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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. The third column Table 18-1 identifies where the description of the Program, Activity, or TLAA is located in either the Catawba UFSAR or in the Catawba 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 Catawba Nuclear Station.

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

The 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. NRC Safety Evaluation Report (SER) Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, NUREG-1772 dated 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 Catawba 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 Program 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 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 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 communication regarding Alloy 600.

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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 operating experience. The susceptibility ranking is documented 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 attachment 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 Catawba, 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 Catawba Nuclear Station and prior to December 6, 2024 (the end of the initial license of Catawba Unit 1).

18.2.1.2 Control Rod Drive Mechanism 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 Mechanism 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 Catawba 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 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.

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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 the 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 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 Boric Acid Corrosion Control Program.

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 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.

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).

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

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

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 Order EA-03-99 (Reference 13) no longer applies and is deemed to be withdrawn.

Licensees of existing, operating pressurized-water reactors as of 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 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 14, 15 and 16).

Preventive Actions – "Mitigation of Alloy 600 Pressurizer Nozzle Weldments"

NRC Bulletin 2004-01, entitled *Inspection of Alloy 82/182 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized Water Reactors*, requested site specific information related to the Alloy 600 material issues on the pressurizer. Six nozzles [(1) PORV, (3) safety valve, (1) spray, and (1) surge] were identified with Alloy 82/182 buttering and weld filler material between the nozzle and safe-ends as susceptible to PWSCC in the Catawba response to the bulletin. A broader industry response to Alloy 600 material issues was developed under a program established by NEI 03-08, *Guideline for the Management of Material Issues*. Under the implementation protocol of NEI 03-08, certain elements of the document MRP 139, entitled *Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline* were categorized as mandatory. These mandatory elements included examination requirements and schedules for Alloy 600 welds that included the PWSCC susceptible pressurizer welds. The geometrical configuration of the pressurizer nozzle welds precluded a qualified volumetric examination of the welds specified by MRP 139. In lieu of modifying the surface profile to allow the qualified volumetric examination, a mitigative strategy was chosen to perform an Alloy 52/152 full structural weld overlay over the existing weld using ASME Code Cases 504-2 and 638-1. For each nozzle location, the overlay was qualified in accordance with ASME XI, IWB rules for primary membrane plus bending stresses assuming a 100% through wall crack of the original weld thickness. Furthermore, the overlay was evaluated for PWSCC and fatigue crack growth. As multiple weld passes are applied on the exterior surface of the piping, residual compressive stresses in both the hoop and axial stress directions are developed on the internal surface. The qualification of the overlay indicates that the beneficial effects of these compressive stresses limit the fatigue and PWSCC crack growth to the original weld material plus the non-credited dilution weld layer of the overlay. Thus, there is no predicted crack growth in the qualified weld overlay thickness. This mitigative strategy serves to increase the margin to failure of the reactor coolant pressure boundary, increase equipment reliability, minimize the possibility of an emergent repair and preclude more frequent inspections.

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:

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1. A bare metal visual inspection of the gap between the manway cover and pressurizer manway for evidence of manway diaphragm plate seal weld leakage.

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 15).

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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 Catawba 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 Catawba 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 Catawba will be inspected using a volumetric technique. If no parameters are known that would distinguish the susceptible locations, one of the twelve available at Catawba 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 Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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 2006 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 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 Action & 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

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 Catawba, this volumetric examination will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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 Catawba, the initial *Main Fire Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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 Catawba, the initial *Jockey Pump Strainer Inspection* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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 the filtered water tanks and connecting aluminum piping and valves. The internal carbon steel surfaces of the tanks are coated with an epoxy coating. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel tanks. This preventive maintenance activity inspects the internal coating of the fire protection system pressure maintenance accumulator tanks and filtered water tanks to check the condition of the coating to identify coating failures and the condition of the connecting aluminum piping 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 Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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 Catawba, the *Turbine Building Manual Hose Station Flow Test* will be implemented following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

18.2.9 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 Feedwater

Auxiliary Steam

Boron Recycle

Feedwater

Liquid Radwaste

Steam Generator Blowdown Recycle

The only portions of Boron Recycle and Liquid Radwaste 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 Catawba component design code of record.

Defined inspection locations exist in the several systems within the scope of license renewal. Auxiliary Feedwater and Feedwater and Steam Generator Blowdown Recycle each contain multiple inspection locations in susceptible regions. Other defined inspection locations cover several systems that are exposed to the same steam supply environment. Auxiliary Steam, Boron Recycle and Liquid Radwaste 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.

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 Catawba Units 1 and 2 control the *Flow Accelerated Corrosion Program* for Catawba station.

18.2.10 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.

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 – 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 – 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, foreign material exclusion and seismic concerns at Duke Energy's nuclear plants. The *Fluid Leak Management Program* is described and controlled by AD-MN-ALL-0006, *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 23 and 26].

18.2.11 Galvanic Susceptibility Inspection

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

Condenser Circulating Water

Diesel Generator Room Sump Pump

Exterior Fire Protection

Filtered Water

Interior Fire Protection

Liquid Radwaste

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 Catawba site 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 Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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.

18.2.12 Heat Exchanger Activities

18.2.12.1 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.

18.2.12.2 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.

18.2.12.3 Diesel Generator Engine Cooling Water Heater Exchangers

The purpose of the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* is to manage fouling of admiralty 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 admiralty 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.

18.2.12.4 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.

18.2.12.5 Diesel Generator Engine Starting Air

The purpose of the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* is to manage loss of material for parts of the diesel generator engine starting air aftercoolers exposed to treated water. The *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss than can affect the pressure boundary function. The program is credited with managing loss of material for carbon steel, copper alloys, 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 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.

18.2.14 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 Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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.

18.2.15 Inservice Inspection Plan

The Catawba *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 Catawba 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

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.

18.2.15.1 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 Catawba, *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 Catawba Nuclear Station.

18.2.16 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.

The Inspection Program for Civil Engineering Structures and Components is nominally performed every 5 years with the exact schedule being established with consideration of refueling outages for each unit. The interval may be increased to a nominal 10-year frequency with appropriate justification based on the structure, environment, and related inspection results.

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 corrective action program. The corrective action program applies to all structures and components within the scope of the *Inspection Program for Civil Engineering Structures and Components*.

18.2.17 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 Catawba systems:

Liquid Radwaste System - stainless steel components exposed to an unmonitored borated water, unmonitored treated water, or a raw water environment; carbon steel and cast iron components exposed to a raw 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 the 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.

At Catawba, 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 and treated 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 unmonitored borated and treated water environments. The inspection results will be applied to the stainless steel components in the unmonitored borated and treated water environments.

For carbon steel, cast iron, and stainless steel components at Catawba exposed to raw water environments, the sample population will include components located in and around the Liquid Radwaste System sumps. The inspection results will be applied to carbon steel, cast iron, and stainless steel components in the raw water environments.

For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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.

18.2.18 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 electrical 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 8], 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 Catawba, the first inspection per the *Non-EQ Insulated Cables and Connections Aging Management Program* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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 Action & 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.

18.2.19 Pressurizer Spray Head Examination

The MNS and CNS transition to a fire protection program based on NFPA 805 removed the licensing requirement to attain cold shutdown within a prescribed time frame in the event of a fire, instead requiring the plants to attain and maintain “safe and stable conditions”, which can be achieved at hot standby. Accordingly, the pressurizer spray head is no longer functionally required for compliance with fire protection regulations. Since the pressurizer spray heads are not credited in other regulated events in 10 CFR 54.4(a)(3), or for any design basis events, and have no potential adverse interactions with safety related SSCs or safety functions, it follows that they no longer perform an intended function at MNS or CNS, and therefore are no longer within scope of license renewal. Since the pressurizer spray heads are no longer in the scope of license renewal, there are no ongoing aging management requirements under 10 CFR 54, and the Pressurizer Spray Head Examination activity has been deleted at MNS and CNS.

18.2.20 Preventive Maintenance Activities

18.2.20.1 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.

18.2.20.2 Condenser Circulating Water Pump Expansion Joint Inspection

Scope – The scope of the *Condenser Circulating Water Pump Expansion Joint Inspection* is the expansion joints at the discharge of the condenser circulating water pumps that fall within the scope of license renewal. There are four of these expansion joints on each unit at Catawba.

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

Parameters Monitored or Inspected – The parameters inspected by the *Condenser Circulating Water Expansion Pump Joint Inspection* are signs of cracking and wear from exposure to the internal and external environments.

Detection of Aging Effects – The *Condenser Circulating Water Pump Expansion Joint Inspection* is a one-time visual inspection that will detect the presence and extent of degradation on the internal and external surfaces of the expansion joints.

Monitoring & Trending – The *Condenser Circulating Water Pump Expansion Joint Inspection* will visually inspect the internal and external surfaces of the license renewal expansion joints for specific signs of cracking, checking, crazing, cuts, tears, blistering, ply separation, flattened arch, abnormal bulges, scale, flakes, and soft spots.

For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

Should industry data or other evaluation 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 *Condenser Circulating Water Pump Expansion Joint Inspection* is any signs of cracking and wear will be evaluated.

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 effect 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 *Condenser Circulating Water Pump Expansion Joint Inspection* will be implemented in accordance with controlled plant procedures.

18.2.21 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. Each capsule contains 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 of the Catawba Unit 2 reactor vessel. Catawba

Unit 1 reactor vessel specimen 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 Catawba Units are provided in Table 5-40.

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 Catawba UFSAR, Sections 5.3.1.6.2 and 5.3.1.6.3 [Reference 9]. 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. The validation of the transport calculational methodology is provided in WCAP-14040-A [Reference 10]. 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.

18.2.22 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). The decision to perform inspections on Catawba Unit 1 and Catawba Unit 2 will depend on an evaluation of the internals inspections performed on McGuire Units 1 and 2.

With respect to dimensional changes due to void swelling, Catawba 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 Catawba Unit 1 and Catawba 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.

18.2.23 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 Catawba systems:

Exterior Fire Protection

Filtered Water

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 an equivalent test on one cast iron pump casing in the Exterior Fire Protection System at Catawba. The Brinnell Hardness Test or an 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 Catawba 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 an equivalent test on a sample of brass valves at Catawba 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 Catawba, 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 Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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.

18.2.24 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:

Exterior Fire Protection

Interior Fire Protection

Nuclear Service Water

Filtered 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

Deleted Per 2007 Update

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 11] 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.

18.2.25 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 Catawba systems:

Diesel Generator Room Sump Pump System

Groundwater Drainage System

Turbine Building Sump Pump 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 Catawba 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, Groundwater Drainage, and Turbine Building Sump Pump Systems.

For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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. 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.

The portion of the Catawba Turbine Building Sump Pump System within the scope of license renewal is carbon steel piping connecting the Liquid Waste System to the sump. This system was not selected for inspection because it is only applicable to one material and only at the Catawba station. Therefore, the results of the *Sump Pump Systems Inspection* are encompassing and will be applied to the Turbine Building Sump Pump System components subject to a raw water environment.

Acceptance Criteria – The acceptance criterion 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.

18.2.26 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 Catawba systems:

Containment Valve Injection Water

Drinking Water

Solid Radwaste

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 Catawba will inspect stainless steel components, welds, and heat affected zones, as applicable, in the Drinking Water System. The Drinking Water System receives water from the local municipality that has contaminants in excess of limits below which a concern would not exist for cracking and loss of material in stainless steel. Because of the higher starting level of contaminants, the environment in the Drinking Water System is more likely to lead to cracking or loss of material if it is occurring and bounds the environments of the Containment Valve Injection Water and Solid Radwaste Systems. 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. Therefore, the inspection results will serve as a leading indicator and can be applied to the Containment Valve Injection Water and Solid Radwaste Systems.

For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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.

18.2.27 Underwater Inspection of Nuclear Service Water Structures

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

Low Pressure Service Water Intake Structure

Nuclear Service Water Intake Structure

Nuclear Service Water Pump Structure

Standby Nuclear Service Water Discharge Structures

Standby Nuclear Service Water Intake Structure

Standby Nuclear Service Water Pond Outlet

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 nominally performed every five years for Catawba Nuclear Service Water and Standby Nuclear Service Water Intake structures and other Catawba structures. 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.

18.2.28 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 Catawba, this one-time inspection will be completed following issuance of the renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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.

18.2.29 Waste Gas System Inspection

Scope – The scope of the *Waste Gas System Inspection* is carbon steel, stainless steel, and brass 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 Catawba 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 four 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 brass, 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's program owner.

(1) For the brass seal water control valves on the waste gas compressors at Catawba exposed to unmonitored treated water, an inspection will be performed on one of the two seal water control valves. If no parameters are known that would distinguish the susceptible locations, one of the two available at will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the other brass seal water control valve.

(2) For carbon steel components exposed to unmonitored treated water environments at Catawba, 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 Catawba will be examined. If no parameters are known that would distinguish the susceptible locations at Catawba, one of the eight available at Catawba 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.

(3) For stainless steel components exposed to unmonitored treated water environments at Catawba, inspections will be performed on the seal water path of the waste gas compressor. One of two possible locations at Catawba will be examined. If no

parameters are known that would distinguish the susceptible locations at Catawba, one of the two available at Catawba 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.

(4) For the carbon steel components exposed to a gas environment at Catawba, 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 Catawba, one location at Catawba 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 Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba 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.

18.2.30 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. AD-MN-ALL-0006, *Fluid Leak Management Program*.
8. *Guideline for the Management of Adverse Localized Equipment Environments,* EPRI, Palo Alto, CA: 1999. EPRI TR-109619.
9. *Catawba Nuclear Station Updated Final Safety Analysis Report,* as revised.
10. WCAP-14040-A, Revision 4, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*, May 2004.
11. ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components, Subsection ND Class 3 Components, 1971 edition.
12. NRC Bulletin 2003-02, "Leakage from Reactor Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," August 21, 2003.
13. 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.
14. 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.
15. Barron, Henry B. (Duke) to U.S. Nuclear Regulatory Commission, Duke Response to NRC Bulletin 2004-01, July 27, 2004.
16. McCollum, William R. (Duke) to U.S. Nuclear Regulatory Commission, Supplement to Response to NRC Bulletin 2004-01, September 21, 2004.
17. Barret, R. (NRC) to Marion, A. (NEI), Flaw Evaluation Guidelines, April 11, 2003.
18. Morris, James R. (Duke) to U.S. Nuclear Regulatory Commission, Revision to Relief Request 06-GO-001 in Response to September 20, 2006 Conference Call, September 27, 2006.
19. Jones, Ronald A. (Duke) to U.S. Nuclear Regulaory Commission, Relief Request 07-GO-001, January 24, 2007.

20. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Case N-729-1, "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 10CFR50 Industry Codes and Standards; Amended Requirements; Final Rule, Wednesday September 10, 2008 - pages 52742 and 52749.

23. PD-EG-PWR-1611, Boric Acid Corrosion Control Program (Program Description)

24. 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 W86182 Weld Filler Material With and Without Application of Listed Mitigation Activities.

25. 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.

26. AD-EG-PWR-1611, Boric Acid Corrosion Control Program-Implementation (Administrative Procedure)

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18.3 Additional Commitments

18.3.1

Battery rack inspections conducted in accordance with ITS SR 3.8.4.4, SLC 16.7-9-14, and SLC 16.7-9-16 shall include the structural supports and anchorages.

18.3.2

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

18.3.3

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 Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1.)

18.3.4

For Catawba, 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 December 6, 2024 (the end of the initial license of Catawba Unit 1).

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