliary Steam, Boron Recycle,, Liquid Waste Recycle, and Liquid Waste Monitor and Disposal systems are all part of the same steam supply that spans these several systems. The steam is supplied from Auxiliary Steam and several inspection locations exist in this run of piping. The final system within the scope of license renewal falling within the scope of the Flow Accelerated Corrosion Program is Turbine Exhaust. The only in scope portion of Turbine Exhaust susceptible to flow accelerated corrosion is a few feet of ½" diameter piping. Because of the pipe size, ultrasonic scanning versus ultrasonic testing can be performed on this section of piping in lieu of establishing defined inspection locations.

Inspection frequency varies for each location, depending on previous inspection results, calculated rate of material loss, analytical model review, changes in operating or chemistry conditions, pertinent industry events, and plant operating experience. Inspection results are monitored and trended to determine the calculated rate of material loss, to detect changes in operating or chemistry conditions, and schedule for the next inspection.

Acceptance Criteria – Using the inspection results and including a safety margin, the projected component wall thickness at the time of the next plant outage must be greater than the allowable minimum wall thickness under the component design code of record.

Corrective Action & Confirmation Process – If the calculated component wall thickness at the time of the next outage is projected to be less than the allowable minimum wall thickness with safety margin under the component design code of record, then the component will be repaired or replaced prior to system start-up. The as-inspected component can also be justified for continued service through additional detailed engineering analysis.

Specific corrective actions are implemented in accordance with the Flow Accelerated Corrosion Program or the corrective action program. These programs apply to all components within the scope of the Flow Accelerated Corrosion Program.

Administrative Controls – Engineering Program Manuals for McGuire Units 1 and 2 and control the Flow Accelerated Corrosion Program.

18.2.11 Boric Acid Corrosion Control Program

Scope – The scope of the Boric Acid Corrosion Control Program includes electrical, mechanical, and structural components within the scope of license renewal that are located in the Auxiliary and Reactor Buildings where exposure to leaks from borated water systems is possible. Mechanical and structural components constructed of carbon steel, low alloy steel, and other susceptible materials are included within the scope of the program.

Preventive Actions – The programmatic implementation of the Boric Acid Corrosion Control Program is accomplished through visual surveillance and systematic trending of findings. Walkdowns of the Auxiliary and Reactor Buildings are conducted at the start of each refueling outage for the purpose of identifying leakage or evidence of leakage from borated water systems. All active leaks are monitored on an appropriate frequency depending on accessibility and rate of leakage. Furthermore, walkdowns of the Reactor building are conducted at the end of each outage prior to startup.

Parameters Monitored or Inspected – Systems, structures and components within the Auxiliary Building and Reactor Building are inspected for indications of leaks from systems containing borated water. Indications include, but are not limited to, the presence of boron crystals, pitting, and any other degradation beyond normal rust and surface discoloration that may indicate a loss of material.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending below, the Boric Acid Corrosion Control Program will detect boric acid intrusion and/or loss of material due to boric acid wastage prior to loss of structure or component intended function(s).

Monitoring & Trending – Walkdowns of the Auxiliary and Reactor Buildings are conducted at the start of each refueling outage for the purpose of identifying leakage or evidence of leakage from borated water systems. Information on leaks (e.g., equipment, system, leakage type and rate) is captured in the Fluid Leak Management Database to facilitate trending of leakage, if necessary. The Fluid Leak Management Database is periodically reviewed to identify adverse trends and opportunities to improve maintenance, engineering, and operational practices.

Acceptance Criteria – The external surfaces of structures and components within the scope of the Boric Acid Corrosion Control Program, including surroundings (e.g., insulation and floor areas), are expected to

be free from pitting and corrosion, abnormal discoloration or accumulated residues that may be evidence of leakage from proximate borated water systems.

Corrective Action & Confirmation Process – When the programmatic activities described in the Boric Acid Corrosion Control Program lead to detection of an unacceptable condition, the following corrective actions are required:

- Locate leak source and areas of general corrosion.
- Evaluate pressure-retaining components suffering loss of material for continued service, repair or replacement.
- Evaluate other affected components such as supports and other structural members for continued service, repair or replacement.

Specific corrective actions are implemented in accordance with the Boric Acid Corrosion Control Program or the corrective action program. These programs apply to all structures and components within the scope of the Boric Acid Corrosion Control Program.

Administrative Controls – Nuclear System Directive NSD 104, Materiel Condition/Housekeeping, Foreign Material Exclusion and Seismic Concerns [Reference 6] establishes high level expectations in the areas of Materiel Condition/Housekeeping, Foregin Material Exclusion and Seismic Concerns at Duke Energy's nuclear plants. The Fluid Leak Management Program is described and controlled by Nuclear System Directive NSD 413, Fluid Leak Management Program [Reference 7]. Guidance for the disposition of boric acid leakage is provided in PD-EG-PWR-1611, Boric Acid Corrosion Control Program (Program Description) and AD-EG-PWR-1611, Boric Acid Corrosion Control Program-Implementation (Administrative Procedure) [References 19 and 25].

18.2.12 Galvanic Susceptibility Inspection

Scope – The scope of the Galvanic Susceptibility Inspection includes galvanic couples exposed to gas, unmonitored treated water, and raw water environments in the following McGuire systems:

- Condenser Circulating Water
- Containment Ventilation Cooling Water
- Diesel Generator Room Sump Pump
- Exterior Fire Protection
- Interior Fire Protection
- Nuclear Service Water
- Waste Gas

The galvanic couples within these systems are carbon steel, cast iron, and ductile iron (anodes) coupled to copper alloys or stainless steel (cathodes) and copper alloys (anodes) coupled to stainless steel (cathode). In galvanic couples, the loss of material occurs in the anodes. Copper alloys are copper, brass, bronze, and copper-nickel.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameter inspected by the Galvanic Susceptibility Inspection is pipe wall thickness, as a measure of loss of material, of carbon steel-stainless steel couples exposed to raw water environments.

Detection of Aging Effects – The Galvanic Susceptibility Inspection is a one-time inspection that will detect the presence and extent of any loss of material due to galvanic corrosion.

Monitoring & Trending – The Galvanic Susceptibility Inspection will inspect a select set of carbon steel-stainless steel couples at McGuire using a volumetric examination technique. Visual examination will also be used should access to internal surfaces become available. The susceptibility and aggressiveness of galvanic corrosion is determined by the material position on the galvanic series and the corrosiveness of the surrounding environment. Since inspection of all couples is impractical, certain locations will be inspected where galvanic corrosion is more likely to occur. These more susceptible locations are where the materials are the farthest apart on the galvanic series surrounded by the most corrosive of the three environments identified above. For the couples noted above, carbon steel and stainless steel are the farthest apart on the galvanic series and raw water is the most corrosive environment. An inspection of selected locations of carbon steel-stainless steel couples in raw water will determine whether loss of material due to galvanic corrosion will be an aging effect of concern for the period of extended operation. A sentinel population of carbon steel-stainless steel couples located in raw water systems will be inspected. Engineering practice at Duke for the past several years has been to use stainless steel as a replacement material in raw water systems. Since engineering practice will continue to use stainless steel as an acceptable substitute material, the size of the sentinel population will be dependent on the number of susceptible locations at the time of the inspection. The results of this inspection will be applied to all galvanic couples in the systems listed in the Scope attribute above.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

No actions are taken as part of this activity to trend inspection results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The acceptance criterion for the Galvanic Susceptibility Inspection is no unacceptable loss of material that could result in a loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The Galvanic Susceptibility Inspection will be implemented in accordance with controlled plant procedures.

18.2.13 Heat Exchanger Activities

18.2.13.1 Component Cooling Heat Exchangers

The purpose of the Performance Testing Activities – Component Cooling Heat Exchangers is to manage fouling of admiralty brass and stainless steel heat exchanger tubes that are exposed to raw water. The Performance Testing Activities – Component Cooling Heat Exchangers is a performance monitoring

program that monitors specific component parameters to detect the presence of fouling which can affect the heat transfer function of the component.

The purpose of the Heat Exchanger Preventive Maintenance Activities – Component Cooling is to manage loss of material for parts of the component cooling heat exchanger exposed to raw water. The Heat Exchanger Preventive Maintenance Activities– Component Cooling is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary function. The program is credited with managing loss of material for admiralty brass, carbon steel, and stainless steel materials. Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions.

18.2.13.2 Containment Spray Heat Exchangers

The purpose of the Performance Testing Activities – Containment Spray Heat Exchangers is to manage fouling of stainless steel and titanium heat exchanger tubes that are exposed to raw water. The Performance Testing Activities – Containment Spray Heat Exchangers is a performance monitoring program that monitors specific component parameters to detect the presence of fouling, which can affect the heat transfer function of the component.

The purpose of the Heat Exchanger Preventive Maintenance Activities – Containment Spray is to manage loss of material for parts of the containment spray heat exchanger exposed to raw water. The Heat Exchanger Preventive Maintenance Activities – Containment Spray is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary function. The program is credited with managing loss of material for stainless steel and titanium materials. Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions.

18.2.13.3 Diesel Generator Engine Cooling Water Heat Exchangers

The purpose of the Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers is to manage fouling of copper and brass heat exchanger tubes that are exposed to raw water. The Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers is a performance monitoring program that monitors specific component parameters to detect the presence of fouling, which can affect the heat transfer function of the component.

The purpose of the Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water is to manage loss of material for parts of the diesel generator engine cooling water heat exchanger exposed to raw water. The Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary function. The program is credited with managing the subject aging effects for brass and copper heat exchanger tubes. Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions.

18.2.13.4 Control Area Chilled Water

The purpose of the Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water is to manage fouling and loss of material of parts of the control room area chillers exposed to raw water. The Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water is a condition monitoring program that monitors specific component parameters to detect the presence and assess the

extent of material loss that can affect the pressure boundary functions and periodically cleans the chiller tubes to manage fouling. The Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water is credited for managing loss of material or fouling for admiralty brass, carbon steel, copper-nickel alloy, and stainless steel materials. Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions.

18.2.13.5 Pump Motor Air Handling Units

The purpose of Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units is to manage loss of material and fouling of copper heat exchanger tubes that are exposed to raw water. The Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units is a new condition monitoring program that will detect the presence and assess the extent of material loss that can affect the pressure boundary function and will periodically clean the heat exchanger tubes to manage fouling. While fouling is managed currently by cleaning, this comprehensive program to manage both loss of material and fouling is a new plant program for license renewal. The scope of Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units is the tubes in the following McGuire heat exchangers of the Auxiliary Building Ventilation System:

- Containment Spray Pump Motor Air Handling Units
- Residual Heat Removal Pump Motor Air Handling Units
- Fuel Pool Cooling Pump Motor Air Handling Units

Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions. A destructive or non-destructive examination will be performed on one of the twelve cooling units within the scope of the program following issuance of renewed licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

18.2.13.6 Pump Oil Coolers

The purpose of Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers is to manage loss of material and fouling of copper-nickel heat exchanger tubes that are exposed to raw water. The Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers is a new condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary function and periodically cleans the heat exchanger tubes to manage fouling. While fouling is managed currently by periodic cleaning, this comprehensive program to manage both loss of material and fouling is a new plant program for license renewal. The scope of Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers is the tubes in the following McGuire heat exchangers of the Nuclear Service Water System:

- Centrifugal Charging Pump Bearing Oil Cooler
- Centrifugal Charging Pump Speed Reducer Oil Cooler
- Reciprocating Charging Pump Bearing Oil Cooler
- Reciprocating Charging Pump Fluid Drive Oil Cooler
- Safety Injection Pump Bearing Oil Cooler

Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions. A non-destructive examination will be performed on 100% of the tubes of one of the sixteen coolers within the scope of the

program following issuance of renewed licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

18.2.14 Ice Condenser Engineering Inspection

The Ice Condenser Engineering Inspection manages loss of material due to corrosion of the steel structural components in the ice condenser environment. The Ice Condenser Engineering Inspection includes periodic visual inspections of the ice condenser upper plenum, lower plenum, and top deck blankets to identify degradation that could impact the ability of the ice condenser to perform its intended function. The Ice Condenser Engineering Inspection is a condition monitoring program.

18.2.15 Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program

Scope – The scope of the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program includes inaccessible non-EQ medium-voltage cables within the scope of 10 CFR 54.4 that are exposed to significant voltage and to standing water (for any period of time).

Key Definitions and Assumptions: Inaccessible cables are those that are not able to be approached and viewed easily, such as in conduits or cable trenches; all others are accessible. A cable that has a portion of the cable routing that is inaccessible is an inaccessible cable. Non-EQ means not subject to 10 CFR 50.49 Environmental Qualification requirements. Medium-voltage cables are those applied at a system voltage greater than 2kV. Significant voltage is defined as exposure to system voltage for more than twenty-five percent of the time. Cables that are direct buried, run in horizontally-run buried conduit or run in outside cable trenches are assumed to be exposed to standing water.

Preventive Actions – Preventive actions are not included in the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program.

Parameters Monitored or Inspected – Medium-voltage cables within the scope of the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined before each test and will be a proven test for providing an indication of the condition of the conductor insulation related to aging effects caused by moisture and voltage stress. Each test performed for a cable may be a different type of test.

Detection of Aging Effects – Medium-voltage cables within the scope of the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program are tested at least once every 10 years. This is an adequate frequency to preclude failures of the conductor insulation.

Monitoring & Trending – Trending actions are not included in the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program.

The first test of the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

Acceptance Criteria – The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested.

Corrective Actions & Confirmation Process – Further investigation through the corrective action program is performed when the acceptance criteria are not met. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other medium-voltage cables within the scope of this program. Confirmatory actions, as needed, are implemented as part of the corrective action process.

Administrative Controls – The Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program is controlled by plant procedures.

18.2.16 Inservice Inspection Plan

The McGuire Inservice Inspection Plan, implements the requirements of 10 CFR 50.55a for Class 1, 2, and 3 components and Class 1, 2, 3, and MC component supports. The examinations are performed to the extent practicable within the limitations of design, geometry and materials of construction of the component. The period of extended operation for McGuire will contain the 5th and 6th ten-year inservice inspection intervals.

The Inservice Inspection Plan includes the following inspections and activities:

- ASME Section XI, Subsection IWB and IWC (secondary side of steam generators) Inspections
- ASME Section XI, Subsection IWF Inspections
- McGuire Unit 1 Cold Leg Elbow
- Small Bore Piping
- A VT-1 examination of the reactor vessel internals clevis insert fasteners will be performed in lieu of the VT-3 examination currently required by ASME Section XI.

18.2.16.1 McGuire Unit 1 Cold Leg Elbow

Reduction in fracture toughness due to thermal embrittlement can be an aging effect for certain types of cast austenitic stainless steel in locations where temperatures continuously exceed 482ºF. In a May 19, 2000 letter to NEI, Christopher I. Grimes, Chief License Renewal and Standardization Branch clarified that not all cast austenitic stainless steels are subject to thermal embrittlement [Reference 8]. The piping components and reactor coolant pumps fabricated from cast austenitic stainless steel were evaluated using the acceptance criteria set forth in the above letter. For those components requiring evaluation, only the McGuire 1, 27 ½-inch ID Loop B cold leg elbow exceeds the NRC-established threshold and is susceptible to thermal embrittlement which requires aging management for license renewal.

The McGuire Unit 1 27 ½-inch ID Loop B cold leg elbow is fabricated from SA-351 CF8, was statically cast, and contains no niobium. The elbow is the only piping item that exceeds the delta ferrite screening criterion, therefore, reduction of fracture toughness by thermal embrittlement is an aging effect requiring aging management for this elbow. The ferrite number is calculated at 22% using Hull's equivalent factors.

An augmented inspection with elements from Code Case N-481 will be used to manage reduction of fracture toughness by thermal embrittlement for the affected elbow during the period of extended operation. The inspection will be added to the Inservice Inspection Plan:

1. A VT-2 visual examinations will be performed each outage of the exterior of the affected elbow during the system leakage test.
2. A VT-1 visual examination will be performed of the external surfaces of the welded joints that connect the affected elbow to adjacent piping segments prior to entering the period of extended operation. VT-1 inspections of the welded joints will be repeated in the fifth and sixth inspection intervals.

A detailed evaluation to demonstrate the safety and serviceability of the elbow will be performed. This evaluation will be completed by June 12, 2021, the end of the initial license of McGuire Unit 1.

18.2.16.2 Small Bore Piping

Small bore piping is defined as piping less than 4-inch NPS. This piping does not receive volumetric inspection in accordance with ASME Section XI, 1989 Edition, Examination Category B-J or B-F. Cracking has been identified as an aging effect requiring programmatic management for Reactor Coolant System small bore piping for the period of extended operation.

A set of susceptible small bore piping locations will be volumetrically examined on each unit. Locations to be examined will be determined based on consideration of damage mechanisms. Damage mechanisms to be considered include fatigue, stress corrosion, and flow assisted corrosion/flow wastage. Cracking due to thermal fatigue resulting from stratification of fluids and turbulent penetration flow is an aging effect that will be addressed.

For McGuire, Small Bore Piping Examinations will be performed during each inservice inspection interval during the period of extended operation following issuance of renewed operating licenses for McGuire Nuclear Station.

18.2.17 Inspection Program For Civil Engineering Structures and Components

The Inspection Program for Civil Engineering Structures and Components is intended to meet the requirements of 10 CFR 50.65, Requirements for monitoring the effectiveness of maintenance at nuclear power plants (the Maintenance Rule). This program:

(1) monitors and assesses mechanical components, civil structures and components and their condition in order to provide reasonable assurance that they are capable of performing their intended functions in accordance with the current licensing basis;

(2) includes nuclear safety-related structures which enclose, support, or protect nuclear safety-related systems and components, non-safety related structures whose failure may prevent a nuclear safety-related system or component from fulfilling its intended function, and non safety-related structures which support equipment relied on during certain regulated events.

NEI 96-03, Industry Guideline for Monitoring the Condition of Structures at Nuclear Power Plants, has been used as guidance in the preparation of the Inspection Program for Civil Engineering Structures and Components. Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:

- NRC Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants
- ACI 349.3, Evaluation of Existing Nuclear Safety-Related Concrete Structures

Specific corrective actions are implemented in accordance with the Problem Investigation Process. The Problem Investigation Process applies to all structures and components within the scope of the Inspection Program for Civil Engineering Structures and Components.

18.2.18 Liquid Waste System Inspection

Scope – The scope of the Liquid Waste System Inspection is cast iron, stainless steel and carbon steel components exposed to unmonitored treated and borated water environments or raw water environments in the following McGuire systems:

- Component Cooling System – The WL Evaporator equipment is drained and out of service. Although the equipment is still in place it cannot be put back in service without extensive maintenance. Should the WL Evaporator equipment be placed back in service, the associated portions of the Component Cooling System will be subject to appropriate inspections.

- Liquid Waste Recycle System - stainless steel components exposed to an unmonitored borated water environment;

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameters inspected by the Liquid Waste System Inspection are wall thickness, as a measure of loss of material, and visible signs of cracking and loss of material.

Detection of Aging Effects – The Liquid Waste System Inspection will detect the presence and extent of loss of material due to crevice and pitting corrosion and cracking due to stress corrosion/intergranular attack in stainless steel components exposed to unmonitored borated and treated water environments.

In addition, this activity will detect the presence and extent of loss of material due to crevice, pitting, microbiologically influenced corrosion and cracking due to stress corrosion in stainless steel components exposed to raw water environments.

Finally, this activity will detect the presence and extent of loss of material due to crevice, general, pitting, and microbiologically influenced corrosion in carbon steel and cast iron components exposed to raw water environments.

Monitoring & Trending – The Liquid Waste System Inspection will use a volumetric technique to inspect the material/environment combinations located in each system listed above. As an alternative, visual examination will be used should access to internal surfaces become available. Selection of the specific areas for inspection for the system material/environment combinations will be the responsibility of the site's program owner.

Component Cooling System

The WL Evaporator equipment is drained and out of service. Therefore, the associated heat exchangers are beyond the scope of license renewal. Should the WL Evaporator equipment be placed back in service, three of the four heat exchangers would be within the scope of license renewal and one of the required heat exchangers would then be inspected. The inspection results would be applied to the other three stainless steel heat exchanger components exposed to unmonitored treated water environments.

Liquid Waste Recycle System

At McGuire, the Liquid Waste System Inspection will use a combination of volumetric and visual examination of a sample population of subject components. For stainless steel components exposed to unmonitored borated water environments, the sample population will include components located in stagnant or low flow areas near collection tanks where contaminants are likely to collect and concentrate to create an environment more corrosive than the general system borated water environments. The inspection results will be applied to the stainless steel components in the unmonitored borated water environments.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

No actions are taken as part of this activity to trend inspection results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The acceptance criterion for the Liquid Waste System Inspection is no unacceptable loss of material and cracking of stainless steel components and loss of material of carbon steel and cast iron components that could result in a loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, then no further action is required. If engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The Liquid Waste System Inspection will be implemented in accordance with controlled plant procedures.

18.2.19 Non-EQ Insulated Cables and Connections Aging Management Program

Scope – The scope of the Non-EQ Insulated Cables and Connections Aging Management Program includes accessible (able to be approached and viewed easily) non-EQ (not subject to 10 CFR 50.49 Environmental Qualification requirements) insulated electrical cables and connections (power, instrumentation and control applications) installed in the Reactor Buildings, Auxiliary Building and Turbine Building. The non-EQ insulated cables and connections within the scope of this program includes non-EQ cables used in low-level signal applications that are sensitive to reduction in insulation resistance such as radiation monitoring and nuclear instrumentation.

Preventive Actions – No actions are taken as part of the Non-EQ Insulated Cables and Connections Aging Management Program to prevent or mitigate aging degradation.

Parameters Monitored or Inspected – Accessible non-EQ insulated cables and connections installed in the Reactor Buildings, Auxiliary Building and Turbine Building are visually inspected per the Non-EQ Insulated Cables and Connections Aging Management Program for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination. Cable and connection jacket surface anomalies are precursor indications of conductor insulation aging degradation from heat or radiation in the presence of oxygen and may indicate the existence of an adverse localized equipment environment. An adverse localized equipment environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the insulated cable or connection.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending, the Non-EQ Insulated Cables and Connections Aging Management Program will detect aging effects for accessible non-EQ insulated cables and connections caused by heat and radiation prior to loss of intended function.

Monitoring & Trending – Accessible non-EQ insulated cables and connections installed in the Reactor Buildings, Auxiliary Building and Turbine Building are visually inspected per the Non-EQ Insulated Cables and Connections Aging Management Program at least once every 10 years. EPRI TR-109619, Guideline for the Management of Adverse Localized Equipment Environments [Reference 9], is used as guidance in performing the inspections.

Trending actions are not required as part of the Non-EQ Insulated Cables and Connections Aging Management Program.

For McGuire, the first inspection per the Non-EQ Insulated Cables and Connections Aging Management Program will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

Acceptance Criteria – The acceptance criterion for inspections performed per the Non-EQ Insulated Cables and Connections Aging Management Program is no unacceptable visual indications of cable and connection jacket surface anomalies that suggest conductor insulation degradation exists, as determined by engineering evaluation. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

Corrective Actions & Confirmation Process – Further investigation through the corrective action program is performed when the acceptance criteria are not met. When an adverse localized equipment environment is identified for a cable or connection, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections. Corrective actions may include, but are not limited to, testing, shielding or otherwise changing the environment, relocation or replacement of the affected cable or connection. Corrective actions should consider the potential for moisture in the area of degradation. Confirmatory actions, as needed, are implemented as part of the corrective action program.

Administrative Controls – The Non-EQ Insulated Cables and Connections Aging Management Program will be controlled by an engineering support program.

18.2.20 Pressurizer Spray Head Examination

Note: The Pressurizer Spray Head Examination affects both McGuire and Catawba and is being provided in each station's UFSAR to provide added assurance that both stations are aware of the commitment to perform the examination, initially at McGuire.

Scope – The scope of the Pressurizer Spray Head Examination is the internal spray heads of the McGuire and Catawba pressurizers.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored of Inspected – The parameter inspected by the Pressurizer Spray Head Examination is cracking of the pressurizer spray head due to thermal embrittlement.

Detection of Aging Effects – The Pressurizer Spray Head Examination is a one-time inspection that will detect the presence of cracking due to thermal embrittlement for the pressurizer spray heads.

Monitoring & Trending – The Pressurizer Spray Head Examination is a visual examination (VT-1) of the pressurizer spray head. No actions are taken as part of this program to trend inspection or test results.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 for McGuire Unit 1. Any required inspection of the Unit 2 pressurizer spray head will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by March 3, 2023 for McGuire Unit 2.

For Catawba, if necessary following the results of the McGuire Unit 1 examination, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station by December 6, 2024 for Catawba Unit 1 and by February 24, 2026 for Catawba Unit 2.

Acceptance Criteria – The acceptance criterion for Pressurizer Spray Head Examination will be in accordance with ASME Section XI, VT-1 examinations.

Corrective Action & Conformation Process – If the results of the inspection do not meet the specified acceptance criterion, then corrective actions will be taken such as replacing the affected spray heads. If cracks are detected in the initial spray head visual examination, then visual examinations will be conducted on the spray heads for McGuire Unit 2 and Catawba Units 1 and 2. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The Pressurizer Spray Head Examination will be implemented by plant procedures and the work management system.

18.2.21 Preventive Maintenance Activities

18.2.21.1 Condenser Circulating Water System Internal Coating Inspection

The Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection manages loss of material and cracking that could lead to loss of pressure boundary function. The program has two purposes for license renewal. The first purpose of this inspection is to manage loss of material of the internal surfaces of the large diameter intake and discharge piping in the Condenser Circulating Water System. The internal carbon steel surfaces of the large diameter intake and discharge piping in the Condenser Circulating Water System are coated to prevent the raw water environment from contacting the internal surfaces. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel intake and discharge piping. This inspection will periodically check the condition of the coating and look for coating degradation.

The second purpose of the Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection is to manage loss of material and cracking of the external surfaces of components in the underground environment by providing symptomatic evidence of the condition of the piping external surfaces. The external surfaces are coated with a coal tar epoxy that prevents the underground environment from contacting the external surfaces. Continued presence of an intact coating precludes loss of material and cracking of components whose external surfaces are exposed to the underground environment. Inspection of the internal surfaces will provide symptomatic evidence of the condition of the external surfaces of buried components.

18.2.21.2 Refueling Water Storage Tank Internal Coating Inspection

The purpose of the Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection is to manage loss of material of the internal surfaces of the carbon steel refueling water storage tanks. The internal carbon steel surfaces of the refueling water storage tank are coated with a phenolic epoxy paint that prevents borated water and air from contacting the internal surfaces. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel refueling water storage tank that could lead to loss of pressure boundary function. This preventive maintenance activity inspects the internal coating of the refueling water storage tanks to check the condition of the coating and to identify coating failures. The Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection is a condition monitoring program.

18.2.22 Reactor Vessel Integrity Program

Scope – The scope of the Reactor Vessel Integrity Program includes all reactor vessel beltline materials as defined by 10 CFR 50.61(a)(3).

Preventive Actions - No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The Reactor Vessel Integrity Program monitors reduction of fracture toughness of reactor vessel beltline materials by irradiation embrittlement.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending the Reactor Vessel Integrity Program will detect the effects of reduction of fracture toughness prior to loss of the reactor vessel intended functions.

Monitoring & Trending – Each reactor vessel had six specimen capsules located in guide baskets welded to the outside of the neutron shield pads and were positioned directly opposite the center portion of the core. McGuire Unit 1 capsules contain reactor vessel steel specimens oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting shell plate located in the core region. McGuire Unit 2 reactor vessel specimens are oriented both parallel and normal to the major working direction of the limiting core region shell forging. Associated weld metal and weld heat affected zone metal specimens are also included in each capsule. Capsule withdrawal schedules for the McGuire Units are provided in Table 5-33. The limiting weld material is not contained in a McGuire Unit 1 surveillance capsule, but is contained in a sister plant surveillance capsule and integrated into the McGuire Unit 1 surveillance program.

Surveillance capsule specimens are tested in accordance with approved industry standards. The test results from the encapsulated specimens represent the actual behavior of the material in the vessel. Data from testing of the surveillance capsule specimens are used to analyze Pressurized Thermal Shock, Upper Shelf Energy and to generate pressure-temperature curves for future operation of each unit. Additional information that is used to perform these analyses is as follows:

Fluence Received by the Specimens – Dosimeters such as Ni, Cu, Fe, Co-Al, shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are contained in the capsules. The dosimeters permit evaluation of the flux seen by the specimens. In addition, thermal monitors made of low melting point alloys are included to monitor the temperature of the specimens. A description of the methodology used to evaluate fluence received by the specimens using dosimetry measurements and fluence calculations, assuming the same neutron spectrum at the specimens and the vessel inner wall, is described in McGuire UFSAR, Sections 5.4.3.7.1 and 5.4.3.7.2 [Reference 10]. The correlations have indicated good agreement and form the bases for ensuring that the calculations of the integrated flux at the vessel wall are conservative WCAP-14040 [Reference 11]. Projections of neutron exposure at the vessel wall to end of life are based on the assumption that irradiation data from three previous fuel cycles are representative of all future fuel cycles.

Effective Full Power Years – The effective full power years of plant operation are based on reactor vessel incore power readings. The Operator Aid Computer collects incore instrument data and reactor engineers determine effective full power year values by comparing burnup to the thermal power to calculated burnup. This data is collected continuously for all four units.

Cavity Dosimetry –The cavity dosimetry provides a method for verification of fast neutron exposure distribution within the reactor vessel beltline region and establishes a mechanism to enable long term monitoring of neutron exposure once all of the capsules have been removed from the vessel.

Monitoring of Plant Changes – Actions will be taken to ensure that the capsule data tested during the current term of operation remains valid during the period of extended operation by monitoring changes to design and operation such as the neutron spectra relative to the conditions of existing capsule data or the reactor vessel inlet temperature. These types of changes will be assessed and the applicable analyses will be updated as necessary.

Acceptance Criteria – The acceptance criteria for the Reactor Vessel Integrity Program are:

- Charpy specimens removed from the surveillance capsules will be laboratory tested to ensure reactor vessel fracture toughness properties exhibit upper shelf energy greater than 50 ft-lbs.

- Calculations of reference temperature for pressurized thermal shock (RT_{PTS}) must be below the screening criteria of 270°F for plates, forgings, and longitudinal welds and 300°F for circumferential welds, respectively.
- Acceptable pressure-temperature curves for heatup and cooldown of the units must be maintained in Technical Specifications
- Capsules included in the Reactor Vessel Integrity Program must be withdrawn as scheduled.

Corrective Action & Confirmation Process – Specific corrective action and confirmation will be implemented as follows:

- If the Charpy upper-shelf energy drops below 50 ft-lbs, it must be demonstrated that margins of safety against fracture are equivalent to those of Appendix G of ASME Section XI.
- If the projected reference temperature exceeds the screening criteria, licensees are required to submit an analysis and/or schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, licensees shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed.
- If the pressure-temperature curves are not maintained current, actions are taken as required by Technical Specifications.
- If a capsule is not withdrawn as scheduled, the NRC will be notified and a revised withdrawal schedule will be updated and submitted to the NRC.

Administrative Controls – The administrative controls that apply to the Reactor Vessel Integrity Program are:

- Submittal of reports required by 10 CFR Part 50 Appendix H which include a capsule withdrawal schedule, a summary report of capsule withdrawal and test results within one year of capsule withdrawal and if needed a date when a Technical Specification change will be made to change pressure-temperature limits or procedures to meet pressure- temperature limits.
- RT_{PTS} analysis will be updated as required by 10 CFR 50.61.
- Pressure-Temperature curves are maintained in the plant Technical Specifications.
- As surveillance capsules are withdrawn and either tested or stored, documentation will be updated accordingly and submitted to the NRC in accordance with 10 CFR 50, Appendix G.

18.2.23 Reactor Vessel Internals Inspection

Note: The Reactor Vessel Internals Inspection affects both McGuire and Catawba and is being provided in each station's UFSAR to provide added assurance that both stations are aware of the commitment to perform the examination, initially at McGuire.

Scope – The scope of the Reactor Vessel Internals Inspection consists of the reactor vessel internals stainless steel items that may be separated into three groups – (1) items comprised of plates, forgings, and welds, (2) bolting (baffle-to-baffle, baffle-to-former, and barrel-to-former), and (3) items fabricated from cast austenitic stainless steel (CASS).

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The Reactor Vessel Internals Inspection monitors the following parameters:

Visual inspections will be performed for items comprised of plates, forgings, and welds to detect cracking which could be initiated by irradiation assisted stress corrosion, enhanced by reduction of fracture toughness due to irradiation embrittlement.

Volumetric inspections will be performed for bolting to detect cracking due to irradiation assisted stress corrosion enhanced by reduction of fracture toughness due to irradiation embrittlement, and loss of preload by stress relaxation due to irradiation creep.

For items fabricated from CASS, crack propagation of existing flaws caused by reduction of fracture toughness by thermal embrittlement and irradiation embrittlement will be monitored.

Dimensional changes due to void swelling will be monitored in lead components for items comprised of plates, forgings, welds, and bolting.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending**,** the Reactor Vessel Internals Inspection will detect cracking, reduction of fracture toughness, dimensional changes, and loss of preload prior to loss of the reactor vessel internals intended function(s).

Monitoring & Trending – The Reactor Vessel Internals Inspection includes the following inspection activities:

For plates, forgings, and welds, a visual inspection will be performed to detect the effects of cracking by irradiation assisted stress corrosion cracking enhanced by reduction of fracture toughness by irradiation embrittlement. The visual inspection method selected for the inspection of RV internal plates, forging, and welds will be sufficient to detect cracks in the components prior to any growth to a size that is greater than the critical crack size (critical crack length) for the material.

For baffle bolts, a volumetric inspection will be performed at McGuire Unit 1 to assess cracking.

For items fabricated from CASS, an analytical approach to assess the effect of reduction of fracture toughness on the applicable reactor vessel internals items will be performed. The specific inspection method will depend on the results of these analyses.

McGuire Unit 1 will be inspected in the fifth inservice inspection interval. McGuire Unit 2 will be inspected early in the sixth inservice interval (prior to the last year of the 20-year period of extended operation).

With respect to dimensional changes due to void swelling, McGuire will rely on the results of inspections to be performed at Oconee. Items comprised of plates, forgings, and welds will be inspected at all three Oconee Units to assess the effects of void swelling. Activities are in progress to develop and qualify the inspection method. The results of the Oconee inspections will be used to determine if change in dimensions due to void swelling is a concern for the reactor vessel internals of McGuire Unit 1 and McGuire Unit 2, and if additional inspections are necessary.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The Reactor Vessel Internals Inspection includes the following acceptance criteria:

For the items comprised of plates, forgings, and welds, critical crack size will be determined by analysis and submitted for review and approval to the NRC staff prior to the inspection.

For baffle bolts, any detectable crack indication is unacceptable for a particular baffle bolt. The number of baffle bolts needed to be intact and their locations will be determined by analysis.

For items fabricated from CASS, critical crack size will be determined by analysis. Acceptance criteria for all aging effects will be developed and submitted for review and approval to the NRC staff prior to the inspection.

For items subject to dimensional changes due to void swelling, activities are in progress to develop and qualify the inspection method. Acceptance criteria will be developed and submitted for review and approval to the NRC staff prior to the inspection.

Corrective Action & Confirmation Process – If the results of the inspection are not acceptable, then actions will be taken to repair or replace the affected items or to determine by analysis the acceptability of the items. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The Reactor Vessel Internals Inspection will be implemented by plant procedures and the work management system.

18.2.24 Selective Leaching Inspection

Scope – The scope of the Selective Leaching Inspection is the brass and cast iron components exposed to raw water in the following McGuire systems:

- Conventional Wastewater Treatment
- Diesel Generator Room Sump Pump
- Exterior Fire Protection
- Groundwater Drainage
- Interior Fire Protection

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameter inspected by the Selective Leaching Inspection is the hardness of the wetted surface of cast iron pump casings and brass valve bodies. Selective leaching (a form of galvanic corrosion) is the dissolution of one metal in an alloy at the metal surface which leaves a weakened network of corrosion products that is revealed by a Brinnell Hardness check or equivalent as reduction in material hardness.

Detection of Aging Effects – The Selective Leaching Inspection is a one-time inspection that will detect the presence and extent of any loss of material due to selective leaching.

Monitoring & Trending – Of the cast iron components in the systems above, the Selective Leaching Inspection will perform a Brinnell Hardness Test or equivalent test on one cast iron pump casing in the Exterior Fire Protection System at McGuire. The Brinnell Hardness Test or equivalent test is most easily performed on a pump casing and will be indicative of all cast iron components in the systems listed above. The Exterior Fire Protection System contains a raw water environment that is susceptible to selective leaching and will be bounding for the other environments in the other systems. If no parameters are known that would distinguish among the pump casings, one of the three cast iron pump casings in the Exterior Fire Protection System at McGuire will be examined based on accessibility and operational concerns. The results of this inspection will be applied to the other cast iron components exposed to raw water environments in the systems listed above.

The Selective Leaching Inspection will also perform a Brinnell Hardness Test or equivalent test on a sample of brass valves at McGuire in the Interior Fire Protection System. Valves selected for inspection should be continuously exposed to stagnant or low flow raw water environments. If no parameters are

known that would distinguish the susceptible locations at McGuire, a select set of susceptible locations will be examined based on accessibility, operational, and radiological concerns. The results of this inspection will be applied to the brass components exposed to raw water environments in the systems listed above.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

No actions are taken as part of this program to trend inspection results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The acceptance criteria for the Selective Leaching Inspection is no unacceptable loss of material due to selective leaching that could result in a loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effect will not cause a loss of the component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the applicable aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The Selective Leaching Inspection will be implemented in accordance with controlled plant procedures.

18.2.25 Service Water Piping Corrosion Program

Scope – For license renewal, the Service Water Piping Corrosion Program is credited with managing loss of material for components in the following systems:

- Containment Ventilation Cooling Water
- Exterior Fire Protection
- Condenser Circulating Water
- Interior Fire Protection
- Nuclear Service Water

Additionally, the Service Water Piping Corrosion Program is credited with managing loss of material for heat exchanger sub-components in the following systems:

- Containment Spray
- Diesel Generator Cooling Water
- Control Area Chilled Water

Preventive Actions – No actions are taken as part of the Service Water Piping Corrosion Program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The Service Water Piping Corrosion Program inspections are focused on carbon steel piping components exposed to raw water. Among the installed component materials, carbon steel is the more susceptible to general loss of material and serves as a leading indicator of the general material condition of the system components. Inspection of carbon steel piping provides symptomatic evidence of loss of material of other components and other materials exposed to raw water. The specific parameter monitored is pipe wall thickness as an indicator of loss of material.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending below, the Service Water Piping Corrosion Program will detect the more uniform loss of material such as that due to general corrosion as well as particulate erosion that may occur in areas of higher flow velocity. The program will also detect loss of material due to localized corrosion due to crevice, pitting, and microbiologically-influenced corrosion (MIC).

Monitoring & Trending – The Service Water Piping Corrosion Program manages all of the system components within license renewal that are susceptible to the various corrosion mechanisms and is not focused on individual components within each specific system. The intent of the Service Water Piping Corrosion Program is to inspect a number of locations with conditions that are characteristic of the conditions found throughout the raw water systems above. The results of these inspection locations would then be applied to similar locations throughout all the raw water systems within the scope of license renewal. This characteristic-based approach recognizes the commonality among the component materials of construction and the environment to which they are exposed. Inspection results are used to determine and expand, as necessary, the number of inspection locations in a given characteristic set.

Monitoring under the Service Water Piping Corrosion Program focuses on carbon steel pipe. For components constructed of cast and ductile iron, galvanized steel and copper alloys, experience has shown that loss of material for these components will occur at a rate somewhat less than the carbon steel pipe. Therefore, the results of the carbon steel pipe inspections will provide a leading indicator of the condition of these materials.

For the carbon and galvanized steel, cast and ductile iron, and copper alloy component materials that can experience loss of material from both uniform and localized mechanisms, it is the gross material loss due to uniform mechanisms that is of primary concern under the Service Water Piping Corrosion Program. Gross wall loss can lead to structural instability concerns and could directly impact component intended function. Monitoring for degradation, including general and localized corrosion, is accomplished using ultrasonic test techniques. Monitoring for general and localized corrosion is supplemented by visual inspections of the inside of the piping if access to the interior surfaces is allowed such as during plant modifications. Monitoring of localized corrosion is additionally supplemented by exterior piping inspections that reveal pinhole leaks caused by localized corrosion. Additional detail concerning exterior piping inspections is provided below.

When pipe wall thickness is determined by volumetric wall thickness measurements using ultrasonic testing, several measurements are taken around the circumference of the piping. These measurements are then assessed in relation to the specific acceptance criteria for that location. Because the phenomena is slow-acting, inspection frequency varies for each location. The frequency of re-inspection depends on previous inspection results, calculated rate of material loss, piping analysis review, pertinent industry events, and plant operating experience. Refer to Acceptance Criteria for additional details. Component results are catalogued, and future inspection or component replacement schedules are determined as a part of the program.

Supplemental visual inspection detect localized corrosion due to pitting and microbiologically-influenced corrosion (MIC) that reveals itself through pinhole leaks in the piping components. The geometry of the pinholes means that they are not a structural integrity concern. Further, these pinhole leaks cannot individually lead to loss of the component intended function, since sufficient flow at prescribed pressures can still be provided by the system. These localized concerns will lead to structural integrity concerns

only when a significant number of pinholes are present. When indications of a pinhole are found, volumetric wall thickness measurements are taken in the area. A trend of indications of through-wall leaks due to pitting corrosion or MIC provides evidence when localized corrosion may become a structural integrity concern and will trigger corrective actions by the Service Water Piping Corrosion Program. Methods in place to identify incidents of through-wall leaks are system walkdowns, operator rounds, system testing, and maintenance activities.

While the emphasis of the Service Water Piping Corrosion Program remains on potential areas of severe degradation, including general and localized corrosion, the management of loss of material due to localized corrosion of component materials exposed to raw water is supplemented by the monitoring and trending of relevant plant operating experience of non-structural, through-wall leaks identified during various plant activities.

Acceptance Criteria – The Service Water Piping Corrosion Program manages loss of material for nuclear safety related and non-nuclear safety related components.

For nuclear safety-related components designed to ASME Section III, Class 3 rules, acceptance criteria are defined as meeting ASME code requirements [Reference 12] in order to assure structural integrity. Several factors are used to determine structural integrity at an inspection location. These factors include consideration of actual as-found wall thickness, calculated rate of material loss, use of the piping stress analyses to determine a minimum required thickness and projected time to reach the minimum wall thickness which, in turn, will establish the re-inspection interval or component replacement schedule.

For the non-nuclear safety related components that have no seismic design requirements, the acceptance criterion is the minimum wall thickness calculated on a location-specific basis. These minimum values have been determined based on design pressure or structural loading using the piping design code of record and then applying additional conservatism.

Corrective Action & Confirmation Process – Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The Service Water Piping Corrosion Program is governed by site specifications and implemented using controlled plant procedures and work orders. The procedures and work processes provide steps for performance of the activities and require the documentation of the results.

18.2.26 Sump Pump Systems Inspection

Scope – The scope of the Sump Pump Systems Inspection is a limited set of mechanical components constructed of carbon steel, cast iron, and stainless steel exposed to sump environments in the following McGuire systems:

- Diesel Generator Room Sump Pump System
- Conventional Waste Water Treatment System
- Groundwater Drainage System

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameter inspected by the Sump Pump Systems Inspection is wall thickness as a measure of loss of material.

Detection of Aging Effects – The Sump Pump Systems Inspection is a one-time inspection that will detect the presence and extent of loss of material due to crevice, general, pitting, and microbiologically influenced corrosion.

Monitoring & Trending – The Sump Pump Systems Inspection will inspect sump components at McGuire located within the Diesel Generator Room Sump Pump System using a volumetric examination technique. The Diesel Generator Room Sump Pump System was selected for inspection because the system contains a representation of all of the materials present within the other sump environments. The sump environment in the Diesel Generator Room Sump Pump System is a potential combination of leakage of raw water, fuel oil, and treated water. Inspection of the Diesel Generator Room Sump Pump System will provide a representative review of the condition of mechanical component materials subject to a sump environment.

Inspection locations will be at piping low points, pump casings, and valve bodies where materials are continuously wetted by the raw water environment or subject to alternate wetting and drying. The results of this inspection will be applied to the mechanical components in the Conventional Waste Water Treatment and Groundwater Drainage.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

No actions are taken as part of this activity to trend inspection or test results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

The Groundwater Drainage System contains raw water that is considered to be relatively pure and not subject to mixing with treated water or contaminants from other plant systems. This environment is considered to be less severe than the other sump pump environments. Additionally, the system contains a limited selection of materials within the system boundaries at McGuire. Therefore, the results of the Sump Pump Systems Inspection are encompassing and will be applied to the Groundwater Drainage System components subject to a raw water environment.

Acceptance Criteria – The acceptance criteria for the Sump Pump Systems Inspection is no unacceptable loss of material that could result in the loss of the component intended function(s), as determined by engineering evaluation.

Corrective Action & Confirmation Process – If the engineering evaluation determines that continuation of the aging effect will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The Sump Pump Systems Inspection will be implemented in accordance with controlled plant procedures.

18.2.27 Treated Water Systems Stainless Steel Inspection

Scope – The scope of Treated Water Systems Stainless Steel Inspection is stainless steel components exposed to unmonitored treated water environments in the following McGuire system:

- Nuclear Solid Waste Disposal

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameters inspected by the Treated Water Systems Stainless Steel Inspection are pipe wall thickness, as an indicator of loss of material, and evidence of cracking.

Detection of Aging Effects – The Treated Water Systems Stainless Steel Inspection is a one-time inspection that will detect the presence and extent of any loss of material or cracking of stainless steel components exposed to unmonitored treated water environments.

Monitoring & Trending – The Treated Water Systems Stainless Steel Inspection at McGuire will inspect stainless steel components, welds, and heat affected zones, as applicable, in the McGuire Nuclear Solid Waste Disposal System. The McGuire Nuclear Solid Waste Disposal System components within the scope of license renewal is a mixture of unmonitored treated water and spent resins sluiced from demineralizers in various systems. The environment is expected to contain contaminants in excess of the limits below which a concern would not exist for cracking and loss of material in stainless steel. A concentration of any contaminants present would occur in areas of low flow or stagnant conditions. As a result, inspections will be performed in stagnant and low flow lines around the spent resin storage tanks using volumetric techniques. In addition to the volumetric examination, a visual examination of the interior of a valve will be conducted to determine the presence of pitting corrosion.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

No actions are taken as part of this activity to trend inspection results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The acceptance criterion for the Treated Water Systems Stainless Steel Inspection is no unacceptable loss of material or cracking that could result in the loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The Treated Water Systems Stainless Steel Inspection will be implemented in accordance with controlled plant procedures.

18.2.28 Underwater Inspection of Nuclear Service Water Structures

Scope – The scope of the Underwater Inspection of Nuclear Service Water Structures includes the following structures:

Standby Nuclear Service Water Discharge Structures

Standby Nuclear Service Water Intake Structure

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The Underwater Inspection of Nuclear Service Water Structures requires examination of the structure for the following parameters: loss of material of steel components and loss of material and cracking of concrete components.

Detection of Aging Effects – In accordance with information provided in Monitoring & Trending, the Underwater Inspection of Nuclear Service Water Structures will detect loss of material of steel components and loss of material and cracking of concrete components prior to loss of structure or component intended functions.

Monitoring & Trending – The Underwater Inspection of Nuclear Service Water Structures detects aging effects through visual examination. The inspection is performed every five years at McGuire. No actions are taken as part of this program to trend inspection or test results. Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:

- NRC Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants
- ACI 349.3, Evaluation of Existing Nuclear Safety-Related Concrete Structures
- ACI 201, Guide for Making a Condition Survey of Concrete in Service

Acceptance Criteria – The acceptance criteria are no unacceptable visual indication of (1) loss of material for steel components and (2) loss of material and cracking for concrete components, as determined by the accountable engineer. The qualifications of the accountable engineer are in accordance with the guidance provided in NRC Regulatory Guide 1.127.

Corrective Action & Confirmation Process – Structures and components which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repair, as required. Structures and components which are deemed unacceptable are documented under the corrective action program. Specific corrective actions and confirmatory actions, as needed, are implemented in accordance with the corrective action program. All prior inspection reports are reviewed to ensure implementation of recommended corrective actions.

Administrative Controls – The Underwater Inspection of Nuclear Service Water Structures is implemented by plant work management system using model work orders.

18.2.29 Ventilation Area Pressure Boundary Sealants Inspection

Scope – The scope of the Ventilation Area Pressure Boundary Sealants Inspection is the pressure boundary structural sealants installed in the ventilation pressure boundary of the Control Room, ECCS Pump Room, Annulus, and Fuel Handling areas. Pressure boundary structural sealants include, but are not limited to, sealants in the interface between a structural wall, floor or ceiling and a non-structural component such as duct, piping, electrical cables, doors, and non-structural walls.

Preventive Actions – No actions are taken as a part of this one-time inspection to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – Ventilation Area Pressure Boundary Sealants Inspection is a visual inspection for cracking or shrinkage of the structural sealants.

Detection of Aging Effects – In accordance with the information provided in Monitoring & Trending, Ventilation Area Pressure Boundary Sealants Inspection will detect cracking or shrinkage of the ventilation area pressure boundary structural sealants.

Monitoring & Trending – The Ventilation Area Pressure Boundary Sealants Inspection will visually inspect a representative sample of structural sealants at each station. Locations of inspections will be based on severity of the local ambient conditions taking into consideration temperature and radiation. The sample locations selected will provide a leading indication of the condition of all structural sealants within the scope of this activity.

No actions are taken as part of this program to trend inspection results.

For McGuire, this one-time inspection will be completed following issuance of the renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1)

Acceptance Criteria – The acceptance criterion for the Ventilation Area Pressure Boundary Sealants Inspection is no unacceptable cracking or shrinking that could result in the loss of the intended function of the structural sealant as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of structural sealant intended function, under any current licensing basis design condition for the period of extended operation, no further action is required. If the engineering evaluation determines that continuation of the aging effects could cause a loss of structural sealant function under current licensing design conditions for the period of extended operation, then programmatic oversight will be defined by engineering. Specific corrective actions, including repair or replacement of the ventilation area pressure boundary structural sealants, will be implemented in accordance with the corrective action program.

Administrative Controls – Ventilation Area Pressure Boundary Sealants Inspection surveillances will be implemented by written procedure.

18.2.30 Waste Gas System Inspection

Scope –The scope of the Waste Gas System Inspection is carbon steel and stainless steel materials that are exposed to unmonitored treated water environments and carbon steel materials that are exposed to gas environments within the license renewal boundaries of the McGuire Waste Gas Systems.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The parameters monitored or inspected by the Waste Gas System Inspection are wall thickness, as a measure of loss of material, and evidence of cracking.

Detection of Aging Effects – The Waste Gas System Inspection is a one-time inspection that will detect the presence and extent of any loss of material due to general, crevice, or pitting corrosion or cracking due to stress corrosion in brass, carbon steel, and stainless steel materials subject to an unmonitored treated water environment. The Waste Gas System Inspection will also detect the presence and extent of any loss of material due to general corrosion in carbon steel materials subject to a gas environment.

Monitoring & Trending – The Waste Gas System Inspection will use a volumetric technique to inspect three sets of material/environment combinations. As an alternative, visual examination will be used should access to internal surfaces become available. The Waste Gas System is primarily a gas environment with unmonitored treated water environments from condensation of entrained water vapor and effluent from the recombiners and separators. Specific component/environment inspection combinations will include carbon steel, and stainless steel components exposed to an unmonitored treated water environment. Also, carbon steel components exposed to a gas environment will be inspected. Selection of the specific areas for inspection for the above material/environment combinations will be the responsibility of the site program owner.

(1) For carbon steel components exposed to unmonitored treated water environments at McGuire, inspections will be performed on the lower portions of decay tanks and associated drain lines where condensate is likely to accumulate. One of eight possible locations at McGuire will be examined. If no parameters are known that would distinguish the susceptible locations at McGuire, one of the eight available at McGuire will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the remainder of the Waste Gas System carbon steel components within the scope of license renewal exposed to unmonitored treated water environment.

(2) For stainless steel components exposed to unmonitored treated water environments at McGuire, inspections will be performed on the seal water path of the waste gas compressor. One of two possible locations at McGuire will be examined. If no parameters are known that would distinguish the susceptible locations at McGuire, one of the two available at McGuire will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the remainder of the Waste Gas System stainless steel components within the scope of license renewal exposed to unmonitored treated water environment.

(3) For the carbon steel components exposed to a gas environment at McGuire, an inspection will be performed on components within the scope of license renewal located between the volume control tanks and the waste gas compressor phase separators. If no parameters are known that would distinguish the most susceptible locations at McGuire, one location at McGuire will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the remainder of the Waste Gas System carbon steel components within the scope of license renewal exposed to gas environments.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

No actions are taken as part of this activity to trend inspection or test results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

The Waste Gas System is primarily a gas environment composed of nitrogen, hydrogen, oxygen, and fission product gases. The section of the Waste Gas System between the volume control tanks and the waste gas compressors phase separators will contain a warm, moist gas that could result in the cooler internal surfaces of the carbon steel components being wet due to condensation. As a result, corrosion of the carbon steel surfaces is more likely due to the presence of moisture and would serve as a leading indicator for the remainder of the carbon steel components within the scope of license renewal exposed to the gas environment in the Waste Gas System. Therefore, the results of the inspection can be applied to the remainder of the carbon steel components exposed to gas environments.

Acceptance Criteria – The acceptance criteria for the Waste Gas System Inspection is no unacceptable loss of material or cracking that could result in a loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the applicable aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight is required to be defined by

engineering. Specific corrective actions will be implemented in accordance with the Corrective Action Program.

Administrative Controls – The Waste Gas System Inspection will be implemented in accordance with controlled plant procedures.

18.2.31 References

1. M. S. Tuckman (Duke) letter dated July 30, 1991, NRC Bulletin 88-09, Thimble Tube Thinning in Westinghouse Reactors, McGuire Nuclear Station, Docket Nos. 50-369 and 50-370; Catawba Nuclear Station, Docket Nos. 50-413 and 50-414.
2. WCAP-12866, Bottom Mounted Instrumentation Flux Thimble Wear, January 1991.
3. 10 CFR Part 50, §50.55a, Codes and Standards.
4. W. T. Russell (NRC) letter dated November 19,1993 to William Rasin, (NUMARC), Safety Evaluation for Potential Reactor Vessel Head Adapter Tube Cracking.
5. EPRI NSAC-202L-R2, Recommendations for an Effective Flow Accelerated Corrosion Program, Revision 2, April 1999.
6. Nuclear System Directive 104, Materiel/Condition/Housekeeping, Foreign Material Exclusion and Seismic Concerns, Revision 33.
7. Nuclear System Directive 413, Fluid Leak Management Program, Revision 9.
8. C. I. Grimes (NRC) letter dated May 19, 2000 to D. J. Walters (NEI), License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," Project No. 690.
9. Guideline for the Management of Adverse Localized Equipment Environments, EPRI, Palo Alto, CA: 1999. EPRI TR-109619.
10. McGuire Nuclear Station Updated Final Safety Analysis Report, as revised.
11. WCAP-14040, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, June 1994.
12. ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components, Subsection ND Class 3 Components, 1971 edition.
13. NRC Bulletin 2003-02, "Leakage from Reactor Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Inegrity," August 21, 2003.
14. NRC Order EA-03-009, "Issuance of first revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," February 20, 2004.
15. NRC Bulletin 2004-01, "Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at PWRS," May 28, 2004.
16. Barron, Henry B. (Duke) to U. S. Nuclear Regulatory Commission, Duke Response to NRC Bulletin 2004-01, July 27, 2004.
17. McCollum, William R. (Duke) to U. S. Nuclear Regulatory Commission, Supplement to Response to NRC Bulletin 2004-01, September 21, 2004.

18. Barret, R. (NRC) to Marion, A. (NEI), Flaw Evaluation Guidelines, April 11, 2003.
19. PD-EG-PWR-1611, Boric Acid Corrosion Control Program (Program Description), Rev 0.
20. ASME Code Case N-729-1, ASME Boiler and Pressure Vessel Code. Section XI, Division 1. Case N-721-1. "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds." March 28, 2006.
21. ASME Code Case N-722-1, Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials.
22. Federal Register, 10 CFR Part 50, Industry Codes & Standards; Amended Requirements; Final Rule; September 10, 2008 - Pages 52742 and 52749.
23. ASME Code Case N-770-1, Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities.
24. Federal Register, 10 CFR Part 50, American Society of Mechanical Engineers (ASME) Codes and New and Revised ASME Code Cases; Final Rule June 21, 2011.
25. AD-EG-PWR-1611, Boric Acid Corrosion Control Program-Implementation (Administrative Procedure), Rev. 0

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18.3 Additional Commitments

18.3.1 Battery Rack Inspections

Battery rack inspections conducted in accordance with ITS SR 3.8.4.3, SLC 16.8.3.3, SLC 16.9.7.12, and SLC 16.9.7.17 shall include the structural supports and anchorages.

18.3.2 Steam Generator Surveillance Program

The inspections of the Steam Generator Surveillance Program follow the requirements of Technical Specification 5.5.9 "Steam Generator (SG) Program".

18.3.3 Additional Chemistry Commitment – Visual Inspection of Auxiliary Feedwater and Main Feedwater Piping

Visual inspections of the interior surfaces of Auxiliary Feedwater System and Main Feedwater System components and piping will be performed when available. The inspection results will be documented in writing and available for inspection following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1.)

18.3.4 Fuse Holder Program

For McGuire, Duke commits to implement the final version of the fuse holder interim staff guidance (initially provided to NEI by NRC letter dated May 16, 2002 and when finalized by the staff) by June 12, 2021 (the end of the initial license of McGuire Unit 1).

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