# Appendix B

# Aging Management Programs and Activities

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# APPENDIX B: AGING MANAGEMENT PROGRAMS AND ACTIVITIES

## B.1 INTRODUCTION

Aging management programs and activities that are credited during the aging management review are described in the remaining sections of Appendix B. The demonstrations, along with the program and activity descriptions, meet the requirement specified in §54.21(a)(3). Along with the technical information contained in Chapters 2, 3, and 4, Appendix B is designed to allow the NRC to make the finding contained in §54.29(a)(1).

**§54.29 Standards for issuance of a renewed license**

*A renewed license may be issued by the Commission up to the full term authorized by §54.31 if the Commission finds that:*

*(a) Actions have been identified and have been or will be taken with respect to the matters identified in Paragraphs (a)(1) and (a)(2) of this section, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the plant's CLB in order to comply with this paragraph are in accord with the Act and the Commission's regulations. These matters are:*

*(1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under §54.21(a)(1); and*

*(2) time-limited aging analyses that have been identified to require review under §54.21(c).*

*(b) Any applicable requirements of Subpart A of 10 CFR Part 51 have been satisfied.*

*(c) Any matters raised under §2.758 have been addressed.*

The aging management programs described in Appendix B include existing, ongoing programs as well as new programs that are currently not implemented. These descriptions of programs and activities are intended to provide an overview of the range of actions required to manage aging. Some of the descriptions have used a series of specific attributes to facilitate the description of the actions. These attributes are defined in Section B.2, of Appendix B.

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## B.2 PROGRAM AND ACTIVITY ATTRIBUTES

Attributes that are utilized in most of the program and activity descriptions for license renewal, with a few exceptions, are described in Appendix B.2. The following information sources served as primary inputs to the attribute definitions used in Appendix B:

1. Application for Renewed Operating Licenses, Oconee Nuclear Station [Reference B - 1]
2. NEI 95-10, Revision 2, Sections 4.2 and 4.3 [Reference B - 2]
3. Draft Standard Review Plan for License Renewal, Appendix A.1 [Reference B - 3]
4. NEI letter dated October 13, 2000 to U. S. Nuclear Regulatory Commission [Reference B - 4]

### B.2.1 TYPES OF PROGRAMS AND ACTIVITIES

The attributes described in the Section B.2.2 are applicable to the following types of programs and activities:

**One-time Inspections**

A one-time inspection is performed for components when the presence of aging effects requiring management are not indicated but cannot be ruled out. The inspection is performed one time only and inspects components for specific indications that are linked to degradation caused by specific aging effects. No actions are taken as part of a one-time inspection to prevent the aging effect or trend inspection results.

For McGuire, these new inspections will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, these new inspections 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). Completion of these new inspections prior to the end of the initial licenses is consistent with conclusions previously made by the Nuclear Regulatory Commission in NUREG-1723.

**Prevention Programs**

The actions of a prevention program preclude specific aging effects from occurring. For example, a coating precludes corrosion of the base metal from occurring.

**Mitigation Programs**

A mitigation program attempts to slow the effects of aging. For example, water chemistry mitigates internal corrosion of piping.

**Condition Monitoring Programs**
A condition monitoring program inspects or examines the presence or extent of aging effects. For example, the ASME Section XI Inservice Inspection Program which requires visual, surface and volumetric examinations is a condition monitoring program.

**Performance Monitoring Programs**
Performance monitoring programs test the ability of a structure or component to perform its intended function. For example, heat balances test the heat transfer function of heat exchanger tubes.

### B.2.2 ATTRIBUTE DEFINITIONS

The attribute definitions used to describe new and existing programs and activities are provided below.

**Scope –** This program attribute identifies the specific structures or components managed by the program or activity.

**Preventive Actions –** This program attribute describes the actions taken in the period of extended operation to either prevent aging effects from occurring or mitigate (i.e., lessen or slow down) aging degradation for prevention and mitigation programs. This attribute is not applicable for one-time inspections, condition monitoring and performance monitoring programs.

**Parameters Monitored or Inspected –** This program attribute describes “what” is being monitored or inspected for all inspections and programs. These descriptions include the observable parameters or indicators to be monitored or inspected for each aging effect managed. The observable parameters should be linked to the degradation of the structure or component intended functions in the period of extended operation.

**Detection of Aging Effects –** The detection of aging effects should occur before there is a loss of structure and component intended function(s).

**Monitoring & Trending –** This program attribute describes “when,” “where” and “how” program data is collected; i.e., all aspects of activities to collect data as part of the program. This description includes aspects such as method or technique (e.g., visual, volumetric, surface inspection), frequency, sample size, and timing of new/one-time inspections. This attribute also provides information that links the parameters to be monitored or inspected to the aging effects being managed.

Trending is a comparison of the current monitoring results with previous monitoring results in order to make predictions for the future and to initiate actions as necessary.

**Acceptance Criteria –** This program attribute describes the acceptance criteria for ensuring the structure or component intended function is maintained during the period of extended operation. The acceptance criteria may be based on design or current licensing basis information as well as established industry codes or standards.

**Corrective Action & Confirmation Process –** This program attribute describes the actions to be taken in the extended period of operation when the acceptance criteria or standard is not met. The corrective action and confirmation process that is described for each aging management program or activity applies to all structures and components within the scope of the program or activity. In some cases the program itself includes its own corrective action and confirmation process.

In other cases, the corrective action process is credited for corrective action and confirmation process. The corrective action process is a formal corrective action program which facilitates the correction of conditions adverse to quality. Corrective actions are documented. Data are periodically reviewed to identify positive or negative changes and to initiate additional actions, as necessary. The corrective action process is implemented by Nuclear System Directives NSD 208, Problem Investigation Process and NSD 223, Trending of PIP Data.

**Administrative Controls –** This program attribute describes the administrative structure under which the programs and activities are executed. Examples of various administrative structures include program manuals, nuclear station directives, engineering support documents, plant procedures, and work orders. The administrative controls provide for a review and approval process.

**Operating Experience –** This program attribute provides the objective evidence that supports the determination that the program or activity provides reasonable assurance that the effects of aging will be adequately managed such that the structure or component intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation (i.e., 20-years from the end of the initial operating license).

Plant specific operating experience includes licensee event reports, reports documenting the results of the credited program or activity, as well as plant maintenance and operating records.

Several programs and activities contained in Appendix B are equivalent to the corresponding programs and activities that were described in the Application for Renewed Operating Licenses of Oconee Nuclear Station, Units 1, 2, and 3. In these instances, the pertinent section of NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," is referenced [Reference B - 5].

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# B.3 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

## B.3.1 ALLOY 600 AGING MANAGEMENT REVIEW

*Note: The ALLOY 600 AGING MANAGEMENT REVIEW is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Alloy 600 Aging Management Review* is to ensure that nickel-based alloy locations are adequately inspected by the *Inservice Inspection Plan* (Appendix B.3.20) or other existing programs such as the *Control Rod Drive Mechanism and Other Vessel Head Penetration Program* (Appendix B.3.9), the *Reactor Vessel Internals Inspection* (Appendix B.3.27)*, and the Steam Generator Integrity Program* (Appendix B.3.31). The review will demonstrate the general oversight and management of cracking due to primary water stress corrosion cracking (PWSCC).

The *Alloy 600 Aging Management Review* will identify Alloy 600/690, 82/182 and 52/152 locations. A ranking of susceptibility to PWSCC will be performed for the nickel-based alloy locations. A review will be performed to ensure that nickel-based alloy locations are adequately inspected by the *Inservice Inspection Plan* (Appendix B.3.20) or other existing programs such as the *Control Rod Drive Mechanism and Other Vessel Head Penetration Program* (Appendix B.3.9), the *Reactor Vessel Internals Inspection* (Appendix B.3.27)*, and the Steam Generator Integrity Program* (Appendix B.3.31). This review will utilize industry and Duke specific operating experience. Inspection method and frequency of inspection for the Alloy 600/690, 82/182, and 52/152 locations for the period of extended operation will be adjusted as needed based on the results of this review. In addition, supplemental inspections for the period of extended operation will be developed as needed.

For McGuire, this review will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this review 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). The results of this review will be incorporated into the unit specific inservice inspection (ISI) plans for the ISI intervals during the period of extended operation.

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## B.3.2 BATTERY RACK INSPECTIONS

*Note: The BATTERY RACK INSPECTIONS are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Loss of material due to corrosion is an aging effect requiring programmatic management for steel battery racks for the extended period of operation. The *Battery Rack Inspections* are credited with managing loss of material that could impact the intended function of structural support. The *Battery Rack Inspections* are condition monitoring programs.

The regulatory basis for inspecting battery racks is found in the McGuire and Catawba Technical Specifications and Selected Licensee Commitments as identified:

<u>McGuire</u>
EPL System - Technical Specification (SR) 3.8.4.3
EPQ System - Selected Licensee Commitment 16.8.3.3
EQD System - Selected Licensee Commitment 16.9.7.12
ETM System - Selected Licensee Commitment 16.9.7.17

<u>Catawba</u>
EPL System - Technical Specification (SR) 3.8.4.4
EPQ System - Technical Specification (SR) 3.8.4.4
EQD System - Selected Licensee Commitment 16.7-9.2
ETM System - Selected Licensee Commitment 16.7-9.4

**Scope –** The scope of the *Battery Rack Inspections* include the battery racks for the following systems:

- EPL System (Vital Batteries)
- EPQ System (Diesel Generator Batteries)
- ETM System (Standby Shutdown Facility Batteries)
- EQD System (Standby Shutdown Facility Diesel Batteries)

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

**Parameters Monitored or Inspected –** The *Battery Rack Inspections* provide visual examination of the battery racks for physical damage or abnormal deterioration, including loss of material.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Battery Rack Inspections* will detect loss of material prior to loss of battery rack intended function.

**Monitoring & Trending –** The *Battery Rack Inspections* perform visual inspections to detect loss of material in accordance with McGuire and Catawba Technical Specifications and Selected Licensee Commitments. The inspections are based on guidance provided in IEEE 450-1980 [Reference B - 6]. No actions are taken as part of this program to trend inspection results.

- EPL System battery racks are inspected every 18 months in accordance with Technical Specifications.

- EPQ System battery racks are inspected every 18 months in accordance with McGuire Selected Licensee Commitment and Catawba Technical Specification.
- 
- EQD System battery racks are inspected every 18 months in accordance with McGuire and Catawba Selected Licensee Commitments.
- 
- ETM System battery racks are inspected every 18 months in accordance with McGuire and Catawba Selected Licensee Commitments.

**Acceptance Criteria –** The acceptance criterion is no visual indication of loss of material.

**Corrective Action & Confirmation Process –** Areas which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repaired as required. Structures and components that 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.

**Administrative Controls –** The *Battery Rack Inspections* are governed by McGuire and Catawba Technical Specifications and Selected Licensee Commitments. The *Battery Rack Inspections* are implemented by controlled plant procedures, as required by Technical Specification 5.4, and work management system using model work orders.

**Operating Experience –** A review of McGuire and Catawba-specific surveillance records did not identify any instances where abnormal deterioration, which would include loss of material, of the battery racks had occurred.

***<u>Conclusion</u>***

The *Battery Rack Inspections* have been demonstrated to be capable of detecting and managing loss of material. The *Battery Rack Inspections* described above are equivalent to the *Battery Rack Inspections* described and evaluated in NUREG-1723, Section 3.8.3.2.1 [Reference B - 5]. Based on the above review, the continued implementation of the *Battery Rack Inspections* provides reasonable assurance that loss of material will be managed such that the intended functions of the battery racks will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.3 BORAFLEX MONITORING PROGRAM

*Note: The BORAFLEX MONITORING PROGRAM is applicable only to McGuire Nuclear Station.*

Degradation due to gamma irradiation has been identified as an aging effect requiring programmatic management for the Boraflex neutron-absorbing panels in spent fuel storage racks for the extended period of operation. The function of the Boraflex panels is to ensure that reactivity of the storage fuel assemblies is maintained within required limits. Boraflex has been shown to degrade as a result of gamma irradiation and exposure to the spent fuel pool environment. The B4C poison material can be removed, thereby reducing the poison worth of the Boraflex sheets. This phenomenon is documented in NRC Generic Letter 96-04, “Boraflex Degradation in Spent Fuel Pool Storage Racks.” The *Boraflex Monitoring Program* is credited with managing aging of Boraflex panels for the period of extended operation. The *Boraflex Monitoring Program* is a performance monitoring program.

**Scope –** The scope of the *Boraflex Monitoring Program* includes all Boraflex neutron-absorbing panels in the McGuire Units 1 and 2 spent fuel storage racks. Catawba Nuclear Station does not use Boraflex.

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

**Parameters Monitored or Inspected –** The *Boraflex Monitoring Program* monitors the Boraflex panel average storage rack poison material Boron 10 areal density. The panel average Boron 10 areal density is used as an input to the spent fuel pool storage rack criticality calculations. In addition, the silica levels are monitored in the spent fuel pool. The silica levels provide an indication of the depletion of boron carbide from Boraflex.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Boraflex Monitoring Program* will monitor Boraflex panel areal density prior to loss of intended function.

**Monitoring & Trending –** The *Boraflex Monitoring Program* includes in-situ testing of the Boron 10 areal density. The frequency of testing is every three years. Testing may be performed more frequently based on engineering judgment, spent fuel pool water chemistry, and modeling projections of Boraflex degradation. Selection of Boraflex panels for in-situ testing is based upon predicted Boron 10 areal density loss.

**Acceptance Criteria –** The acceptance criteria are based on maintaining the minimum areal density of B4C assumed in the criticality calculations. The requirements are listed in

McGuire Selected Licensee Commitment (SLC) 16.9.24, *Spent Fuel Pool Storage Rack Poison Material.*

**Corrective Action & Confirmation Process –** The specified corrective actions are identified in McGuire Selected Licensee Commitments 16.9.24, *Spent Fuel Pool Storage Rack Poison Material*.

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 *Boraflex Monitoring Program* is governed by McGuire Selected Licensee Commitment (SLC) 16.9.24.

**Operating Experience –** Blackness testing was performed at McGuire on the spent fuel storage racks in 1991. The testing was performed to measure the amount of pullback at the ends of the Boraflex panels caused by shrinkage, as well as the size and frequency of gap formation. Shrinkage and gap formation were observed in both pools at McGuire. The data was incorporated into revised criticality analyses for the storage racks and $k_{eff}$ was verified less than or equal to 0.95 [Reference B - 7].

In 1996 as a result of industry-wide experience with degradation of Boraflex, the NRC issued Generic Letter 96-04, *Boraflex Degradation in Spent Fuel Pool Storage Racks* [Reference B - 8]. Generic Letter 96-04 provided descriptions of several industry experiences and a discussion of relevant experimental data from test programs. The staff stated that on the basis of test and surveillance information from plants that had detected areas of Boraflex degradation, no safety concern existed that warranted immediate action. In issuing Generic Letter 96-04, the staff requested that all licensees with installed spent fuel pool storage racks containing the neutron absorber Boraflex provide an assessment of the physical condition of the Boraflex.

The Duke response to this request was provided in a letter to the NRC dated October 22, 1996 [Reference B - 7] and supplemented in December 22, 1997 [Reference B - 9]. The response indicated, in part, that Duke had acquired the RACKLIFE computer code which had been developed by the Electric Power Research Institute for the purpose of assessing overall Boraflex thinning based upon cumulative gamma exposure, storage rack design parameters, and dissolved silica concentration in the spent fuel pool. In-situ measurements were performed that verified that the *Boraflex Monitoring Program* accurately predicts the Boron 10 areal density.

***Conclusion***

The *Boraflex Monitoring Program* has been demonstrated to be capable of detecting and managing degradation of the Boraflex panels. The *Boraflex Monitoring Program* described above is equivalent to the Boraflex Monitoring Program described and evaluated in NUREG-1723, Section 4.2.10 [Reference B - 5]. Based on the above review, the continued implementation of the *Boraflex Monitoring Program* provides reasonable assurance that degradation of the Boraflex panels will be managed such that the intended function of the Boraflex panels will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.4 BORATED WATER SYSTEMS STAINLESS STEEL INSPECTION

*Note: The BORATED WATER SYSTEMS STAINLESS STEEL INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Borated Water Systems Stainless Steel Inspection* is to characterize any loss of material or cracking of stainless steel components exposed to alternate wetting and drying in borated water environments. Uncertainty exists as to whether alternate wetting and drying of the borated water could cause aging in stainless steel components such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect stainless steel components exposed to an alternate wetting and drying borated water environment to detect the presence and extent of any loss of material or cracking. The *Borated Water Systems Stainless Steel Inspection* is a one-time inspection.

**Scope –** The scope of the *Borated Water Systems Stainless Steel Inspection* is stainless steel components exposed to an alternate wetting and drying borated water environment in the following McGuire and 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 McGuire and 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 each site will be inspected using volumetric technique. If no parameters are known that would distinguish the susceptible locations at each site, one of the twelve available at each site will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the specific stainless steel components exposed to an alternate wetting and drying borated water environment in the Refueling Water System.

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

**Operating Experience –** The *Borated Water Systems Stainless Steel Inspection* is a one-time inspection activity for which there is no operating experience. However, an equivalent inspection was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

***Conclusion***

The *Borated Water Systems Stainless Steel Inspection* described above is equivalent to the Reactor Building Spray System Inspection described and evaluated in NUREG-1723, Section 3.5.3.2 [Reference B - 5]. Based on the above review, implementation of the *Borated Water Systems Stainless Steel Inspection* will adequately verify that no need exists to manage aging effects on the component or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.5 BOTTOM-MOUNTED INSTRUMENTATION THIMBLE TUBE INSPECTION PROGRAM

*Note: The BOTTOM MOUNTED INSTRUMENTATION THIMBLE TUBE INSPECTION PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Bottom Mounted Instrumentation Thimble Tube Inspection Program* is to identify loss of material due to wear in the bottom mounted instrumentation (BMI) thimble tubes prior to leakage. The thimble tubes are part of the reactor coolant pressure boundary. The *Bottom Mounted Instrumentation Thimble Tube Inspection Program* is a condition monitoring program.

**Scope –** The scope of the *Bottom Mounted Instrumentation Thimble Tube Inspection Program* includes all 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* 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.

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

**Monitoring & Trending –** Inspection of the BMI thimble tubes is performed using eddy current testing. All of the 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 B - 11]. 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 criteria for the BMI thimble tubes is 80% through wall (thimble tube wall thickness is not less than 20% of initial wall thickness). This acceptance criteria was developed by Westinghouse in WCAP 12866, “Bottom Mounted Instrumentation Flux Thimble Wear,” and reported to the NRC by Duke [Reference B - 10].

**Corrective Action & Confirmation Process –** Thimble tubes that are predicted to exceed the acceptance criteria 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 are determined using engineering calculations using methodology based on the information Westinghouse developed in WCAP-12866 [Reference B - 11].

**Operating Experience –** Flux thimble wear was first identified as an issue when three flux thimble wore through over a three month period at the Salem plant in 1981. Since that time numerous plants both in the U. S. and abroad have detected thimble wear in varying degrees, ranging from small amounts to through wall. Westinghouse has determined the cause of this wear to be flow induced vibration of the flux thimble inside of the reactor vessel lower internals support column. Wear of the thimbles is a concern since the thimble serves as a portion of the reactor coolant system pressure boundary.

On July 26, 1988, the NRC issued IE Bulletin 88-09: *Thimble Tube Thinning in Westinghouse Reactors* [Reference B - 12]. The NRC requested that inspection programs be implemented that included:

- The establishment, with technical justification, of an appropriate thimble tube wear acceptance criterion (for example, percent through wall loss). This acceptance criterion should include allowances for such items as inspection methodology and wear scar geometry uncertainties.
- The establishment, with technical justification, of an appropriate inspection frequency (for example, every refueling outage).
- The establishment of an inspection methodology that is capable of adequately detecting wear of the thimble tubes (such as eddy current testing).

Duke has implemented a program at McGuire and Catawba Nuclear Stations that meets these criteria based on a proprietary study performed for the Westinghouse Owners Group [Reference B - 11].

Since the issuance of IE Bulletin 88-09 three inspections have been performed on Catawba Unit 1 and three on Catawba Unit 2 thimble tubes. The inspections on Unit 1 were performed during End Of Cycle (EOC) 1EOC-3 (1988), 1EOC-7 (1993) and 1EOC-11 (1999). The Inspections on Unit 2 were performed during 2EOC-2 (1989), 2EOC-3 (1990), and 2EOC-5 (1993). Inspections are not detecting significant changes in wear rates for either Unit. Currently no tubes are capped on Unit 1 and two tubes are capped on Unit 2 due to wear concerns. Wear projections performed in the referenced calculations have determined that further testing will not be required until 1EOC-7 (2008) and 2EOC-15 (2007), respectively for Units 1 and 2, barring significant changes in cycle length or reactor geometry.

Similar inspections have been performed on McGuire Units 1 and 2. Unit 1 has been inspected twice, during 1EOC-5 (1988) and 1EOC-14 (2001) with 10 tubes showing detectable wall loss. Two additional tubes were capped due to other types of damage. Unit 2 was inspected during 2EOC-5 (1989) and 2EOC-8 (1993), with eight tubes showing wear. The future inspections are currently planned to occur at 1EOC-19 (2008) for Unit 1 and 2EOC-16 (2005) for Unit 2.

***Conclusion***

The *Bottom Mounted Instrumentation Thimble Tube Inspection Program* has been demonstrated to be capable of identifying loss of material due to wear in the thimble tubes prior to leakage. Based on the above review, the continued implementation of the *Bottom Mounted Instrumentation Thimble Tube Inspection Program* provides reasonable assurance that the aging effect will be managed and that the bottom mounted instrumentation will continue to perform its intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

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### B.3.6 CHEMISTRY CONTROL PROGRAM

*Note: The Chemistry Control Program is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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 and cracking 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* is a mitigation program.

**Scope –** The scope of the *Chemistry Control Program* is the mechanical components exposed to borated water, closed cooling water, fuel oil, and treated water environments in the following Catawba and McGuire systems:

- Auxiliary Feedwater System
- Auxiliary Steam System
- Auxiliary Ventilation System (CNS Only)
- Boron Recycle System
- Building Heating Water or Heating Water System
- Chemical and Volume Control System
- Component Cooling System
- Condensate System (CNS Only)
- Condensate Storage System (CNS Only)
- Containment Spray System
- Control Area Chilled Water System
- Control Area Ventilation or Control Room Area Ventilation
- Conventional Chemical Addition System (MNS Only)
- Diesel Generator Cooling Water System
- Diesel Generator Fuel Oil System
- Diesel Generator Lube Oil System
- Demineralized Water or Make-up Demineralized Water System
- Equipment Decontamination System (CNS Only)
- Feedwater System
- Feedwater Pump Turbine Exhaust System or Turbine Exhaust System
- Ice Condenser Refrigeration System
- Liquid Radwaste or Liquid Waste Recycle System
- Liquid Waste Monitor and Disposal System (MNS Only)
- Main Steam System
- Main Steam Supply to Auxiliary Equipment System
- Main Steam Vent to Atmosphere System
- Nuclear Sampling System
- Reactor Coolant System
- Recirculated Cooling Water System (CNS Only)
- Refueling Water System
- Residual Heat Removal System
- Safety Injection System
- Spent Fuel Cooling System
- Standby Shutdown Diesel System
- Steam Generator Blowdown or Steam Generator Blowdown Recycle System
- Steam Generator Wet Lay-up Recirculation System
- Waste Gas System

The scope of the program also includes the spent fuel pool liner, structural stainless steel and plates, and racks located in the spent fuel pool.

**Preventive Actions –** The *Chemistry Control Program* monitors and controls the relevant conditions such as halogens, dissolved oxygen, conductivity, biological activity, and corrosion inhibitor concentrations to manage loss of material and cracking. These corrosive contaminants are either removed, their concentrations minimized, or treatments are added and/or maintained to negate their corrosive tendencies.

**Parameters Monitored or Inspected –** The *Chemistry Control Program* monitors specific parameters such as halogens, dissolved oxygen, conductivity, biological activity, and corrosion inhibitor concentrations. The specific parameters monitored vary depending on the system.

**Detection of Aging Effects –** No actions are taken as a part of this program to detect aging effects.

**Monitoring & Trending –** The *Chemistry Control Program* manages the following environments: (1) borated water, (2) closed cooling water, (3) fuel oil, and (4) treated water.

For components exposed to borated water, the *Chemistry Control Program* draws and analyzes samples for contaminant concentrations to mitigate loss of material and/or cracking of components. Concentrations of contaminants such as dissolved oxygen, halogens, and sulfates are determined on a periodic basis. Monitoring and controlling the environment in the Reactor Coolant, Refueling Water, and Spent Fuel Cooling Systems will control the borated water environment and mitigate aging in the following license renewal systems.

- Boron Recycle
- Chemical and Volume Control
- Containment Spray
- Equipment Decontamination (CNS Only)
- Nuclear Sampling
- Residual Heat Removal
- Safety Injection

Monitoring and trending in the Reactor Coolant, Refueling Water and Spent Fuel Cooling Systems ensures quick detection of unfavorable trends and prompt corrective actions to mitigate corrosion.

For components exposed to closed cooling water, the *Chemistry Control Program* draws and analyzes samples for contaminant and treatment concentrations to mitigate loss of material and/or cracking of components. Concentrations of corrosion inhibitors are determined on a periodic basis to be within specific ranges. Monitoring and controlling the closed cooling water environment in the following systems will mitigate aging of components in these systems.

- Auxiliary Building Ventilation
- Building Heating or Heating Water
- Component Cooling
- Control Area Chilled Water
- Diesel Generator Cooling Water
- Ice Condenser Refrigeration
- Recirculated Cooling Water
- Standby Shutdown Diesel

In addition, monitoring and controlling the closed cooling water environment in the above systems will mitigate aging of heat exchangers in the following systems exposed to the closed cooling water environment of the above systems.

- Chemical and Volume Control
- Control Area or Control Room Area Ventilation
- Diesel Generator Lube Oil
- Residual Heat Removal
- Waste Gas

Monitoring and trending in the systems listed above ensures quick detection of unfavorable trends and prompt corrective actions to mitigate corrosion.

For components exposed to fuel oil, the *Chemistry Control Program* draws and analyzes samples for contaminant concentrations to mitigate loss of material and/or cracking of components. Concentrations of contaminants such as water and biological activity are determined on a periodic basis.

Monitoring and controlling the environment in the Diesel Generator Fuel Oil and Standby Shutdown Diesel Systems will control the fuel oil environment and mitigate aging. Monitoring and trending in these systems ensures quick detection of unfavorable trends and prompt corrective actions to mitigate corrosion.

For components exposed to treated water, the *Chemistry Control Program* draws and analyzes samples for contaminant and treatment concentrations to mitigate loss of material and/or cracking of components. Concentrations of contaminants such as dissolved oxygen, halogens, and sulfates are determined on a periodic basis. Monitoring and controlling the treated water environment in the Demineralized Water, Feedwater, and Steam Generator Wet Lay-up Recirculation Systems will mitigate aging of components exposed to treated water in the following systems.

- Auxiliary Feedwater
- Auxiliary Steam
- Condensate (CNS Only)
- Condensate Storage (CNS Only)
- Conventional Chemical Addition (MNS Only)
- Equipment Decontamination (CNS Only)
- Feedwater Pump Turbine Exhaust or Turbine Exhaust
- Liquid Radwaste or Liquid Waste Recycle
- Liquid Waste Monitor and Disposal (MNS Only)
- Main Steam
- Main Steam Supply to Auxiliary Equipment
- Main Steam Vent to Atmosphere
- Nuclear Sampling
- Steam Generator Blowdown or Steam Generator Blowdown Recycle
- Steam Generator Wet Lay-up Recirculation

Monitoring and trending in the Demineralized Water, Feedwater, and Steam Generator Wet Lay-up Recirculation Systems ensures quick detection of unfavorable trends and prompt corrective actions to mitigate corrosion.

**Acceptance Criteria –** The *Chemistry Control Program* contains system specific acceptance criteria that are based on the guidance provided in the EPRI chemistry guidelines [References B - 13, B - 14, and B - 15], Technical Specifications [References B - 16 and B - 17, Specification 3.8.3], UFSAR [References B - 18 and B - 19], and vendor recommendations for water and fuel oil quality.

**Corrective Action & Confirmation Process –** The *Chemistry Control Program* provides corrective actions when monitored parameters are trending unfavorably but have not violated an acceptance criteria. Additional sampling and analysis are performed after the corrective actions have been taken to confirm the effectiveness of the corrective actions in returning the parameters to acceptable levels. Parameters that have exceeded their acceptance criteria are entered into the corrective action program for a fuller investigation in addition to the corrective actions required by the *Chemistry Control Program* to return the parameter to acceptable levels. Specific corrective actions as a result of the fuller investigation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Chemistry Control Program* is controlled by the site program manuals and implemented by controlled plant procedures. The program manuals at each site provide guidance for maintaining a suitable system environment. These manuals are based on the guidance of the EPRI chemistry guidelines [References B - 13, B - 14, B - 15], Technical Specifications [References B - 16 and B - 17, Specification 3.8.3], the UFSAR [References B - 18 and B - 19] and vendor recommendations to manage loss of material and/or cracking of components exposed to borated water, closed cooling water, fuel oil, or treated water.

**Operating Experience –**A review of operating experience did not reveal a loss of the component intended function of components exposed to borated and treated water that could be attributed to the inadequacy of the *Chemistry Control Program*. This operating experience confirms the effectiveness of the *Chemistry Control Program* for borated and treated water to manage the aging effects when continued into the extended period of operation.

Operating experience did reveal several instances of cracking at welds due to nitrate induced stress corrosion of carbon steel components in the Component Cooling Systems at Catawba and McGuire. A review of industry operating experience identified incidents of nitrate induced stress corrosion cracking of carbon steel components in comparable systems at several other utilities. The investigation determined that biological activity in protected areas were converting the nitrite corrosion inhibitors to nitrates, creating a highly corrosive localized environment. This occurred when nitrate concentrations were allowed to drift to higher than recommended limits. The *Chemistry Control Program* was modified to more rigorously control and lower system corrosion inhibitor concentrations along with the addition of biocides to control biological activity. The conductivity and conductivity to nitrite ratios are monitored as well as nitrate concentration. No other instances of nitrate induced stress corrosion cracking of carbon steel in the Component Cooling System have occurred since these changes were implemented.

A review of operating experience did not reveal any instances of a loss of the component intended function of components exposed to fuel oil that could be attributed to the inadequacy of the *Chemistry Control Program*. Inspection of the Diesel Generator Fuel Oil System storage tanks revealed a light oxide layer and minor pits in the tank low point where water is likely to collect. The remaining internal surfaces of the tanks were free of aging degradation. These inspections and operating experience confirm the effectiveness of the *Chemistry Control Program* for fuel oil to manage the aging effects when continued into the extended period of operation.

***Conclusion***

The *Chemistry Control Program* has been demonstrated to be capable of managing loss of material and/or cracking of components exposed to borated water, closed cooling water, fuel oil, and treated water environments. The *Chemistry Control Program* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.2.2 [Reference B - 5]. Based on the above review, the continued implementation of the *Chemistry Control Program* provides reasonable assurance that the aging effects will be managed and that the components will continue to perform their intended function(s) for the extended period of operation.

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### B.3.7 CONTAINMENT INSERVICE INSPECTION PLAN – IWE

*Note: The CONTAINMENT INSERVICE INSPECTION PLAN - IWE is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Loss of material has been identified as an aging effect requiring management for the ASME Code Class MC pressure retaining steel components and their integral attachments for the extended period of operation.

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 B - 20]. The *Containment Inservice Inspection Plan - IWE* is a condition monitoring program.

**Scope –** The scope of the *Containment Inservice Inspection Plan - IWE* includes examination of items specified in Subsection IWE-1000, except for items that are non-mandatory as documented in §50.55a(b)(2)(ix)(C) and for items whose examinations have been eliminated as a result of approved alternatives submitted in accordance with §50.55a(a)(3). The components within the scope of Subsection IWE at McGuire and Catawba are metal containment pressure retaining components and their integral attachments; metal containment pressure retaining bolting; and metal containment surface areas, including welds and base metal. Subsection IWE exempts from examination (1) components that are outside the boundaries of the containment as defined in the plant-specific design specification; (2) embedded or inaccessible portions of containment components that met the requirements of the original construction code of record; (3) components that become embedded or inaccessible as a result of vessel repair or replacement, provided IWE-1232 and IWE-5220 are met; and (4) piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel. Section 50.55a(b)(2)(ix) specifies additional requirements for inaccessible areas. It states that the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas.

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

**Parameters Monitored or Inspected –** The *Containment Inservice Inspection – IWE* inspects for the following conditions or parameters.  Coated surfaces are examined for evidence of flaking, blistering, peeling, discoloration, and other signs of distress.  Uncoated areas are examined for evidence of cracking, discoloration, wear, pitting, corrosion, gouges, surface discontinuities, dents, and other signs of surface irregularities.  Moisture barriers are examined for wear, erosion, separation from surfaces, embrittlement/cracking, or other defects that may permit moisture intrusion to inaccessible surfaces of the containment.  Bolted connections are examined for defects that could affect leak-tightness or structural integrity.

Table IWE-2500-1 specifies seven categories for examination.  Table IWE-2500-1 references the applicable section in IWE-3500 that identifies the aging effects that are evaluated.  The *Containment Inservice Inspection Plan - IWE* does not require monitoring or inspection of the following items in accordance with Table IWE-2500-1:

- Category E-B, Items E3.10, E3.20, and E3.30 (Containment Penetration Welds, Flange Welds, and Nozzle-to-Shell Welds)
- Category E-D, Items E5.10 and E5.20 (Seals and Gaskets)
- Category E-F, Item E7.10 (Dissimilar Metal Welds)
- Category E-G, Item E8.20 (Bolted Connections – Bolt Torque or Tension Test)

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Containment Inservice Inspection Plan - IWE* will detect loss of material prior to loss of structure or component intended functions.

**Monitoring & Trending –** The frequency and scope of examinations specified in the *Containment Inservice Inspection Plan – IWE* are sufficient to ensure that aging effects would be detected before they would compromise the design basis requirements.  The extent and frequency of examinations are specified in IWE-2400 and IWE-2500.  The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.  Subsection IWE examinations are performed as prescribed during each ten year interval.  The method of examination for each item is specified in IWE-2500 and Table IWE-2500-1.

All surface areas are monitored by virtue of examinations performed in accordance with IWE-2400 and IWE-2500.  When component examination results require evaluation of flaws, evaluation of areas of degradation, or repairs, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period, in accordance with Examination Category E-C (containment surfaces requiring augmented examination). When these reexaminations reveal that the flaws, areas of degradation, or repairs remain essentially unchanged for three consecutive inspection periods, these areas no longer require augmented examination in accordance with Examination Category E-C. IWE-2430 requires that (a) examinations performed during any one inspection that reveal flaws or areas of degradation exceeding the acceptance standards shall be extended to include an additional number of examinations within the same category approximately equal to the initial number of examinations, and (b) when additional flaws or areas of degradation that exceed the acceptance standards are revealed, all of the remaining examinations within the same category must be performed to the extent specified in Table IWE-2500-1 for the inspection interval. Alternatives to these examination requirements are provided in 10 CFR 50.55a(b)(2)(ix)(D), and as documented in approved Requests for Relief, submitted in accordance with 10 CFR 50.55a(a)(3).

**Acceptance Criteria –** The *Containment Inservice Inspection Plan – IWE* implements the acceptance criteria specified in Table IWE-3410-1 for each Examination Category (E-A, E-C, etc.). Areas that do not meet the acceptance standards of Table IWE-3410-1 shall be accepted by engineering evaluation, repair, or replacement, as required by IWE-3122.

**Corrective Action & Confirmation Process –** Subsection IWE states that components whose examination results indicate flaws or areas of degradation that do not meet the acceptance standards listed in Table IWE-3410-1 can be considered acceptable if an engineering evaluation indicates that the flaw or area of degradation is nonstructural in nature or has no effect on the structural integrity of the containment, or if such areas are repaired in accordance with IWE-3122.2 and IWE-4000 or replaced in accordance with IWE-3122.3 and IWE-7000. Such areas are subject to the requirements of IWE-2420(b) and (c), and additional examination requirements of IWE-2430, as modified by 10 CFR 50.55a(b)(2)(ix)(D).

When repairs are performed, the requirements of IWE-3124 apply, and the recorded results of reexaminations must demonstrate that the repair meets the acceptance standards set forth in Table IWE-3410-1. For repairs and replacements, the preservice examination requirements of IWE-2200(d) and the system pressure test requirements of IWE-5000 shall be satisfied, providing additional assurance that the repairs or replacements are acceptable.

**Administrative Controls –** The *Containment Inservice Inspection Plan - IWE* is implemented through administrative procedures. The licensee is responsible for preparation of plans, schedules, and inservice inspection records and reports. IWA-6000 specifically covers the requirements for the preparation, submittal, and retention of records and reports.

**Operating Experience –**

***McGuire Operating Experience***

*Containment Inservice Inspection Plan - IWE* inspections have been performed at McGuire during 1EOC-13, 1EOC-14, 2EOC-12, and 2EOC-13. Inspection results have included the following:

- coatings degradation
- loss of material due to corrosion of Steel Containment Vessel (SCV) shell, stiffener rings, penetration sleeves, process piping, and bolted connections
- missing and cracked/separated moisture barriers

Conditions which required reportability in accordance with 10 CFR 50.55a(b)(2)(ix) are documented in letters to the NRC. For example, the most recent McGuire Containment Inservice Inspection that detected conditions requiring reporting is documented in a letter to the NRC dated January 11, 2001 [Reference B - 21].

Prior to implementation of the *Containment Inservice Inspection Plan- IWE*, inspections were performed in accordance with Appendix J to 10 CFR Part 50. Degradation due to corrosion of the steel containment vessel was identified during these inspections and was documented in LERs 89-20 and 90-06. The corrosion was evaluated and it was determined that the corrosion did not inhibit the ability of the SCV to perform its intended functions. The steel containment vessel was recoated and modifications were made to minimize the potential for reoccurrence.

***Catawba Operating Experience***

*Containment Inservice Inspection Plan - IWE* inspections have been performed at Catawba during 1EOC-11, 1EOC-12, 2EOC-9, and 2EOC-10. Inspection results have included the following:

- coatings degradation
- loss of material due to corrosion of Steel Containment Vessel (SCV) shell, stiffener rings, penetration sleeves, process piping, and bolted connections
- missing/damaged parts on equipment hatch latch bolting
- missing and cracked/separated moisture barriers

Conditions which required reportability in accordance with 10 CFR 50.55a(b)(2)(ix) are documented in letters to the NRC. For example, the most recent Catawba Containment Inservice Inspection that detected conditions requiring reporting is documented in a letter to the NRC dated May 1, 2000 [Reference B - 22].

***<u>Conclusion</u>***

The *Containment Inservice Inspection Plan - IWE* is maintained and implemented in accordance with the requirements of 10 CFR 50.55a and shall remain in effect during the period of extended operation for McGuire and Catawba. The *Containment Inservice Inspection Plan - IWE* has been demonstrated to be capable of detecting and managing loss of material. The *Containment Inservice Inspection Plan - IWE* described above is equivalent to the *Containment Inservice Inspection Plan* described and evaluated in NUREG-1723, Section 3.3.3 [Reference B - 5]. Based on the above review, the continued implementation of *Containment Inservice Inspection Plan - IWE* provides reasonable assurance that the Containment steel components will be managed such that the component intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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## B.3.8 CONTAINMENT LEAK RATE TESTING PROGRAM

*Note: The CONTAINMENT LEAK RATE TESTING PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Loss of material for the Reactor Building pressure boundary components and cracking of the bellows have been identified as aging effects requiring management for the extended period of operation. The *Containment Leak Rate Testing Program* supplements the *Containment Inservice Inspection Plan- IWE*. The *Containment Inservice Inspection Plan- IWE* which implements the provisions of the ASME Code Section XI, Subsection IWE, is the primary method for detection of aging effects for the steel components of containment. The *Containment Leak Rate Testing Program* is a performance monitoring program.

One of the conditions of all operating licenses for water-cooled power reactors is that Containment shall meet the leakage test requirements set forth in 10 CFR Part 50, Appendix J. The purposes of these tests are to assure that:

(a) leakage through the (1) containment and (2) systems and components penetrating containment shall not exceed allowable leakage rate values specified in the Technical Specifications or associated bases, and
(b) periodic surveillances of containment penetrations and isolation valves are performed.

The *Containment Leak Rate Testing Program* contains three types of tests: Type A, which are integrated leak rate tests intended to measure the overall leakage rate of the containment; Type B, which are tests intended to measure leakage of containment penetrations whose design incorporates resilient seals and gaskets including airlock door seals and equipment hatch gaskets; and Type C, which are tests to measure containment isolation valve leakage.

Of these three tests, only Type A and Type B are credited for license renewal. The Type A tests would detect severe corrosion of containment pressure boundary steel components that had degraded to the point of allowing leakage at the test's required pressure condition. The *Containment Leak Rate Testing Program* is implemented per Technical Specifications 3.6.1, *Containment*, and 5.5.2, *Containment Leakage Rate Testing Program* [References B - 16 and B - 17].

**Scope –** The scope of the *Containment Leak Rate Testing Program* includes all pressure boundary components including the steel containment vessel, mechanical penetrations, bellows, electrical penetrations, airlocks, hatches, and flanges.

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

**Parameters Monitored or Inspected –** The parameter monitored by the *Containment Leak Rate Testing Program* is the containment leakage rate. The testing is performed to identify leakage that could indicate loss of material and cracking.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending,** the *Containment Leak Rate Testing Program* will detect leakage which would indicate loss of material and cracking prior to loss of structure or component intended functions.

**Monitoring & Trending –** Aging effects are detected through overall leakage during the Type A tests combined with local leakage testing of the penetrations, bellows, and hatches during the Type B tests. The Type A test is performed once every ten years in accordance with Option B as described in NRC Regulatory Guide 1.163 [Reference B - 23].

The Type B test is performed in accordance with 10 CFR 50, Appendix J Option A requirements. (By letter dated March 1, 2001, Catawba requested an amendment to its Facility Operating Licenses to allow implementation of Option B for Type B and C testing. This license amendment request is under NRC staff review at the time of the submittal of this license renewal application.)

All bellows are leak tested in accordance with Technical Specification surveillance requirements. The parameters to be monitored are leakage rates through the primary containment and the systems and components penetrating primary containment. Unacceptable conditions are identified for corrective action and/or further evaluation.

*Containment Leak Rate Testing Program* maintains data on the components such as leakage rates, total overall leakage, and containment bypass leakage to ensure that the leakage remains below the allowable limits.

**Acceptance Criteria –** The acceptance criteria are defined in Technical Specifications. The containment leakage rate acceptance criterion is less than or equal to 1.0 $L_a$. $L_a$ is the maximum allowable containment leakage rate at the calculated peak containment internal pressure ($P_a$) resulting from the limiting design basis LOCA. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criterion is less than 0.75 $L_a$ for Type A tests.

As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, Option A, leakage test is required to be less than 0.6 $L_a$ for combined Type B and C leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit less than or equal to 1.0 $L_a$.

The space between dual-ply bellows shall be subjected to a low pressure leak test with no detectable leakage. Otherwise, the assembly must be tested with the containment side of the bellows assembly pressurized to $P_a$ and the acceptance criteria is based on the combined leakage rate for all reactor building bypass leakage paths less than or equal to 0.07 $L_a$.

**Corrective Action & Confirmation Process –** Areas that do not meet the acceptance criteria are accepted by engineering evaluation or corrected by repair/replacement activities. When corrective actions are implemented to repair a condition outside the acceptance criteria, confirmation by additional leak rate testing is required to confirm that the deficiency has been corrected.

**Administrative Controls –** The *Containment Leak Rate Testing Program* is governed by Technical Specifications. The *Containment Leak Rate Testing Program* is implemented by plant procedures as required by Technical Specification 5.4.

**Operating Experience –** Numerous Type A and Type B tests have been performed at McGuire and Catawba over the course of operation. Results have shown that all containment steel components such as the steel containment vessel and flued head penetrations have successfully passed the Type A tests. Results of previous Type B tests have identified leakage of the mechanical bellows as described below.

***McGuire Operating Experience***

McGuire has identified several leaking penetration bellows after twenty years of operation, about half of which are attributable to damage incurred during construction. Some of the original McGuire bellows were repaired/replaced prior to initial plant startup. Main Steam penetration 1M-441 bellows was replaced during refueling outage 1EOC-14 (Spring 2001). The remaining bellows with leakage are within Technical Specification limits. The leakage test results are conservatively added to the overall containment leakage and are included in bypass or non-bypass leakage calculations, as appropriate, with each remaining below allowable Technical Specification limits.

***Catawba Operating Experience***

Catawba has identified a few penetration bellows that failed the low-pressure bellows test. The bellows leakage from these tests was added to the overall leakage and included in the containment bypass leakage calculations. The total overall leakage and containment bypass leakage remains below the allowable Technical Specification limits.

***Conclusion***

The *Containment Leak Rate Testing Program* has been demonstrated to be effective in detecting loss of material of pressure boundary components and cracking of the bellows through measurement of leakage. The *Containment Leak Rate Testing Program* described above is equivalent to the *Containment Leak Rate Testing Program* described and evaluated in NUREG-1723, Section 3.3 [Reference B - 5]. Based on the above review, it is reasonable to expect the continued implementation of the *Containment Leak Rate Testing Program* will detect loss of material and cracking through measurement of leakage such that the intended functions of the steel containment vessel, penetrations, bellows and hatches will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

## B.3.9 CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE PENETRATIONS INSPECTION PROGRAM

*Note: The CONTROL ROD DRIVE MECHANISM AND OTHER VESSEL CLOSURE PENETRATION INSPECTION PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted*

The purpose of the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* is to manage cracking of nickel based alloy reactor vessel head penetrations exposed to the borated water environment to assure that the pressure boundary function is maintained during the period of extended operation. The *Fluid Leak Management Program*, which performs walkdowns looking for evidence of leakage, and the *Reactor Coolant System Operational Leakage Monitoring Program*, which monitors system leakage are used in conjunction with the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* to manage aging of the reactor vessel head penetrations. This program is a condition monitoring program credited with managing primary water stress corrosion cracking (PWSCC) of high nickel alloy reactor vessel head penetrations and is a complimentary program to the *Inservice Inspection Plan*.

**Scope –** The scope of the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* includes the control rod drive mechanism nozzles and head vent penetrations of each reactor vessel. These penetrations include 78 Control Rod Drive Mechanism (CRDM) type penetrations, and one head vent penetration. The four auxiliary head adapter penetrations on each head are visually inspected as part of the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* and volumetrically examined by the *Inservice Inspection Plan*.

**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 Program* monitors cracking of nickel based alloy nozzles with partial penetration welds in the reactor vessel closure head.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending** below, *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* 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 Program* will inspect the control rod drive mechanism type penetrations, the head vent penetration and the auxiliary head vent penetration. This program will consist of both visual and volumetric examinations.

Visual inspections apply to all penetrations in the reactor vessel head. Visual inspections of all accessible CRDM type penetrations will be completed every refueling outage. During each 10 year ISI interval , insulation is removed and 100% visual inspection of the outside surface of the head will be performed. This inspection will include CRDM type penetrations, auxiliary head adapter penetrations and the head vent.

Volumetric inspections within this program apply to the CRDM type penetrations and the head vent penetration. The auxiliary head adapter penetrations are inspected volumetrically by the *Inservice Inspection Plan*.

Currently, eddy current inspection is used for detection of cracking. A combination of eddy current, ultrasonic, and liquid penetrate will be used for sizing indications. These methods may be updated based on industry experience.

The number of penetrations inspected will be based on both Duke specific experience gained through inspections performed at Oconee and through industry experience on similar Westinghouse plants shared through the Westinghouse Owner's Group Program.

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

Due to length of time in operation, it is expected that Unit 1 results will provide a leading indicator for Unit 2 results at each station. The results of these inspections will form the basis for timing of future inspections. The timing of these inspections may change based on either Duke specific or industry experience.

**Acceptance Criteria –** For the visual inspection, any boron detected on the outside of the vessel head due to penetration leakage is unacceptable.

For the volumetric examination, axial flaws detected during volumetric inspection will be analyzed and accepted via the NUMARC acceptance criteria which was approved by the NRC in their SER dated November 19, 1993. Circumferential flaws will be analyzed and addressed on a case-by-case basis by the NRC [Reference B - 24].

**Corrective Action & Confirmation Process –** For the visual inspection, if leakage is detected the leakpath will be determined and repairs completed. Specific corrective actions and confirmation are implemented in accordance with the corrective action 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 specific *Thermal Fatigue Management Program* described in Section 4.3.1 of this Application. Specific corrective actions and confirmation will be implemented in accordance with the *Thermal Fatigue Management Program*.

**Administrative Controls –** Inspections will be controlled by site specific procedures. Engineering evaluations are performed in accordance with Duke engineering guidelines.

**Operating Experience –** On April 1, 1997 the NRC issued Generic Letter 97-01, *Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations* [Reference B - 25]. Generic Letter 97-01 indicated that the NRC did not object to individual licensees basing their inspection plans for vessel closure head penetrations on an integrated industry program. McGuire and Catawba Nuclear Stations are participants in the WOG generic program to address Generic Letter 97-01 [Reference B - 26].

The programs to address reactor closure head penetrations are based on WCAP-14901, Revision 0, *Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group*. Plants have been placed into three susceptibility groups based on the probability of having a 75% through wall crack. McGuire and Catawba are in the greater than 15 EFPY grouping (would not expect a 75% through wall crack for more than 15 EFPY from January 1, 1997), which reflects the lowest susceptibility to cracking of the CRDM penetrations.

On April 30, 2001 the Nuclear Regulatory Commission issued Information Notice 2001-05, *Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3*. This Information Notice gives details of inspections performed on the reactor vessel head where significant circumferential indications were found. The root cause for the indications was determined to be PWSCC. Duke is evaluating the recent Oconee experience for additional actions that may be taken.

***Conclusion***

The *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* has been demonstrated to be capable OF managing the aging of nickel based alloy reactor vessel head penetrations. The *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* provides reasonable assurance that the aging effects will be managed and that the control rod drive mechanisms and other head penetrations will continue to perform their intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

### B.3.10 CRANE INSPECTION PROGRAM

*Note: The CRANE INSPECTION PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Loss of material has been identified as an aging effect requiring management for crane rails and girders for the period of extended operation. The *Crane Inspection Program* is credited with managing loss of material for the steel rails and girders within the scope of license renewal. This program has been in effect for many years at the Duke nuclear plants and is based on the guidance contained in ANSI B30.2.0 [Reference B - 27] for cranes, ANSI B30.16 [Reference B - 28] for hoists, and the requirements contained in 29 CFR Chapter XVII, 1910.179 [Reference B - 29]. The *Crane Inspection Program* is a condition monitoring 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. Inspection procedures for cranes and hoists are identified in plant procedures and are in accordance with industry standards, plant experience, and other industry experience. Each crane and hoist is subject to several inspections. Prior to initial use, all new, reinstalled, altered, modified, extensively repaired and newly erected cranes are inspected and the results of the inspections are documented. Additional inspections are conducted prior to crane operation, quarterly, and/or annually depending on the specific crane or hoist. The inspection frequencies for the cranes and hoists are based on the guidance provided by ANSI B30.2.0 and ANSI B30.16 and are considered acceptable. Plant experience supports the established frequency as being timely and effective. No actions are taken as part of this program to trend inspection or test results.

**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.

**Operating Experience –**

***McGuire Operating Experience***
McGuire experience has found no adverse aging conditions with crane rails and girders. The significant operating experience history related to cranes dealt with functional issues.

***Catawba Operating Experience***
Catawba experience has found no adverse aging conditions with crane rails and girders. Most issues that were identified were related to electrical equipment associated with the cranes.

***Conclusion***
The *Crane Inspection Program* has been demonstrated to be capable of detecting and managing loss of material. The *Crane Inspection Program* described above is equivalent to the Crane Inspection Program described and evaluated in NUREG-1723, Section 3.8.3.2.1 [Reference B - 5]. Based on the above review, the continued implementation of the *Crane Inspection Program* provides reasonable assurance that loss of material will be managed such that the intended function of the crane and hoist rails and girders will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

## B.3.11 DIVIDER BARRIER SEAL INSPECTION AND TESTING PROGRAM

*Note: The DIVIDER BARRIER SEAL INSPECTION AND TESTING PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The divider barrier in each Reactor Building is the physical boundary that separates upper containment from lower containment. Several Reactor Building internal structures comprise the divider barrier inside the steel Containment. As part of the divider barrier, elastomeric pressure seals are provided at locations where it is necessary to limit potential Ice Condenser bypass leakage subsequent to a postulated pipe rupture or a loss-of-coolant-accident. Cracking and change in material properties are aging effects requiring management for the pressure seals. The *Divider Barrier Seal Inspection and Testing Program* is credited with managing the aging effects for the period of extended operation. The *Divider Barrier Seal Inspection and Testing Program* is a condition monitoring and a performance monitoring program.

**Scope –** The scope of the *Divider Barrier Seal Inspection and Testing Program* includes the following elastomeric seals:

- Ice Condenser Seals
- Control Rod Drive Mechanism Shield Seals
- Operating Deck Hatches and Access Opening Seals
- Pressurizer Enclosure Seals
- Reactor Coolant Pump Hatch Seals
- Steam Generator Enclosure Seals

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

**Parameters Monitored or Inspected –** Parameters monitored by the *Divider Barrier Seal Inspection and Testing Program* are cracking and change in material properties of elastomeric pressure seals.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Divider Barrier Seal Inspection and Testing Program* will detect cracking and change in material property prior to loss of the pressure seals' intended functions.

**Monitoring & Trending –** The *Divider Barrier Seal Inspection and Testing Program* detects aging effects through visual examination of the seals and coupon testing. The inspections and testing are implemented as required by McGuire and Catawba Technical Specifications (SR) 3.6.14.2, 3.6.14.4 and 3.6.14.5.

The ice condenser seals are visually inspected for the presence of holes, ruptures, abrasions, splice separation or gap, and changes in physical appearances such as discoloration, chemical attack, radiation damage, etc. At least 95% of the ice condenser seal is inspected. In addition, the seal mounting hardware is examined for looseness and loss of material due to corrosion. Two seal coupons are removed and tested to verify the tensile strength of the material. The frequency of the inspection of seals and tests of the coupons is once every 18 months as required by Technical Specification Surveillance Requirements 3.6.14.4 and 3.6.14.5.

The remaining divider barrier seals are visually inspected for cracks, defects in the sealing surface, deterioration of the seal material, and detrimental misalignments. The frequency of the inspection is prior to final closure after each opening and once every 10 years for resilient seals as required by Technical Specification Surveillance Requirement 3.6.14.2.

**Acceptance Criteria –** The acceptance criteria for the *Divider Barrier Seal Inspection and Testing Program* are specified in Technical Specification Surveillance Requirements 3.6.14.2, 3.6.14.4 and 3.6.14.5. The minimum tensile strength of both test coupons is specified in Technical Specification 3.6.14.4. The acceptance criteria for the visual inspection are no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or change in physical appearance. Divider barrier seal mounting hardware (i.e. bolts, nuts etc.) must be properly installed, with no unacceptable indication of corrosion.

**Corrective Action & Confirmation Process –** Discrepancies noted during the inspection are documented in the corrective action program in accordance with the implementing procedure. As part of the corrective action program, corrective actions are identified, root cause is addressed, and any applicability to other areas is addressed. Specific corrective actions and confirmatory actions, as needed, are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Divider Barrier Seal Inspection and Testing Program* is governed by Technical Specification 3.6.14. The *Divider Barrier Seal Inspection and Testing Program* is implemented by plant procedures, as required by Technical Specification 5.4, and the plant work management system.

**Operating Experience –**

***McGuire Operating Experience***
Coupon test results have indicated tensile strength above that specified in Technical Specification (SR) 3.6.14.4 with sufficient margin.

McGuire experience has found no adverse aging conditions on divider barrier seals and mounting hardware. The significant operating experience related to divider barrier seals dealt with installation issues.

***Catawba Operating Experience***
The *Divider Barrier Seal Inspection and Testing Program* has been implemented at Catawba since initial operation. Previous inspections have not identified cracking or change in material properties of the seals. Past coupon tests have indicated tensile strength above that specified in Technical Specification Surveillance Requirement (SR) 3.6.14.4 with sufficient margin.

***<u>Conclusion</u>***
The *Divider Barrier Seal Inspection and Testing Program* has been demonstrated to be capable of detecting and managing cracking and change in material property of the seals. Based on the above review, the continued implementation of the *Divider Barrier Seal Inspection and Testing Program* will manage the identified aging effects such that the seals will continue to perform their intended functions consistent with the current licensing basis throughout the period of extended operation (i.e., 20-years from the end of the initial operating license).

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## B.3.12 FIRE PROTECTION PROGRAM

*Note: The FIRE PROTECTION PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The McGuire and Catawba *Fire Protection Program* utilizes the concept of defense-in-depth to achieve a high degree of fire safety. The McGuire and Catawba *Fire Protection Program* contains many activities to achieve defense-in-depth and minimize the impact of a potential fire. Activities credited for license renewal are:

- Fire Barrier Inspections
- Mechanical Fire Protection Component Tests and Inspections (collectively referred to here as the *Fire Protection Program*)

### B.3.12.1 Fire Barrier Inspections

The *Fire Barrier Inspections* are required by Selected Licensee Commitment (SLC) 16.9.5. The *Fire Barrier Inspections* are a condition monitoring program credited with managing the following aging effects for the period of extended operation:

- Loss of material due to corrosion of fire doors.
- Cracking of fire walls.
- Cracking/delamination and separation of fire barrier penetration seals

**Scope –** The scope of the *Fire Barrier Inspections* includes all fire barriers (walls, floor/ceilings) and all sealing devices in fire barrier penetrations (fire doors and penetrations seals).

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

**Parameters Monitored or Inspected –** The *Fire Barrier Inspections* require visual inspections of fire barriers for the following parameters:

- Loss of material due to corrosion of fire doors.
- Cracking of fire walls.
- Cracking/delamination and separation of fire barrier penetration seals.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Fire Barrier Inspections* will detect loss of material for fire doors, cracking for fire barriers, and cracking/delamination and separation of penetration seals prior to loss of structure or component intended functions.

**Monitoring & Trending –** Aging effects are detected through visual examination of the fire barrier, fire doors, and fire barrier penetration seals. All exposed surfaces of each fire barrier is inspected at least once every 18 months per Selected Licensee Commitment 16.9.5. Fire doors are visually inspected and functionally tested at least every 6 months per Selected Licensee Commitment 16.9.5. 10% of each type of fire barrier penetration seal is inspected at least once every 18 months per Selected Licensee Commitment 16.9.5.

**Acceptance Criteria –** The acceptance criterion are specified in McGuire and Catawba procedures.

The acceptance criteria for fire doors are the doors and frame shall be free of holes and punctures.

Acceptance criteria for fire barriers are no visual indications of through-wall holes, cracks or gaps.

The acceptance criteria for fire barrier penetration seals are no visual indications of cracking, shrinkage, or separation of layers of material. In addition, separation from wall and through-holes shall not exceed limits as specified in the procedure.

**Corrective Action & Confirmation Process –** Any abnormal changes or degradation noted during the inspection are investigated through the corrective action program. Corrective actions may include repair or replacement. Specific corrective actions and confirmatory actions, as needed, are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Fire Barrier Inspections* are governed by the Selected Licensee Commitment 16.9.5 and implemented through plant procedures.

**Operating Experience –** A review of *Fire Barrier Inspections* previously conducted at McGuire and Catawba confirms the reasonableness and acceptability of the inspections and their frequency in that degradation of the fire barrier was detected prior to loss of function. Identified degradation has been associated with installation problems and has not been associated with aging. Several letters have been written to the NRC discussing installation deficiencies associated with fire barrier penetration seals [References B - 30, B - 31, and B - 32]. When a deficiency was noted during a triennial fire protection audit, additional fire barrier penetrations were inspected. It was determined that the deficiencies were related to installation problems. Corrective actions included additional inspections, repair, and or replacement activities.

***Conclusion***

The *Fire Barrier Inspections* have been demonstrated to be capable of detecting and managing aging effects of fire barriers. The *Fire Barrier Inspections* described above are equivalent to the Fire Barrier Inspections described and evaluated in NUREG-1723, Section 3.2.4 [Reference B - 5]. Based on the above review, the continued implementation of the *Fire Barrier Inspections* provides reasonable assurance that aging effects will be managed such that the intended functions of the fire barriers will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.12.2 Mechanical Fire Protection Component Tests and Inspections

The purpose of the Mechanical Fire Protection Component Tests and Inspections (collectively referred to herein as the *Fire Protection Program*) is to manage loss of material and fouling of specific components exposed to raw water within the scope of license renewal in the fire protection systems. The program manages loss of material in sprinklers that can affect the pressure boundary and spray functions of the sprinklers. The program also manages fouling of sprinklers, valves at hydrants, and valves at hose racks that can affect the component function. This program is a condition monitoring program that is credited with managing the subject aging effect for brass and bronze materials exposed to a raw water environment.

Fouling is considered an aging effect requiring management for the fire protection systems because of operating experience at McGuire and Catawba. The fire protection systems use lake water as their water source. Fouling has been evidenced in these systems and is an issue that the stations have been working to manage via chemical treatment, testing, and inspections. For the purpose of license renewal, fouling is being applied to the distribution components (sprinklers, hose station valves, and hydrant valves) of the fire protection systems. As evidenced from the description of this program, managing fouling of the distribution components ensures that the system is capable of performing its function of supplying fire suppression water through the distribution components.

**Scope –** The components within the scope of the *Fire Protection Program* are the sprinklers and fire hydrant valves and hose rack valves of the Interior Fire Protection System and Exterior Fire Protection System.

**Preventive Actions –** No actions are taken are taken as part of the *Fire Protection Program* to prevent aging effects or to mitigate aging degradation.

**Parameters Monitored or Inspected –** The *Fire Protection Program* involves visual inspections to verify sprinkler condition and performs flow tests and flushes of the system to verify that blockage of flow will not prevent system function.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending**, the *Fire Protection Program* will detect loss of material and fouling prior to loss of the component function.

**Monitoring & Trending –** The *Fire Protection Program* manages loss of material and fouling of fire protection components by means of visual inspections and system flow tests and flushes.

Loss of material of sprinklers is managed via visual inspections. Sprinklers are visually inspected at least once every 18 months per Selected Licensee Commitment 16.9.2. Additionally, a sample of sprinklers are either inspected or replaced at 50 years of operation.

Fouling of hose station valves, hydrant valves, and sprinklers is managed by various flow tests and flushes performed on the systems. Distribution loops experience high-volume flow when hydrant valves are periodically opened. This procedure is performed for the outside distribution loop every six months and is governed by Selected Licensee Commitment 16.9-1(a)(iii) at Catawba and Testing Requirement (TR) 16.9.1.3 at McGuire. Additional distribution loop flow tests are performed by procedure less frequently.

The integrity of hose station valves and hydrant valves is assured by supplying water to these components. Each hose station valve is opened at least once every three years per Selected Licensee Commitment 16.9-4. Hydrant valves are fully opened every six months. The hydrant tests are not governed by Selected Licensee Commitments, but are performed by procedure.

The integrity of the sprinkler branch lines is assured by performing sprinkler system flow tests every eighteen months. This procedure is performed by fully opening the inspector's test connection valve, which simulates flow from the most hydraulically remote sprinkler head on each system. This test is governed by Selected Licensee Commitment TR 16.9-2(a)(iv)(1) at Catawba. The test is not governed by Selected Licensee Commitments at McGuire but is performed by procedure.

Fouling of sprinkler branch lines that do not receive flow during this test will be managed by a sample disassembly inspection program. 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 disassembled and the piping visually inspected. Subsequent inspections for the period of extended operation will be determined based on inspection results. If fouling is minimal, it is preferable to terminate the sample inspections because draining and filling activities introduces newly oxygenated water to those portions of the system, which could have an adverse effect on corrosion and fouling of the lines.

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

**Acceptance Criteria –** The acceptance criterion for the visual inspections of sprinklers is that an evaluation is performed for any cracks, corrosion, missing pipe hangers, obstructions to sprinkler spray pattern, and other piping abnormalities that are detected. The acceptance criterion for system flushes and flow tests is that water shall flow through the valve to the discharge point with no obvious signs of flow blockage.

**Corrective Action & Confirmation Process –** Applicable Selected Licensee Commitment direct actions to be taken when certain system operability conditions are not met. Any visual indication will be evaluated and specific corrective actions will be taken. Specific corrective actions and confirmatory actions, as needed, are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Fire Protection Program* is governed by the Selected Licensee Commitment 16.9.1, 16.9.2 and 16.9.4 and implemented by plant procedures and work processes. The procedures and work processes provide steps for performance of the activities and require the documentation of the results.

**Operating Experience –** The *Fire Protection Program* is a standardized program that has been regulatory-driven at nuclear power plants for decades. National Fire Protection Association Codes provide standards for ensuring safe, reliable operation of fire protection systems. The Nuclear Regulatory Commission requires certain testing as portions of the plant's licensing basis. These standards and oversights have proven effective means of ensuring that fire protection systems operate effectively for the duration of the plant's life.

*McGuire Operating Experience*
Fouling of the fire protection systems is being minimized by chemical treatment of the water. Additionally, system engineers monitor flow through the system headers and attempt to minimize header flow to reduce internal buildup of corrosion products. Flow tests have not detected unacceptable fouling in other areas where flows are limited. Over the past three

years, sections of piping have been replaced due to due to pin-hole leaks or where fouling has been detected during permitted internal inspections. All corrective actions have been taken prior to loss of component intended function.

*Catawba Operating Experience*
Fouling of the fire protection systems is being minimized in recent years by chemical treatment of the water. Additionally, system engineers monitor flow through the system headers and attempt to minimize header flow to reduce internal buildup of corrosion products. Due to corrosion product buildup in the system, the Interior Fire Protection System auxiliary building header was cleaned in 1996. All corrective actions have been taken prior to loss of component intended function.

***Conclusion***
The *Fire Protection Program* has been demonstrated to be capable of managing loss of material and fouling in fire protection system components. The *Fire Protection Program* is similar to the corresponding program described and evaluated in NUREG-1723, Section 3.2.4 [B - 5]. Based on the above review, the continued implementation of the *Fire Protection Program* provides reasonable assurance that loss of material and fouling in fire protection system components will be managed and that the subject components will continue to perform their intended functions(s) during the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.13 FLOOD BARRIER INSPECTION

*Note: The FLOOD BARRIER INSPECTION is applicable only to McGuire Nuclear Station. At Catawba, the flood barriers are inspected as part of the INSPECTION PROGRAM FOR CIVIL ENGINEERING STRUCTURES AND COMPONENTS.*

Cracking and change in material properties of flood seals have been identified as aging effects requiring programmatic management for the period of extended operation. The *Flood Barrier Inspection Program* is credited for managing cracking and separation of the elastomeric flood seals to ensure that safety-related equipment is protected from floods and flooding flow paths such that no equipment safety-related intended functions or station safe shutdown capabilities are adversely impacted. The *Flood Barrier Inspection Program* is a condition monitoring program.

In 1987 the NRC issued information notice Information Notice 87-49, *Deficiencies in Outside Containment Flooding Protection* [Reference B - 33]. Information Notice 87-49 notified all nuclear power reactor facilities of a potentially significant problem pertaining to the flooding of safety-related equipment as a result of the inadequate design, installation, and maintenance of features intended to protect against flooding. Suggestions contained in the information notice did not constitute NRC requirements therefore no specific action or written response was required.

As a result of Information Notice 87-49, McGuire initiated a *Flood Barrier Inspection Program*. McGuire *Flood Barrier Inspection Program* ensures flood protection features outside containment are properly installed and maintained.

**Scope –** The scope of the *Flood Barrier Inspection Program* is McGuire Units 1 and 2 internal elastomeric flood seals.

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

**Parameters Monitored or Inspected –** The *Flood Barrier Inspection Program* inspects the elastomeric flood seals for cracking and change in material properties.

**Detection of Aging Effects –** In accordance with information provided in Monitoring & Trending, the *Flood Barrier Inspection Program* will detect cracking and change in material properties of elastomeric flood seals prior to loss of structure or component intended functions.

**Monitoring & Trending –** Aging effects for flood seals are detected by visual inspection. The inspection is performed at an 18 months frequency. No actions are taken as part of this program to trend inspection or test results.

**Acceptance Criteria –** Acceptance criteria are no unacceptable visual indications of cracking and change in material properties of elastomeric flood seals that would result in loss of intended function.

**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, as needed, are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Flood Barrier Inspection Program* is implemented through the plant work management system using model work orders.

**Operating Experience –** A review of past inpsections of the *Flood Barrier Inspection Program* conducted at McGuire confirms the reasonableness and acceptability of the inspection frequency in that degradation of the flood seal is detected prior to loss their intended functions. Previous corrective actions included caulking of minor gaps, door adjustment for seal tightness, resealing of penetrations, replacement of a few boot seals, and replacement of compression straps. The majority of work requests generated were for re-caulking and replacement of boot seals due to torn fabric.

***<u>Conclusion</u>***

The *Flood Barrier Inspection Program* has been demonstrated to be capable of detecting and managing cracking and change in material properties of the elastomeric flood seals. Based on the above review, the continued implementation of the *Flood Barrier Inspection Program* provides reasonable assurance that cracking and change in material properties will be managed such that the intended functions of the elastomeric flood seals will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.14 FLOW ACCELERATED CORROSION PROGRAM

*Note: The FLOW ACCELERATED CORROSION PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Flow Accelerated Corrosion Program* is to manage loss of material of carbon steel components located in systems within the scope of license renewal that have been identified as susceptible to flow accelerated corrosion, also called erosion-corrosion. The *Flow Accelerated Corrosion Program* is a condition monitoring program that monitors specific component or material parameters to detect the presence and assess the extent of flow accelerated corrosion.

**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 (CNS only)
- Auxiliary Steam
- Boron Recycle
- Feedwater
- Liquid Radwaste (CNS)
- Liquid Waste Recycle (MNS)
- Liquid Waste Monitor and Disposal (MNS only)
- Steam Generator Blowdown Recycle (CNS only)
- Turbine Exhaust (MNS only)

The only portions of Boron Recycle, Liquid Radwaste (CNS), Liquid Waste Recycle (MNS), and Liquid Waste Monitor and Disposal (MNS only) 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 B - 34]. Component wall thickness is measured using volumetric examinations such as ultrasonic testing and radiography. Visual examinations are also employed when access to interior surfaces is allowed by component design. Component wall thickness acceptability is judged in accordance with the McGuire and Catawba component design code of record.

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

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

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

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

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

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

**Operating Experience –** A review of inspection data for Steam Generator Blowdown and Recycle (Catawba only), and Auxiliary Steam supplies shows minimal loss of material at the inspection locations. Auxiliary Feedwater (particularly Catawba Unit 2) has revealed loss of material in several locations that has resulted in material replacement in significant lengths of piping, illustrating that the program is effective in managing these components. The carbon steel that remains in the system is monitored and evaluated as described above. Degradation in the Feedwater System has been limited to areas associated with localized velocity. These sections of piping have been replaced with wear-resistant material. Ultrasonic scanning has been performed on the Turbine Exhaust (McGuire only) section of piping and minimal loss of material was detected. No component failures due to flow accelerated corrosion attributed to an inadequate *Flow Accelerated Corrosion Program* have occurred in these systems. This excellent operating experience demonstrates that the *Flow Accelerated Corrosion Program* when continued into the period of extended operation will be effective in managing flow accelerated corrosion to ensure the component intended pressure boundary function under all current licensing basis design conditions.

***Conclusion***

The *Flow Accelerated Corrosion Program* has been demonstrated to be capable of assessing and managing the aging of components within the scope of license renewal that are susceptible to flow accelerated corrosion. The *Flow Accelerated Corrosion Program* is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.7.3.2 [Reference B - 5]. Based on the above review, the continued implementation of the *Flow Accelerated Corrosion Program* provides reasonable assurance that loss of material due to flow accelerated corrosion will be managed and that the subject components will continue to perform their intended functions(s) during the period of extended operation (i.e., 20-years from the end of the initial operating license).

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## B.3.15 FLUID LEAK MANAGEMENT PROGRAM

*Note: The FLUID LEAK MANAGEMENT PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The *Fluid Leak Management Program* is a comprehensive program that contains many activities to manage leakage for the entire plant. For license renewal, the purpose of the *Fluid Leak Management Program* is to manage loss of material due to boric acid wastage of mechanical and structural components within the scope of license renewal that are constructed of carbon steel, low alloy steel, and other susceptible materials that are located in the Auxiliary and Reactor Buildings. The program also manages boric acid intrusion of electrical equipment that is located in proximity to borated water systems.

The *Fluid Leak Management Program* is a mitigation program that contains activities developed as part of Duke's response to NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants* [Reference B - 35]. The program identifies leaks from borated water systems and initiates investigation and repair.

**Scope –** The scope of the *Fluid Leak Management 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.

Mechanical components in the following systems within the scope of license renewal are managed by the *Fluid Leak Management Program*:

- Annulus Ventilation
- Auxiliary Building Ventilation
- Auxiliary Feedwater
- Auxiliary Steam
- Boron Recycle
- (Building) Heating Water
- Chemical and Volume Control
- Component Cooling
- Condensate (CNS only)
- Condensate Storage (CNS only)
- Containment Air Release and Addition (CNS only)
- Fire Protection (Interior and Exterior)
- Fuel Handling Area (or Building) Ventilation
- Groundwater Drainage
- Hydrogen Bulk Storage
- Ice Condenser Refrigeration
- Instrument Air (MNS only)
- Liquid Radwaste (CNS)
- Liquid Waste Monitor and Disposal (MNS only)
- Liquid Waste Recycle (MNS)
- Main Steam
- Main Steam (Supply) to Auxiliary Equipment
- Main Steam Vent to Atmosphere

- Containment Air Return Exchange and Hydrogen Skimmer
- Containment Hydrogen Sample and Purge (CNS only)
- Containment Purge (Ventilation)
- Containment Spray
- Containment Ventilation Cooling Water (MNS only)
- Control Area Chilled Water
- Control (Room) Area Ventilation
- Feedwater
- (Feedwater Pump) Turbine Exhaust
- Nuclear Service Water
- Reactor Coolant
- Recirculated Cooling Water (CNS only)
- Residual Heat Removal
- Safety Injection
- Spent Fuel Cooling
- Steam Generator Blowdown (Recycle)
- Steam Generator Wet Lay-up Recirculation
- Turbine Building Sump Pump System (CNS only)
- Waste Gas

**Preventive Actions –** The programmatic implementation of the *Fluid Leak Management Program* is accomplished through visual surveillance and systematic trending of findings. All active leaks are monitored on an appropriate frequency depending on accessibility and rate of leakage. Timely action serves to mitigate loss of material due to boric acid wastage.

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

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

**Monitoring & Trending –** Walkdowns of the Auxiliary and Reactor Buildings are conducted at the start of each refueling outage for the purpose of identifying leakage or evidence of leakage from borated water systems. Information on all 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 operation practices.

**Acceptance Criteria –** The external surfaces of structures and components within the scope of the *Fluid Leak Management 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 *Fluid Leak Management 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 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 *Fluid Leak Management Program* or the corrective action program. These programs apply to all structures and components within the scope of the *Fluid Leak Management Program*.

**Administrative Controls –** Nuclear System Directive NSD-104, *Housekeeping, Materiel Condition and Foreign Material Exclusion* [Reference B - 36] establishes high-level expectations in the areas of housekeeping, materiel condition and foreign material exclusion at Duke Power Company's nuclear plants. The *Fluid Leak Management Program* is described and controlled by Nuclear System Directive NSD-413, *Fluid Leak Management Program* [Reference B - 37]. Inspections, evaluations, and clean up of boric acid are implemented by controlled plant procedures. Guidance for the disposition of boric acid leakage is provided in an engineering procedure.

**Operating Experience –** The Fluid Leak Management Databases for Catawba and McGuire were searched for boric acid leaks that have been identified through the implementation of the *Fluid Leak Management Program*. The majority of the leaks were identified as inactive with only evidence of past leakage. No evidence of loss of material has been found on either the leaking components or on other components in the area of any identified leak. Corrective actions, which were implemented through the Work Management System, included cleaning the area around the leak and either tightening bolted closures or containing the leak. The frequencies of inspections have been demonstrated to be adequate to identify leaks before any loss of material is a concern, and thus before loss of component intended function(s).

***Conclusion***

The *Fluid Leak Management Program* has been demonstrated to be capable of identifying leaks from borated water systems, and subsequently managing the effects of boric acid intrusion and boric acid wastage. The *Fluid Leak Management Program* described above is equivalent to the Boric Acid Wastage Surveillance Program described and evaluated in NUREG-1723, Section 3.2.1 [Reference B - 5]. Based on the above review, the continued implementation of the *Fluid Leak Management Program* provides reasonable assurance that loss of material due to boric acid wastage will be managed, and that the subject structures and components will continue to perform their intended function(s) during the period of extended operation (i.e., 20-years from the end of the initial operating license).

## B.3.16 GALVANIC SUSCEPTIBILITY INSPECTION

*Note: The GALVANIC SUSCEPTIBILITY INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Galvanic Susceptibility Inspection* is to characterize any loss of material due to galvanic corrosion from exposure to gas, unmonitored treated water, and raw water environments. The gas environment is a mixture of hydrogen, nitrogen, oxygen, fission product gases and water vapor. An unmonitored treated water environment is one that is not within the scope of the site Chemistry Control Program. Uncertainty exists as to whether exposure of galvanic couples to these environments could cause loss of material due to galvanic corrosion such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect components to detect the presence and extent of any loss of material due to galvanic corrosion. The *Galvanic Susceptibility Inspection* is a one-time 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 McGuire and Catawba systems:

- Condenser Circulating Water
- Containment Ventilation Cooling Water (MNS Only)
- Diesel Generator Room Sump Pump
- Exterior Fire Protection
- Interior Fire Protection
- Liquid Radwaste (CNS Only)
- 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 each site using a volumetric examination technique. As an alternative, visual examination will be used should access to internal surfaces become available. The susceptibility and aggressiveness of galvanic corrosion is determined by the material position on the galvanic series and the corrosiveness of the surrounding environment. Since inspection of all couples is impractical, certain locations will be inspected where galvanic corrosion is more likely to occur. These more susceptible locations are where the materials are the farthest apart on the galvanic series surrounded by the most corrosive of the three environments identified above. For the couples noted above, carbon steel and stainless steel are the farthest apart on the galvanic series and raw water is the most corrosive environment. An inspection of selected locations of carbon steel-stainless steel couples in raw water will determine whether loss of material due to galvanic corrosion will be an aging effect of concern for the period of extended operation. A sentinel population of carbon steel-stainless steel couples located in raw water systems will be inspected. Engineering practice at Duke for the past several years has been to use stainless steel as a replacement material in raw water systems. Since engineering practice will continue to use stainless steel as an acceptable substitute material, the size of the sentinel population will be dependent on the number of susceptible locations at the time of the inspection. The results of this inspection will be applied to all galvanic couples in the systems listed in the **Scope** attribute above.

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

**Operating Experience –** The *Galvanic Susceptibility Inspection* is a one-time inspection activity for which there is no operating experience. However, an equivalent inspection was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

***Conclusion***

The *Galvanic Susceptibility Inspection* described above is equivalent to the Galvanic Susceptibility Inspection described and evaluated in NUREG-1723, Section 3.2.9 [Reference B - 5]. Based on the above review, implementation of the *Galvanic Susceptibility Inspection* will adequately verify that no need exists to manage aging effects on the component or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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## B.3.17 HEAT EXCHANGER ACTIVITIES

### B.3.17.1 Component Cooling Heat Exchangers

#### B.3.17.1.1 PERFORMANCE TESTING ACTIVITIES – COMPONENT COOLING HEAT EXCHANGERS

*Note: The PERFORMANCE TESTING ACTIVITIES – COMPONENT COOLING HEAT EXCHANGERS are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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.

**Scope –** The scope of the *Performance Testing Activities – Component Cooling Water Exchangers* is the McGuire and Catawba component cooling heat exchanger tubes.

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

**Parameters Monitored or Inspected –** The *Performance Testing Activities – Component Cooling Heat Exchangers* involves monitoring of flow capacity by performance of a differential pressure test to provide an indication of fouling.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending**, the *Performance Testing Activities – Component Cooling Heat Exchangers* will detect fouling prior to loss of the component heat transfer function.

**Monitoring & Trending –** The *Performance Testing Activities – Component Cooling Heat Exchangers* measures the pressure drop through the heat exchanger tubes. An increase in the pressure drop indicates the presence of fouling.

At McGuire, the pressure drop through the heat exchanger tubes is monitored on a continuous basis. The pressure drop is monitored against the acceptance criteria. Continuous monitoring against the acceptance criteria does not require trending.

At Catawba, a periodic differential pressure test is performed. The test results are trended against a baseline value for indication of tube cleanliness. The frequency of testing at Catawba permits the results of the testing to be trended in order to determine when corrective action is required.

**Acceptance Criteria –** At McGuire, the acceptance criteria are in the form of alarm points. An alarm point is provided for a high differential pressure and for a high-high differential pressure.

For Catawba, the acceptance criterion is in the form of a flow resistance factor value. The acceptable value is based on a design resistance factor for “clean” heat exchanger tubes.

**Corrective Action & Confirmation Process –** If the heat exchangers fail to meet the acceptance criteria, then corrective actions such as flushing or cleaning are undertaken. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Performance Testing Activities – Component Cooling Heat Exchangers* are controlled by plant procedures. The procedures provide steps for performance of the activities and require the documentation of the results.

**Operating Experience –** Operating experience associated with the *Performance Testing Activities – Component Cooling Heat Exchangers* has demonstrated that monitoring of flow through the heat exchangers provides adequate information on the extent of fouling present in the tubes to predict when corrective action is required. Corrective action in the form of flushing or tube cleaning, for example, is performed before the heat transfer function of the heat exchanger tubes is degraded below its required capacity.

Experience has proven that both of these techniques permit the fouling to be monitored and any required corrective actions to be performed prior to the heat transfer function being degraded below acceptable limits.

The results of trending for the heat exchanger tube fouling have resulted in the performance of cleaning activities. Different types of cleaning mechanisms have been tested (i.e., darts, brushes, high pressure water laze, etc.) in order to maximize the effectiveness of the cleaning. Cleaning activities have restored the condition of the tube surfaces by removal of fouling materials. The length of time between required cleanings has been trended in order to determine the most effective cleaning process and methods.

*McGuire Operating Experience*
Experience with flow monitoring at McGuire has indicated that the alarm point setting permits action before the differential pressure limit is reached. The combination of high velocity flushes and better cleanings during outages have almost eliminated on-line cleaning of the heat exchanger tubes.

*Catawba Operating Experience*
Experience with the flow tests at Catawba has indicated that the stainless steel tubes foul slightly faster than the original brass tubes. High velocity flushing every six to eight weeks has been used and been found potentially effective in reducing fouling and prolonging heat exchanger service between tube cleanings.

***Conclusion***
The *Performance Testing Activities – Component Cooling Heat Exchangers* have been demonstrated to be capable of managing fouling in heat exchanger tubes. The implementation of the *Performance Testing Activities – Component Cooling Heat Exchangers* will adequately manage fouling and will provide reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.1.2 HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – COMPONENT COOLING

*Note: The HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – COMPONENT COOLING are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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.

**Scope –** The scope of the *Heat Exchanger Preventive Maintenance Activities – Component Cooling* are the McGuire and Catawba component cooling heat exchanger tubes, tubesheets, and channel heads.

**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 *Heat Exchanger Preventive Maintenance Activities – Component Cooling* inspects the heat exchanger tubes, tubesheet, and channel head surfaces for loss of material.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending**, the *Heat Exchanger Preventive Maintenance Activities – Component Cooling* will detect loss of material due to crevice, galvanic, general, pitting, and microbiologically influenced corrosion and particle erosion prior to loss of the component pressure boundary function.

**Monitoring & Trending –** The *Heat Exchanger Preventive Maintenance Activities – Component Cooling* performs eddy current testing on the heat exchanger tubes to measure wall thickness in order to detect areas with loss of material. Trending is performed by the accountable engineer in order to predict a heat exchanger replacement or repair schedule. Non-destructive testing is performed on approximately 50% of the tubes of each heat exchanger as needed based on operating experience and engineering evaluation of test data.

With the exception of one Catawba heat exchanger that has no coated components, loss of material of the tube sheets and channel heads of all component cooling heat exchangers is managed by a visual inspection of the protective coatings to assure the integrity of the underlying base metal. This inspection is performed at least once every two years. The tubesheet and channel heads of the component cooling heat exchangers are coated with a high solids epoxy. The coating inspection specifically identifies rust blooms, which indicate a coating defect and corrosion of the base metal. No actions are taken as part of this visual inspection to trend results.

One Catawba component cooling heat exchanger does not currently have any coatings applied to the tubesheets or channel heads. These parts of the heat exchanger are monitored by ultrasonic testing to detect loss of material and the results trended. This inspection is performed as required based on trending results.

**Acceptance Criteria –** The acceptance criterion for the *Heat Exchanger Preventive Maintenance Activities – Component Cooling* is no unacceptable loss of material of the tubes, tubesheets, and channel heads that could result in a loss of the component intended function(s) as determined by engineering evaluation.

**Corrective Action & Confirmation Process –** Engineering evaluation is performed to determine whether the tube integrity and coating and base metal continue to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Heat Exchanger Preventive Maintenance Activities – Component Cooling* is controlled by plant procedures and work processes. The procedures and work processes provide steps for the performance of the activity and require the documentation of the results.

**Operating Experience –** Operating experience associated with the *Heat Exchanger Preventive Maintenance Activities – Component Cooling* has demonstrated that the eddy current testing provides adequate information on the extent of wall loss present in the heat exchanger tubes to predict when corrective action is required. Corrective action in the form of tube plugging, for example, is performed before the loss of the component intended function. Plant operating experience has demonstrated that measurement and trending of tube wall thickness provides an accurate indication of material condition.

Operating experience associated with the *Heat Exchanger Preventive Maintenance Activities – Component Cooling* has demonstrated that protective coatings are effective in preventing loss of material on the tube sheets and channel heads. Inspection of the coatings ensures that the protective features of the coatings are maintained intact. Plant operating experience has demonstrated that visual inspection of the coatings provides an accurate indication of material condition.

Experience prior to application of the coatings and with the tube sheets and channel heads that have not been coated indicates that loss of material may occur without protective coatings. Measurement and trending of tube sheet and channel head wall thickness using ultrasonic technique provides an accurate indication of material condition. The frequency of monitoring permits the results to be trended in order to determine when corrective action is required. Experience has proven that this technique permits the loss of material to be trended and any required corrective actions to be performed before the loss of component intended function.

***Conclusion***

The *Heat Exchanger Preventive Maintenance Activities – Component Cooling* have been demonstrated to be capable of managing loss of material in heat exchanger tubes, tube sheets, and channel heads. Based on the above review, the implementation of the preventive maintenance activities will adequately manage loss of material in heat exchanger tubes, tube sheets, and channel heads and will provide reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.2 Containment Spray Heat Exchangers

#### B.3.17.2.1 PERFORMANCE TESTING ACTIVITIES – CONTAINMENT SPRAY HEAT EXCHANGERS

*Note: The PERFORMANCE TESTING ACTIVITIES – CONTAINMENT SPRAY HEAT EXCHANGERS is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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.

**Scope** – The scope of the *Performance Testing Activities – Containment Spray Heat Exchangers* is the McGuire and Catawba containment spray heat exchanger tubes.

**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 *Performance Testing Activities – Containment Spray Heat Exchangers* involves monitoring of heat transfer capability by performance of a heat capacity test.

**Detection of Aging Effects** – In accordance with the information provided under **Monitoring & Trending**, the *Performance Testing Activities – Containment Spray Heat Exchangers* will detect fouling prior to loss of the component intended function(s).

**Monitoring &Trending** – The *Performance Testing Activities – Containment Spray Heat Exchangers* involves calculation of a raw water fouling factor using tube and shell side inlet and outlet temperatures and flow. The results of the fouling factor calculation are trended against a baseline value for indication of tube (heat transfer surface) cleanliness. The procedures are performed on each of the Containment Spray heat exchangers annually at Catawba and every three years at McGuire. Information provided under **Operating Experience** justifies the extended frequency at McGuire.

**Acceptance Criteria** – The acceptance criteria of the *Performance Testing Activities – Containment Spray Heat Exchangers* are established by heat removal capacity calculations maintained by the accountable engineer. The comparison of the calculated to the measured heat removal capacity must ensure that the heat exchangers are able to perform their design basis function.

**Corrective Action & Confirmation Process** – If the heat exchangers fail to meet the acceptance criteria, then corrective actions such as cleaning are undertaken. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls** – The *Performance Testing Activities – Containment Spray Heat Exchangers* is controlled by plant procedures. These procedures provide steps for performance of the activities and require documentation of the results.

**Operating Experience** – Operating experience has demonstrated that heat capacity tests provide adequate indication to predict when corrective action is required for heat transfer surface fouling. Corrective action in the form of tube cleaning, for example, is performed before the loss of the component intended function. Placing the heat exchangers in wet lay-up several years ago has minimized buildup of fouling materials on the tubes. The wet lay-up has proven so successful at McGuire that the frequency of heat capacity testing has been extended to a three-year frequency. Experience has shown that a three-year frequency allows for timely corrective action. Corrective action in the form of tube cleaning, for example, is performed before the heat transfer function of the heat exchanger tubes is degraded below its required capacity.

***Conclusion***

The *Performance Testing Activities – Containment Spray Heat Exchangers* have been demonstrated to be capable of managing fouling of heat exchanger tubes. The implementation of the preventive maintenance activities will adequately manage fouling of heat exchanger tubes and will provide reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.2.2 HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – CONTAINMENT SPRAY

*Note: The HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – CONTAINMENT SPRAY is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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.

**Scope –** The scope of the *Heat Exchanger Preventive Maintenance Activities – Containment Spray* is the McGuire and Catawba containment spray heat exchanger tubes.

**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 *Heat Exchanger Preventive Maintenance Activities – Containment Spray* inspects the heat exchanger tubes to provide an indication of loss of material.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending**, the *Heat Exchanger Preventive Maintenance Activities – Containment Spray* will detect loss of material due to crevice, pitting and microbiologically influenced corrosion prior to loss of the component intended function(s).

**Monitoring & Trending –** The *Heat Exchanger Preventive Maintenance Activities – Containment Spray* performs eddy current testing on the heat exchanger tubes to measure wall thickness in order to detect areas with loss of material. At Catawba, non-destructive testing (NDT) is performed on the perimeter tubes of each containment spray heat exchanger at least every five years. Analysis is required following each NDT to determine the need for further testing, replacement, or repair. The perimeter tubes comprise approximately 15% of the total tubes.

At McGuire, NDT is performed on each heat exchanger as needed based on operating experience and engineering evaluation of test data. Information provided under **Operating Experience** justifies the as-needed frequency as McGuire.

**Acceptance Criteria –** The acceptance criterion for the *Heat Exchanger Preventive Maintenance Activities – Containment Spray* is no unacceptable loss of material of the tubes that could result in a loss of the component intended function(s) as determined by engineering evaluation.

**Corrective Action & Confirmation Process –** Engineering evaluation is performed to determine whether the tube integrity continues to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Heat Exchanger Preventive Maintenance Activities – Containment Spray* is controlled by plant procedures and work processes. The procedures and work process provide steps for performance of the activities and require documentation of the results.

**Operating Experience –** Operating experience associated with the *Heat Exchanger Preventive Maintenance Activities – Containment Spray* has demonstrated that the eddy current testing provides adequate information on the extent of wall loss present in the heat exchanger tubes to predict when corrective action is required. Corrective action in the form of tube plugging, for example, is performed before the loss of the component intended function.

Some tube plugging has occurred, particularly early in service life. At Catawba, tube plugging rate has been essentially flat for the past several years due to operational improvements, including placing the heat exchangers in wet lay-up. The wet lay-up has proven so successful at McGuire that most recent test results indicate negligible tube wall degradation over several years.

***Conclusion***

The *Heat Exchanger Preventive Maintenance Activities – Containment Spray* have been demonstrated to be capable of managing loss of material in heat exchanger tubes. The implementation of the preventive maintenance activities will adequately manage loss of material in heat exchanger tubes and will provide reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.3 Diesel Generator Engine Cooling Water Heat Exchangers

#### B.3.17.3.1 PERFORMANCE TESTING ACTIVITIES – DIESEL GENERATOR ENGINE COOLING WATER HEAT EXCHANGERS

*Note: The PERFORMANCE TESTING ACTIVITIES – DIESEL GENERATOR ENGINE COOLING WATER HEAT EXCHANGERS is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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

**Scope –** The scope of the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* is the tubes of the following:

- Diesel Generator Engine Cooling Water Heat Exchangers (MNS only)
- Diesel Generator Engine Jacket Water Coolers (CNS only)

*Note: These components serve the same function at both plants, but have different names because of the different diesel suppliers.*

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

**Parameters Monitored or Inspected –** At McGuire, the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* involves monitoring of flow capacity by performance of a differential pressure test to provide an indication of fouling.

At Catawba, the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* involves monitoring of heat transfer capability by performance of a heat capacity test to provide an indication of fouling.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending**, the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* will detect fouling prior to loss of the component intended function(s).

**Monitoring & Trending –** Due to different system design features at McGuire and Catawba, different parameters are more appropriately monitored to manage fouling of the heat exchanger tubes.

At McGuire, the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* involves measurement of a differential pressure across the raw water side of the heat exchangers every six months. Differential pressure provides a direct indication of fouling of the heat exchanger tubes.

At Catawba, a heat capacity test computes a tube side fouling factor using tube and shell side inlet and outlet temperatures and flows every six months. Heat capacity provides a direct indication of fouling of the heat exchanger tubes.

**Acceptance Criteria –** At McGuire, the acceptance criterion for the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* is the established differential pressure value that ensures fouling does not prevent the heat exchangers from performing their design basis function.

At Catawba, the acceptance criteria for the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* are established by engineering calculation. The comparison of the test results to the acceptance criteria ensures fouling does not prevent the heat exchangers from performing their design basis function.

**Corrective Action & Confirmation Process –** If the heat exchangers fail to meet the acceptance criteria, then corrective actions such as cleaning are undertaken. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* are controlled by plant procedures. These procedures provide steps for performance of the activities and require documentation of the results.

**Operating Experience –** Operating experience associated with the *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* has demonstrated that fouling factor and tube side differential pressure provide adequate indication to predict when corrective action is required for heat transfer surface fouling. Corrective action in the form of tube cleaning, for example, is performed before the heat transfer function of the heat exchanger tubes is degraded below its required capacity.

With relatively low in-service duration and good valve isolation, the diesel generator engine cooling water heat exchangers usually do not accumulate large amounts of fouling materials on internal tubing surfaces.

***Conclusion***

The *Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers* have been demonstrated to be capable of managing fouling of heat exchanger tubes. The implementation of the preventive maintenance activities will adequately manage fouling and will provide reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.3.2 HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – DIESEL GENERATOR ENGINE COOLING WATER

*Note: The HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – DIESEL GENERATOR ENGINE COOLING WATER is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* is to manage loss of material for parts of the diesel generator engine cooling water heat exchanger exposed to raw water. The *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary function. The program is credited with managing the subject aging effects for brass and copper heat exchanger tubes.

**Scope –** The scope of the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* is the tubes of the following:

- Diesel Generator Engine Cooling Water Heat Exchangers (MNS only)
- Diesel Generator Engine Jacket Water Coolers (CNS only)

Note: These components serve the same function at both plants, but have different names because of the different diesel suppliers.

**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 *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* inspects the heat exchanger tubes to provide an indication of loss of material.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending**, the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* will detect loss of material due to crevice, general, pitting, and microbiologically influenced corrosion and loss of material due to particle erosion prior to loss of the component intended function(s).

**Monitoring & Trending –** The *Heat Exchanger Preventive Maintenance Activities- Diesel Generator Engine Cooling Water* performs eddy current testing on the heat exchanger tubes to measure wall thickness in order to detect areas with loss of material. Trending is performed by the accountable engineer in order to predict a heat exchanger replacement or repair schedule. Non-destructive testing is performed on approximately 50% of the tubes of each heat exchanger as needed based on operating experience and engineering evaluation of test data.

**Acceptance Criteria –** The acceptance criterion for the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* is no unacceptable loss of material of the tubes that could result in a loss of the component intended function(s) as determined by engineering evaluation.

**Corrective Action & Confirmation Process –** Engineering evaluation is performed to determine whether the tube integrity continues to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* is controlled by plant procedures and work processes. The procedures and work processes provide steps for performance of the activities and require the documentation of the results.

**Operating Experience –** Operating experience associated with the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* has demonstrated that the eddy current testing provides adequate information on the extent of wall loss present in the heat exchanger tubes to predict when corrective action is required. Corrective action in the form of tube plugging, for example, is performed before the loss of the component intended function.

Due to operating experience at Catawba, the frequency of eddy current tests has been increased at both sites. During 1992-93, Catawba Unit 2 diesel generator engine cooling water heat exchangers experienced circumferential cracking of the tubes. Complete tube severance occurred on several tubes. Investigation revealed that the Unit 2 heat exchangers were set up on a weekly Nuclear Service Water system flush schedule (whereas Unit 1 heat exchangers were not). Circumferential cracks were determined to be linked to the thermal shock received during the Nuclear Service Water flushes. Flushes were discontinued and a special eddy current test probe was employed to determine the extent of circumferential cracking defects. Repairs were in the form of plugging and re-tubing.

***<u>Conclusion</u>***

The *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water* have been demonstrated to be capable of managing loss of material in heat exchanger tubes. The implementation of the preventive maintenance activities will adequately manage loss of material and will provide reasonable assurance that the aging effect will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.4 Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water

*Note: The* HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – CONTROL AREA CHILLED WATER *is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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 with managing loss of material or fouling for admiralty brass, carbon steel, and stainless steel materials.

**Scope –** The scope of the *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* is the McGuire and Catawba control room chiller condenser tubes and channel heads.

**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 *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* inspects the chiller tubes and channel heads to provide an indication of loss of material.

**Detection of Aging Effects** – In accordance with the information provided under **Monitoring & Trending**, the *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* will detect loss of material due to crevice, galvanic, general, pitting, and microbiologically influenced corrosion and particle erosion prior to loss of the component pressure boundary function. The program will also manage fouling prior to loss of heat transfer function.

**Monitoring & Trending** – The *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* performs eddy current testing on the heat exchanger tubes to measure wall thickness in order to detect areas with loss of material. Non–destructive testing (NDT) is performed on approximately 50% of the control room chiller condensers at least every five years. Analysis is required following each NDT to determine the need for further testing, replacement, or repair.

Fouling of the internal portions of the chiller tubes exposed to raw water is managed by routine cleaning. At least annually, the tubes are rodded out and cleaned. No actions are taken as part of this activity to trend inspection results.

Loss of material of the channel heads is managed by an annual visual inspection of the protective coatings to assure the integrity of the underlying base metal. The channel heads of the control room area chillers are coated with a high solids epoxy. The coating inspection specifically identifies rust blooms, which indicate a coating defect and corrosion of the base metal. No actions are taken as part of this activity to trend inspection results.

**Acceptance Criteria –** The acceptance criterion for the *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* is no unacceptable loss of material of the tubes and channel heads that could result in a loss of the component intended function(s) as determined by engineering evaluation.

**Corrective Action & Confirmation Process –** Engineering evaluation is performed to determine whether the tube integrity and coating and base metal continues to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* is controlled by plant procedures and work processes. The procedures and work processes provide steps for the performance of the activity and require the documentation of the results.

**Operating Experience –** Operating experience associated with the *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* has demonstrated that the eddy current testing provides adequate information on the extent of wall loss present in the chiller tubes to predict when corrective action is required. Corrective action in the form of tube plugging, for example, is performed before the loss of component intended function.

Periodic tube cleaning has proven to be an effective method of managing fouling of the tubes that could lead to loss of heat transfer. The control area chiller operates during normal plant operation. Routine surveillance of the chiller's operating parameters indicates that the periodic cleaning is effective in managing fouling of the chiller tubes.

Experience prior to application of the coatings of the carbon steel channel heads indicated that loss of material was occurring. Due to the inspection results, the channel heads were coated recently. Future inspection of the coatings ensures that the protective features of the coatings are maintained intact.

***Conclusion***

The *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* has been demonstrated to be capable of managing fouling of chiller tubes and loss of material of chiller tubes and channel heads. The implementation of the preventive maintenance activities will adequately manage loss of material and will provide reasonable assurance that the aging effects will be managed and that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.5 Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air

*Note: The Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air is applicable only to Catawba Nuclear Station.*

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 raw 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 that can affect the pressure boundary function. The program is credited with managing loss of material for carbon steel and stainless steel materials.

**Scope –** The scope of the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* is the tubes and channel heads of the diesel generator engine starting air aftercooler.

**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 *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* inspects the aftercooler tube and channel head surfaces for loss of material.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending**, the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* will detect loss of material due to crevice, galvanic, general, pitting, and microbiologically influenced corrosion and loss of material due to particle erosion prior to loss of the component intended function.

**Monitoring & Trending –** The *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* manages loss of material of the tubes and channel heads by means of two visual inspections. Loss of material of the tube internal surfaces is managed by an annual inspection. The inspection uses a borescope to visually inspect the tubes.

Loss of material of the channel heads is managed by an annual visual inspection of the protective coatings to assure the integrity of the underlying base metal. The channel heads of the diesel generator engine starting air aftercoolers are coated with a high solids epoxy. The coating inspection specifically identifies rust blooms, which indicate a coating defect and corrosion of the base metal.

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

**Acceptance Criteria –** The acceptance criterion for the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* is no unacceptable loss of material of the tubes and channel heads that could result in a loss of the component intended function(s) as determined by engineering evaluation.

**Corrective Action & Confirmation Process –** Engineering evaluation is performed to determine whether the tube integrity and coating and base metal continue to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* is controlled by work processes. The work processes provide steps for performance of the activities and require the documentation of the results.

**Operating Experience –** Operating experience associated with the *Heat Exchanger Preventive Maintenance Activities– Diesel Generator Engine Starting Air* has demonstrated that visual inspection of the aftercooler tubes and channel heads provides adequate information on the extent of wall loss present in the aftercooler components to predict when corrective action is required. Corrective action in the form of tube plugging or coating repair, for example, is performed before the loss of the component intended function.

Results of the inspection led to the replacement of the aftercooler tubes and the coating of the tube sheets and channel heads. Original equipment Monel tubes in the diesel generator engine starting air aftercoolers were retubed with stainless steel in 1996-1997. Monel tubes had shown signs of serious pitting damage. Replacement stainless steel tubes are also showing signs of pitting as well, but to a lesser degree than the Monel, and are being evaluated for retubing.

***Conclusion***

The *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* has been demonstrated to be capable of managing loss of material for aftercooler tubes and channel heads. The implementation of the preventive maintenance activities will adequately manage loss of material and will provide reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.17.6 Heat Exchanger Preventive Maintenance Activities- Pump Motor Air Handling Units

*Note: Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units is applicable only to McGuire Nuclear Station.*

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

**Scope –** The scope of *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* is the tubes in the following McGuire heat exchangers of the Auxiliary Building Ventilation System:

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

**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 *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* will inspect the heat exchanger tubes to provide an indication of loss of material. Fouling of the internal portions of the heat exchanger tubes exposed to raw water is managed by tube cleaning.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending** below, *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* will detect loss of material prior to loss of the component intended pressure boundary function. The program will also manage fouling prior to loss of heat transfer function.

**Monitoring & Trending –** The *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* will perform either a destructive or non-destructive examination of one of the twelve total cooling coils within the scope of the program. The examination method will permit inspection of the inside surfaces of the tubes for loss of material.

The selection of the specific inspection locations will take into consideration the normal operating environments. The containment spray pump motor air handling units and residual heat removal pump motor air handling units are normally isolated. The fuel pool cooling pump motor air handling units are normally in service and should experience the most susceptible service environment for loss of material to occur. One of the fuel pool cooling pump motor air handling units cooling coils will therefore be examined as a representative of the total program scope.

Tube cleaning is performed to manage fouling of the heat exchanger tubes. No actions are taken as part of this activity to trend inspection or test results.

This new comprehensive program will be implemented following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

**Acceptance Criteria –** The acceptance criterion for the *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* tube examination is no unacceptable loss of material of the tubes that could result in a loss of component intended function as determined by engineering evaluation.

**Corrective Action & Confirmation Process –** Engineering evaluation of the tube examination results of the sample will be performed to determine whether the tube integrity of all of the cooling coils continues to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls** – The *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* will be controlled by plant procedures and work processes. The procedures and work processes will provide steps for performance of the activities and require documentation of the results.

**Operating Experience** – The *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* tube examination is a new activity for which there is no plant-specific operating experience. There have been no age-related tube failures in any of the cooling coils within the scope of this program, as confirmed through periodic leak detection. A few tube leaks have been detected and repaired, but were determined not to be age-related.

Periodic tube cleaning has been performed in the past. Routine differential pressure testing determines when cleaning is required. This method has been effective in managing fouling of the heat exchanger tubes and will continue to be performed during the period of extended operation.

***Conclusion***

The *Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units* has been demonstrated to be capable of managing loss of material and fouling in heat exchanger tubes. The implementation of this comprehensive program will adequately manage loss of material and fouling in heat exchanger tubes, and will provide reasonable assurance that these components will continue to perform their intended function(s) during the period of extended operation.

### B.3.17.7 Heat Exchanger Preventive Maintenance Activities- Pump Oil Coolers

*Note: The HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – PUMP OIL COOLERS is applicable only to McGuire Nuclear Station.*

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

**Scope –** The scope of *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* is the tubes in the following McGuire heat exchangers of the Nuclear Service Water System:

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

**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 *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* inspects the heat exchanger tubes to provide an indication of loss of material. Fouling of the internal portions of the heat exchanger tubes exposed to raw water is managed by routine cleaning.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending** below, *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* will detect loss of material prior to a loss of the component intended pressure boundary function. The program will also manage fouling prior to loss of heat transfer function.

**Monitoring & Trending –** The *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* will perform eddy current testing on the heat exchanger tubes to measure wall thickness in order to detect areas with loss of material. Non-destructive testing (NDT) will be performed on 100% of the tubes. Following initial inspections, an appropriate frequency will be established based on inspection results.

Tube cleaning is performed to manage fouling of the heat exchanger tubes. No actions are taken as part of this activity to trend inspection or test results.

This new comprehensive program will be implemented following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

**Acceptance Criteria –** The acceptance criterion for the *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* eddy current testing activity is no unacceptable loss of material of the tubes that could result in a loss of the component intended function(s) as determined by engineering evaluation.

**Corrective Action & Confirmation Process –** Engineering evaluation will be performed to determine whether tube integrity continues to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls** – *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* will be controlled by plant procedures and work processes. The procedures and work processes will provide steps for performance of the activities and require documentation of the results.

**Operating Experience** – The *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* eddy current testing is a new activity for which there is no plant-specific operating experience. Eddy current examinations are volumetric methods accepted by the industry to be effective for detecting age-related degradation in heat exchanger tubes. There have been no tube failures in any of the heat exchangers within the scope of this program, as confirmed through periodic leak detection.

Periodic tube cleaning has been performed in the past. Cleaning every two to three years has been effective in managing fouling of the heat exchanger tubes. This periodic tube cleaning will continue to be performed during the period of extended operation.

***Conclusion***

The *Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers* has been demonstrated to be capable of managing loss of material and fouling in heat exchanger tubes. The implementation of this comprehensive program will adequately manage loss of material and fouling in heat exchanger tubes, and will provide reasonable assurance that the tubes will continue to perform their intended function(s) during the period of extended operation.

## B.3.18 ICE CONDENSER INSPECTIONS

*Note: The ICE CONDENSER INSPECTIONS are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The McGuire and Catawba ice condenser systems are an Engineered Safety Feature. The purpose of the ice condenser is to absorb the thermal energy released abruptly during a loss-of-coolant-accident or a secondary line break, thereby limiting the initial peak pressure and temperature in the Containment. Many activities are associated with maintaining the ice condenser system. The activities credited for license renewal are:

- Ice Basket Inspection
- Ice Condenser Engineering Inspection

### B.3.18.1 Ice Basket Inspection

Loss of material of the ice condenser steel ice baskets has been identified as an aging effect requiring management for the period of extended operation. The functional integrity of the ice condenser ice baskets ensures the ice condenser will perform its intended safety function. The *Ice Basket Inspection* is credited for managing aging effects for the period of extended operation. The *Ice Basket Inspection* is a condition monitoring program.

**Scope –** The scope of the *Ice Basket Inspection* includes all of the ice baskets located in the ice condenser.

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

**Parameters Monitored or Inspected –** The parameter monitored during the *Ice Basket Inspection* is loss of material.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Ice Basket Inspection* will detect loss of material of the ice baskets prior to loss of structure or component intended function.

**Monitoring & Trending –** The *Ice Basket Inspection* requires a visual inspection performed at a 40 months frequency as required by Technical Specification Surveillance Requirement (SR) 3.6.12.6. The sample for the Technical Specification surveillance includes two ice baskets from each of three azimuthal groups of bays. The azimuthal groups of bays are defined in Technical Specification 3.6.12.5.

The *Ice Basket Inspection* also requires a visual inspection every refueling outage. During refueling outages, each basket that is replenished, emptied and refilled with ice is visually inspected. The baskets are selected based on their ice weight and sublimation history.

Results of the *Ice Basket Inspection* are retained to permit adequate confirmation of the inspection programs. In particular, these records identify the inspectors, the results of the inspection and whether or not the results were acceptable, discrepancies and their cause, and any corrective action resulting from the inspection.

**Acceptance Criteria –** The acceptance criterion for the *Ice Basket Inspection* is no unacceptable visual indication of loss of material of the ice baskets that would prevent the ice condenser from performing its intended function.

**Corrective Action & Confirmation Process –** Ice condenser ice baskets which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repaired as required. Damaged baskets and cruciforms are repaired as needed using site procedures. Structures and components which are deemed unacceptable by the accountable engineer are documented under the corrective action program. Specific corrective actions and confirmatory actions, as needed, are implemented as part of site procedures and in accordance with the corrective action program.

**Administrative Controls –** The *Ice Basket Inspection* is governed by Technical Specification Surveillance Requirement (SR) 3.6.12.6 and implemented by plant procedures as required by Technical Specification 5.4.

**Operating Experience –**

***McGuire Operating Experience***
A review of the *Ice Basket Inspection* conducted at McGuire confirms the reasonableness and acceptability of the inspection frequency in that degradation of the ice basket is detected prior to loss of function.

Identified deficiencies were associated primarily with missing screws and minor dents on the ice baskets. These deficiencies were attributed to ice basket maintenance (i.e., weighing, replenishing ice, etc.), and are not age-related. Repairs were performed at the time of inspection under the guidance of site procedures.

***Catawba Operating Experience***
Previous *Ice Basket Inspections* have identified missing screws and other minor degradation such as dents and torn ligaments. These deficiencies were attributed to ice basket maintenance (i.e., weighing, replenishing ice, etc.), and are not age related.

***Conclusion***

The *Ice Basket Inspection*, governed by Technical Specification Surveillance Requirement (SR) 3.6.12.6, has been demonstrated to be capable of detecting and managing loss of material. Based on the above review, the continued implementation of the *Ice Basket Inspection* provides reasonable assurance that loss of material will be managed such that the intended functions of the ice baskets will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.18.2 Ice Condenser Engineering Inspection

Loss of material due to corrosion of steel components in the ice condenser environment has been identified as an aging effect requiring management for the period of extended operation. The *Ice Condenser Engineering Inspection* is credited with managing loss of material of the ice condenser upper plenum, lower plenum, and top deck blankets for the extended period of operation. The *Ice Condenser Engineering Inspection* is a condition monitoring program.

**Scope –** The scope of the *Ice Condenser Engineering Inspection* includes the ice condenser structural components in the upper plenum, lower plenum, and top deck blankets.

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

**Parameters Monitored or Inspected –** The parameter monitored with the *Ice Condenser Engineering Inspection* is loss of material.

**Detection of Aging Effects –** In accordance with information provided in Monitoring & Trending, the *Ice Condenser Engineering Inspection* will detect loss of material prior to loss of structure or component intended functions.

**Monitoring & Trending –** The *Ice Condenser Engineering Inspection* requires visual inspections of the structural components in the upper plenum, lower plenum, and top deck blankets. The inspection is performed every outage.

Records of ice condenser system engineer walkdown inspections are retained to permit adequate confirmation of the inspection programs. In particular, these records identify the inspector(s), the results of the inspection and whether or not the results were acceptable, deficiencies and their cause, and any corrective action resulting from the inspection.

In addition, the ice condenser system engineer generates periodic component health reports using the walkdown and monitoring data that has been assembled during that time period. Current trending information is retained in files.

**Acceptance Criteria –** The acceptance criteria are no adverse conditions that could prevent the ice condenser from performing its intended function. Acceptance criteria include no unacceptable visual indication of material condition including corrosion, glycol leaks, and missing or loose fasteners.

**Corrective Actions & Confirmation Process –** Ice condenser structural components which do not meet the acceptance criteria are evaluated by the ice condenser system engineer for continued service and repaired as required. Structures and components which are deemed unacceptable by the ice condenser system engineer are documented under the corrective action program. Specific corrective actions and confirmatory actions, as needed, are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Ice Condenser Engineering Inspection* is implemented as part of an engineering support program.

**Operating Experience –**

***McGuire Operating Experience***
A review of previous *Ice Condenser Engineering Inspections* conducted at McGuire confirms the reasonableness and acceptability of the inspection frequency in that degradation of ice condenser structural components is detected prior to loss of function. Identified deficiencies were primarily associated with installation problems and frost buildup, minor gouges on door skin, missing cover screws, slight glycol leak residual on walls and beams, minor rust on blanket fasteners, and flaking paint at ice condenser exterior end walls. The majority of work orders were generated for cosmetic repairs and removal of excess frost. The identified deficiencies were not age related with the exception of minor rusting on blanket fasteners. The minor rust did not result in any loss of intended function. This operating experience concurred with the statement in the McGuire UFSAR that the low temperature and humidity of the ice condenser environment is a non-corrosive environment [Reference B - 38, Section 6.2.2.18.2].

***Catawba Operating Experience***
A review of previous *Ice Condenser Engineering Inspections* conducted at Catawba confirms the reasonableness and acceptability of the inspection frequency in that degradation of ice condenser structural components is detected prior to loss of function.

Identified deficiencies were primarily associated with frost buildup, minor dents and gouges in door skins, missing fasteners, loose or torn tape on top deck blankets, tears and small punctures in top deck blankets, and glycol leaks. The deficiencies were not age-related; they are attributed to maintenance activities. This operating experience concurred with the statement in the Catawba UFSAR that the ice condenser materials of construction are not impaired by long term exposure to the ice condenser environment [Reference B - 39 , Section 6.7.18.2].

***Conclusion***

The *Ice Condenser Engineering Inspection* has been demonstrated to be capable of detecting and managing loss of material. Based on the above review, the continued implementation of the *Ice Condenser Engineering Inspection* provides reasonable assurance that loss of material will be managed such that the intended functions of the ice condenser structural components will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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## B.3.19 INACCESSIBLE NON-EQ MEDIUM-VOLTAGE CABLES AGING MANAGEMENT PROGRAM

*Note: The INACCESSIBLE NON-EQ MEDIUM-VOLTAGE CABLES AGING MANAGEMENT PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.*

The purpose of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is to demonstrate that the aging effects of inaccessible non-EQ medium-voltage cables caused by moisture and voltage stress will be adequately managed so that there is reasonable assurance that inaccessible non-EQ medium-voltage cables will perform their intended function in accordance with the current licensing basis during the period of extended operation. The intended function of a medium-voltage cable is to provide electrical connections to specified sections of an electrical circuit to deliver voltage or current. The *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is a condition monitoring program.

**Scope –** The scope of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* includes inaccessible (for example, in conduit or direct buried) non-EQ (not subject to 10 CFR 50.49 Environmental Qualification requirements) medium-voltage cables that are exposed to significant moisture simultaneously with significant voltage. Significant moisture is defined as exposure to long-term (over a long period such as a few years), continuous (going on or extending without interruption or break) standing water. Periodic exposures to moisture for shorter periods are not significant (for example, rain and drain exposure that is normal to yard cable trenches). Significant voltage is defined as exposure to system voltage for more than twenty-five percent of the time. The moisture and voltage exposures described as significant in these definitions are not significant for medium-voltage cables that are designed for these conditions (for example, continuous wetting and continuous energization is not significant for submarine cables).

**Preventive Actions –** No preventive actions are required as part of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program.* Periodic actions may be taken to prevent inaccessible non-EQ medium-voltage cables from being exposed to significant moisture such as inspecting for water collection in cable manholes and conduit and draining water as needed. Testing of a cable per this program is not required when such preventive actions are taken since the significant moisture criteria defined under **Scope** would not be met.

**Parameters Monitored or Inspected –** The specific cable insulation material parameters tested as part of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* are defined by the specific type of test performed and the specific cable tested.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* will detect aging effects for inaccessible non-EQ medium-voltage cables caused by moisture and voltage stress prior to loss of intended function.

**Monitoring & Trending –** Inaccessible non-EQ medium-voltage cables exposed to significant moisture and significant voltage are tested per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* to provide an indication of the condition of the conductor insulation and the ability of the cable to perform its intended function. The specific type of test performed will be determined before each test. Each test performed for a cable may be a different type of test. Inaccessible non-EQ medium-voltage cables exposed to significant moisture and significant voltage are tested at least once every 10 years.

Trending actions are not required as part of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* since the ability to trend test results is dependent on the specific type of test chosen. In addition, baseline data (cable insulation material parameters when the cable was new) is not normally available and methods for accurately predicting remaining life are not developed.

For McGuire, the first test per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, the first test per 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 performed per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* are defined by the specific type of test performed and the specific cable tested.

**Corrective Action & 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 inaccessible non-EQ medium-voltage cables. 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* will be controlled by plant procedures.

**Operating Experience –** The *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is a new program for which there is no operating experience. However, an equivalent program was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

***Conclusion***

The *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is equivalent to the program described and evaluated in Section 3.9.3.2 of NUREG 1723 [Reference B - 5]. The above review demonstrates that the aging effects of inaccessible non-EQ medium-voltage cables caused by simultaneous moisture and voltage stress will be adequately managed by the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* so that there is reasonable assurance that inaccessible non-EQ medium-voltage cables will perform their intended function in accordance with the current licensing basis during the period of extended operation.

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### B.3.20 INSERVICE INSPECTION PLAN

*Note: The INSERVICE INSPECTION PLAN is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Throughout the service life of nuclear power plants, Class 1 components and associated Class1 supports must meet the requirements set forth in Section XI of the ASME Code and Addenda that are incorporated by reference in 10 CFR 50.55a(b). These requirements are subject to the limitation listed in 10 CFR 50.55a, to the extent practical within the limitations of design, geometry and materials of construction of the component or support.

Inservice examinations and system pressure tests conducted during successive 120-month inspection intervals, following the initial 120-month inservice inspection interval, must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed in paragraph 10 CFR 50.55a(b).

The period of extended operation will contain the fifth and sixth inservice inspection intervals. The *Inservice Inspection Plan* for each interval of the renewal license period of extended operation for McGuire and Catawba will comply with 10 CFR 50.55a (g)(4)(ii) except that if an examination required by the Code or Addenda is determined to be impractical, then a relief request will be submitted to the Commission in accordance with the requirements contained in 10 CFR 50.55a (g)(5)(iii) and (iv), for Commission evaluation, as required by 10 CFR 50.55a (g)(6)(i).

The Integrated Plant Assessment performed for McGuire and Catawba credited the ASME Section XI Code requirements for inservice inspection of Class 1 components, Class 2 portions of the steam generators and associated supports as shown in Tables IWB 2500-1 and IWC-2500-1 of the 1989 Edition of ASME Section XI, including mandatory Appendices VII and VIII. Appendix VIII is in accordance with the 1995 Edition through 1996 Addenda. At present, the code of record for the McGuire and Catawba units is the 1989 Edition, no addenda as described in the second interval *Inservice Inspection Plan* for McGuire and Catawba.

*The Inservice Inspection Plan* includes the following inspections and activities:

- ASME Section XI, Subsections IWB and IWC (secondary side of steam generators) Inspections
- ASME Section XI, Subsection IWF Inspections
- McGuire Unit 1 Cold Leg Elbow
- Small Bore Piping

### B.3.20.1 ASME Section XI, Subsections IWB and IWC Inspections

ASME Section XI, Subsections IWB (Class 1) and IWC (Class 2) provide the rules and requirements for inservice inspection, repair, and replacement of pressure retaining components, their integral attachments, piping and secondary pressure boundary portions of the steam generators in light-water cooled plants. These inspections manage cracking of welded joints and bolting material as well as loss of material. *ASME Section XI, Subsections IWB and IWC Inspections* are condition monitoring programs.

**Scope –** All Class 1 pressure-retaining components and their integral attachments are included in the scope of the *ASME Section XI, Subsections IWB and IWC Inspections*. In addition, Subsection IWC, Examination Categories C-A, C-B, C-C, and C-H cover the Class 2 portions of the steam generators.

Items within the scope of this specification may be installed in areas inaccessible for maintenance and inspection. Though a conscious effort was made during design and construction to make items requiring inservice inspection accessible, competing design requirements meant it was not always achievable to do so. For those limited instances where an inaccessible item requiring inspection does exist, Code relief will be sought from the NRC.

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

**Parameters Monitored or Inspected –** Class 1 component welds, integral attachments, piping welds, bolted closures and supports as well as the Class 2 pressure boundary portions of the steam generators (welds and welded attachments) are inspected for cracking and loss of material.

**Detection of Aging Effects –** In accordance with information provided in ***Monitoring & Trending*** the *ASME Section XI, Subsections IWB and IWC Inspections* will detect weld cracking prior to loss of component intended function.

**Monitoring & Trending –** As directed by the *ASME Section XI, Subsections IWB and IWC Inspections*, three different types of examination are required: volumetric, surface, and visual examinations. Volumetric examinations are the most extensive, using methodologies such as radiographic, ultrasonic, or eddy current examinations to locate sub-surface flaws. Surface examinations use methodologies such as magnetic particle or dye penetrant testing to locate surface flaws.

Three levels of visual examinations are employed: The VT-1 visual examination is conducted to assess the condition of the surface of the part being examined, looking for cracks and symptoms of wear, corrosion, erosion or physical damage. It can be done with either direct visual observation or with remote examination using various optical/video devices. The VT-2 examination is conducted specifically to locate evidence of leakage from pressure retaining components. While the system is under pressure for a leakage test, visual examinations are made looking for direct or indirect indications of leakage. The VT-3 examination is conducted to determine the general mechanical and structural condition of components and supports and to detect discontinuities and imperfections such as loss of integrity at bolted connections. The VT-3 examination concentrates on such items as missing parts, debris, corrosion, clearances, and physical displacements.

The extent and frequency of inspection are specified in Tables IWB-2500-1 and IWC-2500-1 and IWF-2500-1. The inspection intervals are not restricted by the Code to the current term of operation and are valid for any period of extended operation.

**Acceptance Criteria -** Flaws detected during examination are evaluated by comparing the examination results to the acceptance standards established in ASME Section XI, IWB-3500 and IWC-3500. Unacceptable indications require detailed analyses, repair, or replacement.

The ASME Section XI acceptance standards ensure that all Service Conditions (A-D) are protected by maintaining the safety margin of the component throughout the service life of the component. When evaluating an operating component for an indication that exceeds the allowable acceptance standards established in IWB-3500 and IWC-3500, ASME Section XI requires the use of the original safety margins for all operating conditions (i.e., normal, upset, emergency and faulted conditions). The safety margins vary for specific cases (e.g., component, geometry, etc.) but are always consistent or conservative with respect to the original design margins.

**Corrective Action & Confirmation Process –** Specific corrective actions and confirmation will be implemented in accordance with ASME Section XI. In accordance with Subsections IWB and IWC, components containing relevant conditions shall be evaluated, repaired, or replaced prior to returning to service.

**Administrative Controls -** The *ASME Section XI, Subsections IWB and IWC Inspections* for both McGuire and Catawba are implemented using controlled plant procedures. Records are maintained in accordance with IWA-6000. Records are prepared in accordance with the requirements provided in IWA-6220. Summary reports of the examinations are submitted to the NRC in accordance with IWA-6230.

**Operating Experience -** The results of the *ASME Section XI, Subsections IWB and IWC Inspections* for McGuire and Catawba, which includes all of IWB and the portions of IWC that cover the steam generators, are submitted to the NRC. McGuire and Catawba are currently in the second inspection interval and have more than 20 years at McGuire and 15 years at Catawba with the inservice inspection of Class 1 components as well as the Class 2 pressure boundary portions of the steam generators. The inspections which have been completed to date have found very few flaws which do not meet the acceptance criteria and which required further evaluation in accordance with ASME Section XI.

For bolting, in addition to the aging management programs listed, information from operating experience indicates that there are additional elements of bolting maintenance procedures that should be considered, such as personnel training, installation and maintenance procedures, plant-specific bolting degradation history, and corrective measures. The NRC captured the lessons from this experience in IE Bulletin 82-02 [Reference B - 40] and directed each licensee to assure that these lessons were being incorporated at their plant. In response to IE Bulletin 82-02, Duke provided the results of the in-house investigation and provided assurance that bolting maintenance practices did indeed consider these lessons learned. In summary, routine maintenance practices have included use of properly trained personnel and procedural guidance to construct bolted closures. The continuation of routine maintenance practices reviewed under IE Bulletin 82-02 will assure aging management of mechanical closure integrity for bolted closures in the Reactor Coolant System.

***Conclusion***

The *ASME Section XI, Subsections IWB and IWC Inspections* have been demonstrated to be capable of managing inservice inspection, repair, and replacement of Class 1 pressure retaining components, their integral attachments, piping and secondary pressure boundary portions of the steam generators. The *ASME Section XI, Subsections IWB and IWC Inspections* described above are equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.4.3.3. Based on the above review, the continued implementation of the *ASME Section XI, Subsections IWB and IWC Inspections* provides reasonable assurance that the aging effects will be managed and that the Class 1 pressure retaining components and Class 2 pressure boundary portions of the steam generators will continue to perform their intended function for the period of extended operation.

### B.3.20.1.1 McGUIRE UNIT 1 COLD LEG ELBOW

Reduction in fracture toughness due to thermal embrittlement can be an aging effect for certain types of cast austenitic stainless steel in locations where temperatures continuously exceed 482ºF. In a May 19, 2000 letter to NEI, Christopher I. Grimes, Chief License Renewal and Standardization Branch clarified that not all cast austenitic stainless steels are subject to thermal embrittlement [Reference B - 41]. The piping components and reactor coolant pumps fabricated from cast austenitic stainless steel were evaluated using the acceptance criteria set forth in the above letter. For those components requiring evaluation, only the McGuire 1, 27 ½-inch ID Loop B cold leg elbow exceeds the NRC-established threshold and is susceptible to thermal embrittlement which requires aging management for license renewal.

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

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

1. A VT-2 visual examination will be performed each outage of the exterior of the affected elbow during the system leakage test.

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

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

### B.3.20.1.2 SMALL BORE PIPING

Small bore piping is defined as piping less than 4-inch NPS. This piping does not receive volumetric inspection in accordance with ASME Section XI, 1989 Edition, Examination Category B-J or B-F. Cracking has been identified as an aging effect requiring programmatic management for Reactor Coolant System small bore piping for the period of extended operation. A risk-informed method to select Class 1 piping welds for inspection in lieu of the requirements specified in ASME Section XI, Table IWB-2500-1, Examination Category B-J and B-F has been completed by Duke for use at McGuire during the third and fourth inservice inspection intervals. Duke plans to complete a similar review for Catawba.

The risk-informed approach is based on WCAP 14572 Revision 1 - NP-A [Reference B - 42] and consists of the following two essential elements: (1) a degradation mechanism evaluation is performed to assess the failure potential of the piping under consideration, and (2) a consequence evaluation is performed to assess the impact on plant risk in the event of a piping failure. As is required by WCAP 14572 Revision 1 – NP-A, the McGuire and Catawba risk-informed submittals will provide equivalent or better risk coverage for the Risk-Informed-Inservice Inspection scope.

The results from these two evaluations are coupled to determine the risk-significance of piping segments within the Reactor Coolant System and are used to select Class 1 piping welds for inspection. Duke has included all Class 1 piping (i.e., large bore, small bore and socket welds) with an internal diameter of greater than 3/8-inch in the evaluation. Class 1 flow through piping with an ID less than or equal to 3/8-inch is within the charging system capacity for Catawba and McGuire.

The risk-informed process used to select piping elements for inspection is consistent with the methodology used to identify aging effects requiring aging management for license renewal. In addition, a risk-informed approach was recently approved by the NRC at ANO-1 [Reference B - 43] to manage cracking of small bore piping during the period of extended operation. Duke also plans to use an NRC approved Risk-Informed-Inservice Inspection method during the period of extended operations for both McGuire and Catawba.

### B.3.20.2 ASME Section XI, Subsection IWF Inspections

Loss of material due to corrosion has been identified as an aging effect requiring programmatic management for Class 1, 2, and 3 piping and component supports for the extended period of operation in Reactor Building, Auxiliary Building Structures, and Nuclear Service Water Structures. *ASME Section XI, Subsection IWF Inspections* are credited with managing the potential loss of material for the ASME Class 1, 2, and 3 piping supports and components supports. The *ASME Section XI, Subsection IWF Inspections* is a condition monitoring program.

**Scope –** The scope of *ASME Section XI, Subsection IWF Inspections* is specified in IWF - 1210 and include ASME Class 1, 2, and 3 piping supports and component supports.

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

**Parameters Monitored or Inspected –** Parameters monitored by *ASME Section XI, Subsection IWF Inspections* include loss of material for Class 1, 2, and 3 piping and component supports.

**Detection of Aging Effects –** In accordance with the information provided in **Monitoring & Trending** this program will detect loss of material for Class 1, 2, and 3 piping and component supports prior to loss of structure or component intended functions.

**Monitoring & Trending –** Required examinations are directed by the *Inservice Inspection Plan*. The extent and frequency of examinations are specified in IWF-2400. Aging effects are detected through visual examination (VT-3). The complete inspection scope is repeated every 10-year inspection interval. No actions are taken as part of this program to trend inspection or test results.

**Acceptance Criteria –** Acceptance criteria are based on visual indication of structural damage or degradation specified in IWF-3400. The criteria are based on VT-3 visual examinations. Unacceptable conditions are noted for correction or further evaluation.

**Corrective Action & Confirmation Process –** Specific corrective actions and confirmation will be implemented in accordance with ASME Section XI. In accordance with IWF-3122, supports containing unacceptable conditions are evaluated or tested, or corrected prior to returning to service. Corrective actions are delineated in IWF-3122.2. IWF-3122.3 provides an alternative for evaluation or testing, to substantiate integrity for intended purpose.

**Administrative Controls –** *ASME Section XI, Subsection IWF Inspections* are implemented as part of the plant *Inservice Inspection Plan*. The licensee is responsible for preparation of plans, schedules, and inservice inspection summary reports. IWA-6000 specifically covers the requirements for the preparation, submittal, and retention of records and reports.

**Operating Experience –** The results of the McGuire and Catawba *Inservice Inspection Plans*, which include the inspection of ASME Class 1, 2, and 3 piping supports and component supports, are submitted to the NRC. McGuire and Catawba are currently on the second inspection interval and have more than 15 years of experience at Catawba and more than 20 years at McGuire with the inservice inspection of ASME Class 1, 2, and 3 piping supports and component supports. Previous inspections have revealed only minor degradation. Observations included loose load bolt on pipe clamps, clearance problems, and loss of material due to corrosion and rust. Most degradation was associated with installation and was not associated with aging. Where corrosion was noted, the supports were cleaned and coated in accordance with IWF. The observed aging effects were minor and had no impact on the ability of the piping supports to maintain their intended functions.

***Conclusion***

The *ASME Section XI, Subsection IWF Inspections* have been demonstrated to be capable managing potential loss of material for Class 1, 2 and 3 component supports. The *ASME Section XI, Subsection IWF Inspections* described above are equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *ASME Section XI, Subsection IWF Inspections* provides reasonable assurance that the aging effects will be managed and that the piping and component supports will continue to perform their intended function for the period of extended operation.

## B.3.21 INSPECTION PROGRAM FOR CIVIL ENGINEERING STRUCTURES AND COMPONENTS

*Note: The INSPECTION PROGRAM FOR CIVIL ENGINEERING STRUCTURES AND COMPONENTS is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The *Inspection Program for Civil Engineering Structures and Components* is credited with managing the following aging effects for the period of extended operation:

- Loss of material due to corrosion for exposed surfaces of steel components: anchorage / embedments; cable tray and conduit supports; checkered plates; equipment component supports; expansion anchors; flood curbs, flood, pressure, and specialty doors; HVAC duct supports; instrument line supports; instrument racks and frames; lead shielding supports; metal roof (MNS only), metal siding; pipe supports; stair, platform, and grating supports; structural steel beams, columns, plates and trusses; sump screens; and the unit vent stack
- Cracking of masonry block walls
- Change in material properties due to leaching of concrete walls and roofs
- Loss of material and cracking for reinforced concrete beams, columns, and walls for the Nuclear Service Water Structures and Low Pressure Service Water Intake Structure (CNS only)
- Cracking and change in material properties of elastomeric flood seals (CNS only)
- Loss of material of composite roofing
- Loss of material of exposed external surfaces of mechanical components
- Loss of material of the steel components of the Yard Drainage System (CNS only)

The *Inspection Program for Civil Engineering Structures and Components* is applicable in meeting the regulatory requirements of 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. The *Inspection Program for Civil Engineering Structures and Components* is a condition monitoring program.

**Scope –** The scope of the *Inspection Program for Civil Engineering Structures and Components* includes the following structures and the exposed external surfaces of mechanical components located within them:

<u>McGuire Nuclear Station</u>
- Auxiliary Building Structures (including the Control Building, Diesel Generator Buildings, Fuel Buildings, Main Steam Doghouses)
- Reactor Buildings (including Unit 1 and 2 internal structures, and Station Vents)
- Standby Nuclear Service Water Intake/Discharge Structures

- Standby Shutdown Facility
- Condenser Cooling Water Intake Structure (fire pump rooms only)
- Turbine Building (including Service Building)
- Yard Structures (including Refueling Water Storage Tank and Reactor Make-up Water Storage Tank foundations, Refueling Water Storage Tank missile wall, and trenches)

<u>Catawba Nuclear Station</u>

- Auxiliary Building Structures (including the Control Complex, Diesel Generator Buildings, Doghouses, Fuel Buildings, Fuel Pools)
- Nuclear Service Water (NSW) and Standby Nuclear Service Water (SNSW) Structures (including NSW and SNSW pump structure, NSW intake structure, SNSW Discharge structures, SNSW intake structure, and SNSW pond outlet)
- Reactor Buildings (including Station Vent, internal Reactor Building structures, and Containment Recirculation Sump Screen Assembly)
- Standby Shutdown Facility
- Turbine Building (including Service Building)
- Yard Structures (including Low Pressure Service Water Intake Structure, Refueling Water Storage Tank foundation and missile shield, Yard Drainage System, and trenches)

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

**Parameters Monitored or Inspected –** The *Inspection Program for Civil Engineering Structures and Components* inspects the structures and the exposed external surfaces of mechanical components within them for the following:

| | |
|---|---|
| Concrete | spalling, cracking, delaminations, honeycombs, water in-leakage, chemical leaching, peeling paint, or discoloration |
| Masonry Walls | significant cracks in joints, unsealed penetrations, missing or broken blocks, or separation from supports |
| Structural Steel | corrosion, peeling paint, beam/column deflection, loose or missing anchors/fasteners, missing or degraded grout under base plates, twisted beams, and cracked welds |
| Equipment Foundations | settlement, cracked concrete |
| Equipment Supports | cracked concrete, loose connections, corroded steel |
| Cable Tray Supports | loose connections, corrosion, distortion, and excessive deflection |
| Roof Systems | structural integrity, deteriorated penetrations (i.e., drains, vents, etc.), signs of water infiltration, cracks, ponding and flashing degradation |
| Seismic Gaps | gaps are present |

| | |
|---|---|
| Siding | structural integrity and visible damage |
| Windows/Doors | missing panes, cracks, deteriorated glazing, broken or cracked frames, missing or damaged hardware, and seal integrity |
| Trenches | cracks, mis-alignment or damage of covers, may spot check trenches by removing covers and inspecting walls and bottoms for cracks |
| Earthen Structures/Dams | erosion, settlement, slope stability, seepage, drainage systems, integrity of rip-rap, and environmental conditions |
| Mechanical Components | loss of material for exposed external surfaces (program will be enhanced to add this) |
| Yard Drainage System | loss of material of steel components (program will be enhanced to add this for Catawba only) |

In addition, certain structures and structural components may be exposed to environments which make them more susceptible to degradation. Examples include, but are not limited to:

| | |
|---|---|
| Chemical attack | Sumps and chemical use areas |
| Freeze/thaw | Trench covers |
| Excessive heat | Pipe penetrations, degradation of caulking, sealants and waterstops |
| Abrasion | High traffic areas |
| Settlement | Expansion joints. |

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Inspection Program for Civil Engineering Structures and Components* will detect loss of material, cracking, and change of material properties prior to loss of structure or component intended functions.

**Monitoring & Trending –** Each structure or component is inspected from the interior and exterior where accessible. Some structures (or portions of structures) may be inaccessible because of radiological considerations, obstructions or other reasons. Plant specific characteristics, industry experience, and/or testing history of such structures under similar environmental conditions may be evaluated in lieu of actual inspection of the inaccessible areas. Whenever normally inaccessible areas are made accessible (i.e., by excavation or other means) an inspection is performed and the results are documented as part of the *Inspection Program for Civil Engineering Structures and Components*. Inspections are performed by a team of at least two people. Inspectors are qualified by appropriate training and experience and approved by responsible plant management.

The *Inspection Program for Civil Engineering Structures and Components* is nominally performed every five years with the exact schedule being established with consideration of refueling outages for each unit. The interval may be increased to a nominal ten-year frequency with appropriate justification based on the structure, environment, and related inspection results. The inspection will be completed in phases as necessary based on the accessibility of each structure, with the goal of completing the inspection and issuing the report within twelve months of starting the inspection. Structures are monitored in accordance with §50.65 (a)(2) provided there is no significant degradation of the structure. Structures which are determined to be unacceptable are monitored in accordance with the provisions contained in §50.65(a)(1) of the Maintenance Rule.

Trending is performed in accordance with §50.65, the Maintenance Rule. Guidance for trending per the Maintenance Rule is provided in EDM-210, *Engineering Responsibilities for the Maintenance Rule*, Section 210.10.

**Acceptance Criteria –** The acceptance criteria are no unacceptable visual indications of loss of material, cracking or change of material properties for concrete, and loss of material for steel, as identified by the accountable engineer. Acceptable structures or components are those which are capable of performing their intended function(s) until the next scheduled inspection and are considered to meet the requirements contained in §50.65(a)(2) of the Maintenance Rule. Unacceptable structures or components are those which are damaged or degraded such that they are not capable of performing their intended function, or if degradation is to the extent and were allowed to continue uncorrected until the next normally scheduled inspection, such that the structure or component may not meet is design basis.

**Corrective Action & Confirmation Process –** Structures and components not meeting the acceptance criteria are evaluated by the accountable engineer for continued service, monitoring, repair, or replacement as required. Structures and components determined to be unacceptable are required to meet the provisions contained in §50.65(a)(1) of the Maintenance Rule. Structures and components which are deemed unacceptable are documented under the corrective action program or corrected using the work management system. Specific corrective actions and confirmation actions, as needed, are implemented in accordance with the corrective action program. Subsequent inspections confirm that the corrective action was implemented and was effective.

**Administrative Controls –** The *Inspection Program for Civil Engineering Structures and Components* is implemented in accordance with a department directive.

**Operating Experience –**

***McGuire Operating Experience***
Previous inspections noted several minor degraded conditions; however, the conditions did not adversely affect the ability of the structures or components to perform their intended functions. All findings have been addressed by the corrective action program or by station work requests. Items that were noted that required additional investigation, repair or other corrective actions included: missing grout under base plates; degraded coatings on steel, concrete, and pipe supports; minor corrosion of steel; deterioration of expansion joints; and minor cracking and spalling of concrete.

Corrective actions included repair or replacement of the affected structure or structural component. The determination of specific corrective actions, including whether or not additional inspections are warranted, were made using the corrective action program.

***Catawba Operating Experience***
Results of previous inspections revealed no serious degradation or condition that would adversely affect the ability of the structures or components to perform their intended functions. Items that required additional investigation, repair or other corrective actions included: missing grout under base plates; degraded coatings on steel, concrete, and block walls; minor corrosion of steel; deformed metal trench covers; and hairline cracking and leaching of concrete.

Corrective actions included repair or replacement of the affected structure or structural component. The determination of specific corrective actions, including whether or not additional inspections are warranted, were made using the corrective action program.

***Conclusion***
The *Inspection Program for Civil Engineering Structures and Components* has been demonstrated to be capable of detecting and managing aging. The *Inspection Program for Civil Engineering Structures and Components* described above is equivalent to the Inspection Program for Civil Engineering Structures and Components described and evaluated in NUREG-1723, Section 3.2.6 [Reference B - 5]. Based on the above review, the continued implementation of the *Inspection Program for Civil Engineering Structures and Components* provides reasonable assurance that aging will be managed such that the intended functions of the structures and components will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.22 LIQUID WASTE SYSTEM INSPECTION

*Note: The LIQUID WASTE SYSTEM INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Liquid Waste System Inspection* is to characterize any loss of material and cracking of system components within the scope of license renewal exposed to unmonitored borated and treated water environments and raw water environments. An unmonitored borated or treated water environment is one that may contain conditions that can concentrate existing levels of contaminants and are not routinely monitored by Chemistry Control Program. Uncertainty exists as to whether exposure to these environments could lead to loss of material and cracking such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect system components in the various environments to detect the presence and extent of any loss of material and cracking. The *Liquid Waste System Inspection* is a one-time inspection.

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

- Component Cooling System (MNS only)- the portion of the Component Cooling System of concern is the stainless steel waste evaporator package exposed to an unmonitored treated water environment of the Liquid Waste Recycle System;
- Liquid Waste Recycle System (MNS)- stainless steel components exposed to an unmonitored borated water environment;
- Liquid Radwaste System (CNS)- 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 each system listed above. As an alternative, visual examination will be used should access to internal surfaces become available. Selection of the specific areas for inspection for the system material/environment combinations will be the responsibility of the system engineer.

*Component Cooling System (MNS only)*
At McGuire, the waste evaporator package consists of four heat exchangers. One of the four heat exchangers will be inspected. The inspection results will be applied to the other three stainless steel heat exchanger components exposed to unmonitored treated water environments.

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

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

**Operating Experience –** The *Liquid Waste System Inspection* is a one-time inspection activity for which there is no operating experience.

***Conclusion***

Based on the above review, implementation of the *Liquid Waste System Inspection* will adequately verify that no need exists to manage the aging effects on the components or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

## B.3.23 NON-EQ INSULATED CABLES AND CONNECTIONS AGING MANAGEMENT PROGRAM

*Note: The NON-EQ INSULATED CABLES AND CONNECTIONS AGING MANAGEMENT PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.*

The purpose of the *Non-EQ Insulated Cables and Connections Aging Management Program* is to demonstrate that the aging effects of accessible non-EQ insulated cables and connections caused by heat or radiation will be adequately managed so that there is reasonable assurance that accessible non-EQ insulated cables and connections will perform their intended function in accordance with the current licensing basis during the period of extended operation. The intended function of insulated cables and connections is to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals. The *Non-EQ Insulated Cables and Connections Aging Management Program* is a condition monitoring program.

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

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

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

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

**Monitoring & Trending –** Accessible non-EQ insulated cables and connections installed in the Reactor Buildings, Auxiliary Building and Turbine Building are visually inspected per the *Non-EQ Insulated Cables and Connections Aging Management Program* at least once every 10 years. EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments* [Reference B - 44], is used as guidance in performing the inspections.

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

For McGuire, the first inspection per the *Non-EQ Insulated Cables and Connections Aging Management Program* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). 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 criteria for inspections performed per the *Non-EQ Insulated Cables and Connections Aging Management Program* is no unacceptable visual indications of cable and connection jacket surface anomalies that suggest conductor insulation degradation exists, as determined by engineering evaluation. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

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

**Operating Experience –** The *Non-EQ Insulated Cables and Connections Aging Management Program* is a new program for which there is no operating experience. However, an equivalent program was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

***Conclusion***

The *Non-EQ Insulated Cables and Connections Aging Management Program* is equivalent to the program described and evaluated in Section 3.9.3.2 of NUREG 1723 [Reference B - 5]. The above review demonstrates that the aging effects of accessible non-EQ insulated cables and connections caused by heat or radiation will be adequately managed by the *Non-EQ Insulated Cables and Connections Aging Management Program* so that there is reasonable assurance that accessible non-EQ insulated cables and connections will perform their intended function in accordance with the current licensing basis during the period of extended operation.

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## B.3.24 Preventive Maintenance Activities

### B.3.24.1 Condenser Circulating Water System Internal Coating Inspection

*Note: The Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection is applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

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.

The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* is a condition monitoring program.

**Scope –** The scope of the *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* is the internal surface of the intake and discharge piping of the Condenser Circulating Water System. This inspection is also applicable to carbon steel, cast iron, ductile iron, galvanized steel, and stainless steel exposed to the underground environments in the following McGuire and Catawba systems:

- Diesel Generator Fuel Oil System [Footnote B.3.24-1]
- Exterior Fire Protection
- Interior Fire Protection (CNS Only)
- Nuclear Service Water System
- Standby Shutdown Diesel System

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

**Parameters Monitored or Inspected –** The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* inspects the coating for chipping, peeling, blistering, and missing coatings as well as signs of corrosion of the underlying carbon steel pipe.

Inspection of the internal coatings of the large diameter Condenser Circulating Water System piping also provides symptomatic evidence that the external coating has degraded to the point that aging of the external surfaces is a concern. During plant construction, plant components were coated externally with coal tar epoxy prior to burial to prevent the underground environment from contacting the external surfaces to preclude loss of material and cracking. Direct inspection of the external coating is not practical due to the potential for damage during excavation and backfill and has a limited scope. Operating experience has revealed a limited number of leaks of buried plant components that Duke believes are attributable to damage during construction. Nonetheless, a method of assessing the condition of the external coating of buried components is needed.

Inspection of the internal surfaces of the buried intake and discharge piping of the Condenser Circulating Water System provides an indirect indication of the condition of the external coating. Increasing number and frequency of through-wall pits revealed by internal coating degradation and in-leakage discovered during inspections that, based on engineering evaluation, have originated from the external surface of the pipe will provide symptomatic evidence of external surface condition. Approximately 530,000 square feet at McGuire and 643,000 square feet at Catawba of internal surface are inspected. This comprises greater than 80 percent of the total buried surface area at McGuire and Catawba. Since all buried

B.3.24-1 Additional information regarding the condition of the external surfaces of buried components in the Diesel Generator Fuel Oil System is obtained by the required surveillances of Technical Specification 3.83 at McGuire and Catawba as well as Selected Licensee Commitment 16.8-5 at Catawba.

components have the same exterior coating exposed to the same underground environment, the inspection results of the Condenser Circulating Water System can be applied to the smaller buried components of other systems where internal inspection is not feasible.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* will detect loss of material and cracking from exposure to soil and groundwater prior to loss of the component intended function(s).

**Monitoring & Trending –** The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* visually inspects the internal coatings of the intake and discharge piping every five years for coating degradation. The internal coating is inspected for chipping, blistering, peeling, and missing coatings as well as signs of corrosion of the underlying pipe that are the result of internally generated degradation.

Externally generated through-wall pits will be revealed through the observance of blistering, peeling, or missing internal coatings as well as signs of corrosion of the underlying pipe and inleakage of soil or groundwater.

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

**Acceptance Criteria –** The acceptance criteria for the *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* are no visual indications of coating defects including but not limited to blistering, peeling, or missing coatings that reveal corrosion of the piping as determined by Engineering.

**Corrective Action & Confirmation Process –** Engineering evaluation is performed to determine whether the coating and base metal continue to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –***Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* is controlled by plant procedures and work processes. The procedures and work processes provide steps for performance of the activities and require documentation of the results.

**Operating Experience –**

***McGuire Nuclear Station***
At McGuire, one complete inspection has been performed on Units 1 and 2 intake and discharge piping and on the Low-Level Intake piping from Cowans Ford Dam through the Low-Level Intake Structure to the Main Intake within the last five years. The internal coating

was observed to be in good condition with random minor defects and corrosion. The Condenser Circulating Water System intake and discharge piping has experienced two leaks. One leak was a crack that developed in a weld near the low-level intake pumps as a result of one or two water hammer events. A pinhole was discovered during a visual inspection of the low-level intake piping. The diameter of the pinhole was larger on the outside diameter than the inside diameter, indicating that the corrosion initiated on the external surface of the pipe. No inspection was performed on the external surface of the pipe. The pinhole was repaired with a steel pipe plug.

***Catawba Nuclear Station***

At Catawba, the Condenser Circulating Water System is scheduled to be entered every outage for blasting and recoating and/or a walkdown of areas that are not recoated. This work is being performed because the original interior coating was not properly applied and is failing. These recoating and walkdown inspections have not identified any through-wall pits originating from the exterior of the pipe. Upon completion of the recoating work, Catawba will go to a five-year inspection frequency.

***External Surface***

During the Catawba Unit 1 outage in the fall of 2000, piping in the Nuclear Service Water System was cleaned to remove the fouling buildup from the pipe walls. Internal inspection of accessible areas after the cleaning discovered a row of small through-wall pits. Excavation of the pipe and examination of the external coating revealed that the coating had been cut during construction allowing the underground environment to contact the external surface. Except for the cut, plant personnel noted the external coating was in good shape. Plant personnel have also identified other instances of externally generated through-wall leaks of buried components by other means that have been attributed to construction-related damage.

## ***Conclusion***

The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* has been demonstrated to be capable of managing loss of material of the internally coated carbon steel intake and discharge piping of the Condenser Circulating Water System by maintaining an intact protective coating. The above description has also demonstrated that this inspection is capable of managing loss of material and cracking of externally coated buried components by providing symptomatic evidence of the condition of the external coating and its ability to protect the external surfaces. The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.2.10 [Reference B - 5]. Based upon the above review, the continued implementation of the *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* provides reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.24.2 Refueling Water Storage Tank Internal Coating Inspection

*Note: The PREVENTIVE MAINTENANCE ACTIVITIES – REFUELING WATER STORAGE TANK INTERNAL COATING INSPECTION is applicable only to McGuire Nuclear Station.*

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

**Scope –** The scope of the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* is the internal surface of the McGuire Units 1 and 2 carbon steel refueling water storage tanks in the Refueling Water System.

The comparable refueling water storage tanks at Catawba are constructed of stainless steel. The *Borated Water Systems Stainless Steel Inspection* (B.3.4) and the *Chemistry Control Program* (B.3.6) are credited with managing the aging effects of the stainless refueling water storage tanks at Catawba.

**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 *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* inspects the phenolic epoxy paint for signs of blistering, chipping, peeling, and missing paint as well as signs of corrosion of the underlying carbon steel tank.

**Detection of Aging Effects –** In accordance with the information provided under **Monitoring & Trending** below, the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* will detect loss of material prior to loss of the component intended function.

**Monitoring & Trending –** The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* visually inspects the internal phenolic epoxy paint every ten years using an underwater video camera. The inspection looks for signs of blistering, chipping, peeling, and missing paint as well as signs of corrosion of the underlying carbon steel tank. Detection of defects in the internal coating results in draining of the tank for further inspection and evaluation of the defects.

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

**Acceptance Criteria –** The acceptance criteria for the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* are no visual indications of coating defects that have led to corrosion of the underlying carbon steel tank surfaces.

**Corrective Action & Confirmation Process –** Engineering evaluation is performed to determine whether the coating and base metal continue to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* is controlled by plant procedures and work processes. The procedures and work processes provide steps for performance of the activities and require documentation of the results.

**Operating Experience –** Recently, the internal surfaces of the refueling water storage tanks for McGuire Units 1 and 2 were inspected during outage 1EOC-13 and 2EOC-12, respectively, using an underwater camera. Video results showed some second coating blistering, so the tanks were drained, visually inspected, and repainted in the necessary locations. No bare metal was exposed as a result of the blistering. A layer of the coating remained in the blistered location. The submerged portion of the tanks showed little to no degradation. However, the roof, which is not a part of the pressure boundary of the tank, did show evidence of coating concerns and was blasted and repainted in several locations. This operating experience demonstrates that this activity when continued through the extended period of operation will continue to be effective in managing loss of material of the carbon steel tank by maintaining the effectiveness of the phenolic epoxy paint.

***Conclusion***

The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* has been demonstrated to be capable of managing loss of material of the internal carbon steel by maintaining an intact protective coating. The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* described above is similar to the corresponding program described and evaluated in NUREG-1723, Section 3.2.10 [Reference B - 5]. Based upon the above review, the continued implementation of the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* provides reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.25 Reactor Coolant System Operational Leakage Monitoring Program

*Note: The Reactor Coolant System Operational Leakage Monitoring Program is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Reactor Coolant System Operational Leakage Monitoring Program* is to provide an additional line of defense against aging effects that may result in leakage due to cracking and loss of mechanical closure integrity. McGuire and Catawba have a continual Reactor Coolant System Technical Specification leakage limit and system surveillance requirement as defined in their Technical Specifications. The *Reactor Coolant Operational Leakage Monitoring Program* is a condition monitoring program that provides reasonable assurance that leakage will be detected prior to loss of reactor coolant system function.

**Scope –** The scope of the *Reactor Coolant System Operational Leakage Monitoring Program* is all Reactor Coolant components that contain coolant; however it is specifically credited with managing aging of bolted closures on the steam generators, pressurizer, and reactor coolant pumps as well as the Inconel penetrations on the reactor vessel head and steam generator tubes.

**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 *Reactor Coolant System Operational Leakage Monitoring Program* monitors Reactor Coolant System operational leakage and steam generator primary to secondary leakage.

**Detection of Aging Effects –** In accordance with the information provided in **Monitoring & Trending**, the *Reactor Coolant System Operational Leakage Monitoring Program* will detect cracking of the Reactor Coolant System pressure boundary and loss of mechanical closure integrity of bolted closures in cases where leakage is occurring.

**Monitoring & Trending –** The method for monitoring reactor coolant system operational leakage is specified in McGuire and Catawba Technical Specifications 3.4.13, *RCS Operational LEAKAGE*, and 3.4.15, *RCS Leakage Detection Instrumentation*.

GDC 30 of Appendix A to 10CFR50 requires means for detecting and, to the extent practical, identifying the location of the source of Reactor Coolant System leakage. Regulatory Guide 1.45 [Reference B - 45] describes acceptable methods for selecting leakage detection systems.

The primary method of detecting leakage into the Containment is measurement of the Containment floor and equipment level sump level. The sump level rate of change is calculated by the plant computer and can detect a 1 gpm leak within an hour. Leakage from the Reactor Coolant, Main Steam and Feedwater Systems can be detected this way. The containment ventilation unit condensate drain tank level change is another method of detecting leakage that is capable of detecting a 1 gpm leak. Radioactivity monitoring of particulate and gaseous radiation levels are also indicative of Reactor Coolant System leakage because the activity levels contained within the Reactor Coolant System during operation of the plant. Primary to secondary leakage from steam generator tubes can be detected by effluent monitoring (for activity) within the secondary steam and feedwater systems.

A reactor coolant water inventory balance is performed every 72 hours at steady state operation as specified in plant technical specifications to verify that leakage is within allowable limits.

Steam Generator primary to secondary leakage is monitored continuously using an operator aid computer point, radiation monitors, condensate steam air ejector off gas or secondary tritium samples depending on monitoring equipment availability and operating Mode.

**Acceptance Criteria –** The acceptance criteria are provided in the McGuire and Catawba Technical Specifications LCO 3.4.13, *RCS Operational LEAKAGE*.

**Corrective Action & Confirmation Process –** Corrective actions for this program are specified in the McGuire and Catawba Technical Specifications 3.4.13, *RCS Operation LEAKAGE*, and 3.4.15, "RCS Leakage Detection Instrumentation." Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *RCS Operational Leakage Monitoring Program* is implemented by written procedures as required by Technical Specification 5.4.1.

**Operating Experience –** A search of Licensee Event Reports (LER) was performed to demonstrate the effectiveness of *the Reactor Coolant System Operational Leakage Monitoring Program* for McGuire and Catawba Nuclear Stations. Many of the LERs were maintenance issues, however several identified age-related events. Some of the issues were leakage due to loose valve bonnet bolts, leakage from an incore thermocouple fitting, a leaking compression fitting and a weld failure due to fatigue resulting from cavitation. In all of the above cases, a determination was made that the events had no significance regarding the health and safety of the public.

Another use of this program, especially prior to steam generator replacement, is monitoring of primary to secondary leakage through the steam generators. Leakage that is still within allowable limits can be monitored and a determination regarding timing of shutdown and repair of steam generator tubes can be made.

***Conclusion***

The *Reactor Coolant System Operational Leakage Monitoring Program* has been demonstrated to be capable of providing an additional line of defense against aging effects that may result in leakage due to cracking and loss of mechanical closure integrity. The *Reactor Coolant System Operational Leakage Monitoring Program* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *Reactor Coolant System Operational Leakage Monitoring Program* provides reasonable assurance that the aging effects will be managed and that the reactor coolant pressure boundary will continue to perform its intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

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## B.3.26 REACTOR VESSEL INTEGRITY PROGRAM

*Note: The Reactor Vessel Integrity Program is generically applicable to both McGuire and Catawba Nuclear Stations, except as otherwise noted.*

The purpose of the *Reactor Vessel Integrity Program* is to manage the reduction of fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor vessel beltline is maintained for the period of extended operation. The program includes an evaluation of radiation damage based on pre-irradiation and post irradiation testing of Charpy V-notch and tensile specimens. The *Reactor Vessel Integrity Program* is a condition monitoring program.

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

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

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

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

**Monitoring & Trending –** Each reactor vessel had six specimen capsules located in guide baskets welded to the outside of the neutron shield pads and were positioned directly opposite the center portion of the core. McGuire Unit 1 and Catawba Unit 2 capsules contain reactor vessel steel specimens oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting shell plate located in the core region. McGuire Unit 2 and Catawba Unit 1 reactor vessel specimens are oriented both parallel and normal to the major working direction of the limiting core region shell forging. Associated weld metal and weld heat affected zone metal specimens are also included in each capsule. Capsule withdrawal schedules for the McGuire and Catawba Units are provided in Table B.3.26-1 and Table B.3.26-2, respectively. The limiting weld material is not contained in a McGuire Unit 1 surveillance capsule, but is contained in a sister plant surveillance capsule and integrated into the McGuire Unit 1 surveillance program.

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

***Fluence Received by the Specimens –*** Dosimeters such as Ni, Cu, Fe, Co-Al, shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are contained in the capsules. The dosimeters permit evaluation of the flux seen by the specimens. In addition, thermal monitors made of low melting point alloys are included to monitor the temperature of the specimens. A description of the methodology used to evaluate fluence received by the specimens using dosimetry measurements and fluence calculations, assuming the same neutron spectrum at the specimens and the vessel inner wall, is described in each station's UFSAR (McGuire UFSAR, Sections 5.4.3.7.1 and 5.4.3.7.2 [Reference B - 38] and Catawba UFSAR, Sections 5.3.1.6.1 and 5.3.1.6.2 [Reference B - 39]). The correlations have indicated good agreement and form the bases for ensuring that the calculations of the integrated flux at the vessel wall are conservative WCAP-14040 [Reference B - 46]. 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.

**Operating Experience –** The *Reactor Vessel Integrity Program* monitors neutron embrittlement to assure their acceptability in accordance with NRC Regulations 10 CFR 50.60 and 10 CFR 50.61. NRC Regulation 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," requires that all light water nuclear power reactors meet the requirements of Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of Part 50. Appendix G specifies fracture toughness requirements for the reactor coolant pressure boundary to provide margins of safety against fracture during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. Appendix H requires monitoring the changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment.

Fracture toughness requirements for protection against pressurized thermal shock events are provided in 10 CFR 50.61.

McGuire and Catawba comply with the requirements of 10 CFR 50.60, Appendices G and H and 10 CFR 50.61, through the *Reactor Vessel Integrity Program*.

***Conclusion***

The *Reactor Vessel Integrity Program* has been demonstrated to be capable of ensuring that reactor vessel degradation is identified and corrective actions are taken prior to exceeding allowable limits. The *Reactor Vessel Integrity Program* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *Reactor Vessel Integrity Program* provides reasonable assurance that the aging effects will be managed and that the reactor vessel will continue to perform its intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

## Table B.3.26-1

## McGuire Reactor Vessel Capsule Withdrawal Schedule

| Unit | Capsule | Withdrawal End of Cycle (EOC) | Projected EOC Date | Estimated Fluence ($n/cm^2 \times 10^{19}$) | Reference |
|---|---|---|---|---|---|
| Unit 1 | U | 1 | 2/24/84 | 0.405 | WCAP-10786 |
| Unit 1 | X | 5 | 10/12/88 | 1.50[a] | WCAP-12354 |
| Unit 1 | V | 8 | 3/12/93 | 2.08 [b][c] | WCAP-13949 |
| Unit 1 | Y | 11 | 2/14/97 | 2.86 [d] | WCAP-14993 |
| Unit 1 (dosimetry analysis & storage) | Z | 8 | 3/12/93 | 2.38 | WCAP-13949 |
| Unit 1 | W | 16 | 4/5/04 | 4.52 | STANDBY |
| Ex-vessel Cavity Dosimetry | N/A | 12 | 5/29/98 | 1.58 | WCAP-15253 |
| | | | | | |
| Unit 2 | V | 1 | 1/25/85 | 0.323 | WCAP-11029 |
| Unit 2 | X | 5 | 7/5/89 | 1.47 [a] | WCAP-12556 |
| Unit 2 | U | 7 | 1/9/92 | 2.04 [b][c] | WCAP-13516 |
| Unit 2 | W | 10 | 4/5/96 | 3.07 [d] | WCAP-14799 |
| Unit 2 (dosimetry analysis & storage) | Z | 8 | 7/1/93 | 2.41 | WCAP-14231 |
| Unit 2 (dosimetry analysis & storage) | Y | 8 | 7/1/93 | 2.08 [b] | WCAP-14231 |
| Ex-vessel Cavity Dosimetry | N/A | 12 | 3/12/99 | -- | WCAP-15334 |

a. Approximate fluence at vessel 1/4 thickness location, at 32 EFPY
b. Approximate fluence at vessel inner wall location, at 32 EFPY
c. Approximate fluence at vessel 1/4 thickness location, at 54 EFPY
d. Approximate fluence at vessel inner wall location at 54 EFPY

## Table B.3.26-2

## Catawba Reactor Vessel Capsule Withdrawal Schedule

| Unit | Capsule | End of Cycle (EOC) | Projected EOC Date | Estimated Fluence ($n/cm^2 \times 10^{19}$) | Reference |
|---|---|---|---|---|---|
| Unit 1 | Z | 1 | 8/8/86 | 0.299 | WCAP -1527 |
| Unit 1 | Y | 6 | 7/10/92 | 1.318 [a] | WCAP-13720 |
| Unit 1 | W | 14 | 11/29/03 | 3.0 [d] | -- |
| Unit 1 (dosimetry analysis & storage) | X | 10 | 12/20/97 | 2.439 | WCAP-15117 |
| Unit 1 (dosimetry analysis & storage) | U | 10 | 12/20/97 | 2.439 | WCAP-15117 |
| Unit 1 | V | 10 | 12/20/97 | 2.334 [b][c] | WCAP-15117 |
| Ex-vessel Cavity Dosimetry | N/A | 13 | 5/18/02 | -- | -- |
| | | | | | |
| Unit 2 | Z | 1 | 12/23/87 | 0.323 | WCAP-11941 |
| Unit 2 | X | 5 | 1/23/93 | 1.23 [a] | WCAP-13875 |
| Unit 2 | W | 14 | 3/9/06 | 3.0 [d] | -- |
| Unit 2 (dosimetry analysis & storage) | U | Standby | Standby | --- | -- |
| Unit 2 (dosimetry analysis & storage) | Y | 9 | 9/13/98 | 2.38 | WCAP-15243 |
| Unit 2 | V | 9 | 9/13/98 | 2.38 [b][c] | WCAP-15243 |
| Ex-vessel Cavity Dosimetry | N/A | 13 | 10/18/04 | -- | -- |

a. Approximate fluence at vessel 1/4 thickness location, at 32 EFPY
b. Approximate fluence at vessel inner wall location, at 32 EFPY
c. Approximate fluence at vessel 1/4 thickness location, at 54 EFPY
d. Approximate fluence at vessel inner wall location at 54 EFPY

## B.3.27 REACTOR VESSEL INTERNALS INSPECTION

*Note: The REACTOR VESSEL INTERNALS INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Reactor Vessel Internals Inspection* is to inspect the condition of reactor vessel internals items in order to assure that the applicable aging effects will not result in loss of the intended functions of the reactor vessel internals during the period of extended operation. The reactor vessel internals stainless steel items 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). Different aging effects will affect the various parts.

Similar reactor vessel internals inspections will be performed at other nuclear plants. Specifically, inspections are planned at Oconee Nuclear Station. In addition, the characterization of the internals aging effects through activities of EPRI and other industry groups focussed on reactor vessel internals will ensure a better understanding of the identified aging effects. These inspections and industry activities will provide significant insights prior to McGuire or Catawba entering their respective periods of extended operation.

McGuire Unit 1 will be Duke's lead Westinghouse plant for reactor vessel internals inspection since it is expected to have the most hours of operation among the four units. After the results of the McGuire Unit 1 and the Oconee inspections are evaluated, additional actions may be taken regarding McGuire Unit 2 and Catawba Units 1 and 2. The Oconee Reactor Vessel Internals Inspection is described in Oconee UFSAR Section 18.3.20 and has been evaluated in NUREG-1723, Section 3.4.3.3.

The *Reactor Vessel Internals Inspection* will supplement *The Inservice Inspection Plan* to assure that aging effects, potentially requiring additional management, will not result in loss of the intended functions of the reactor vessel internals during the period of extended operation.

**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 reduction of fracture toughness due irradiation embrittlement, and dimensional changes due to void swelling.

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.

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.

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. The decision to perform inspections on McGuire Unit 2, Catawba Unit 1 and Catawba Unit 2 and when to perform such inspections will depend on an evaluation of the results of the internals inspections performed at Oconee and on McGuire Unit 1.

With respect to dimensional changes due to void swelling, McGuire and 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 McGuire Unit 1, McGuire Unit 2, 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 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 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.

**Operating Experience –** The *Reactor Vessel Internals Inspection* is a new inspection for which there is no operating experience. However, a similar inspection was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

***<u>Conclusion</u>***

The *Reactor Vessel Internals Inspection* described above is similar to and builds upon the corresponding reactor vessel internals inspection described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the implementation of the *Reactor Vessel Internals Inspection* will assure the reactor vessel internals will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operation license).

### B.3.28 SELECTIVE LEACHING INSPECTION

*Note: The SELECTIVE LEACHING INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Selective Leaching Inspection* is to characterize any loss of material due to selective leaching of system components exposed to raw water environments. Uncertainty exists as to whether long term exposure to raw water environments could cause loss of material due to selective leaching in brass and cast iron components such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect brass and cast iron components exposed to raw water to detect the presence and extent of any loss of material due to selective leaching. The *Selective Leaching Inspection* is a one-time inspection.

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

- Conventional Wastewater Treatment (MNS Only)
- Diesel Generator Room Sump Pump (MNS Only)
- Exterior Fire Protection
- Groundwater Drainage (MNS Only)
- Interior Fire Protection
- Nuclear Service Water (MNS Only)

**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 each site. 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 each site 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 each site 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 each site, a select set of susceptible locations will be examined based on accessibility, operational, and radiological concerns. The results of this inspection will be applied to the brass components exposed to raw water environments in the systems listed above.

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

**Operating Experience –** The *Selective Leaching Inspection* is a new one-time inspection for which there is no operating experience. However, a similar inspection was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

***Conclusion***

The *Selective Leaching Inspection* described above is similar to the Selective Leaching Inspection described and evaluated in NUREG-1723, Section 3.5.3.2 [Reference B - 5]. Based on the above review, implementation of the *Selective Leaching Inspection* will adequately verify that no need exists to manage the aging effect on the components or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.29 SERVICE WATER PIPING CORROSION PROGRAM

*Note: The SERVICE WATER PIPING CORROSION PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Service Water Piping Corrosion Program* is to manage loss of material in order to maintain the pressure boundary function of specific raw water system components within the scope of license renewal. The program will manage the more uniform loss of material such as that due to general corrosion as well as particulate erosion in areas of higher flow velocity. Components subject to these generalized effects are made from carbon and galvanized steel, cast and ductile iron, and copper alloys in the McGuire and Catawba raw water systems. The program also will serve to manage loss of material due to localized corrosion of components made from carbon and galvanized steel, cast and ductile iron, copper alloys and stainless steel. The *Service Water Piping Corrosion Program* is a condition monitoring program.

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

- Containment Ventilation Cooling Water (MNS only)
- Exterior Fire Protection
- Interior Fire Protection
- Nuclear Service Water

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

- Containment Spray
- Control Area Chilled Water
- Diesel Generator Cooling Water
- Diesel Generator Engine Starting Air (CNS only)

**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.

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 uniform loss of material is accomplished using ultrasonic test techniques, supplemented by visual inspections if access to the interior surfaces is allowed such as during plant modifications.

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.

Localized corrosion due to pitting and MIC will reveal 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. A trend of indications of through-wall leaks due to pitting corrosion or MIC will provide 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. This relevant operating experience will form the basis for any future programmatic actions with respect to pitting corrosion and MIC concerns.

While the emphasis of the *Service Water Piping Corrosion Program* remains on gross material loss, the loss of material due to localized corrosion of component materials exposed to raw water will be managed 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 B - 47] 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.

**Operating Experience –** The *Service Water Piping Corrosion Program* was formalized at each site in the early 1990's. The early investigations were conducted as a part of the efforts to address NRC Generic Letter 89-13 [Reference B - 48]. Test results have indicated mostly pitting corrosion problems. Typical corrosion rates have ranged from 3 to 5 mils per year average wall loss, but vary depending on line size and flow regime. Test locations continue to be monitored and evaluated for continued service. Piping replacements have not been required to date based on corrosion rate projections.

Refinement of the predictive capabilities of the program have been made over time and now include monitoring and trending to determine calculated rate of material loss to schedule the next inspection. Operating experience has demonstrated that using measured corrosion rates provides adequate information on the extent of loss of material to predict when replacement of components might be necessary. Stress analysis of components has been able to refine acceptance criteria and extend the life of some pipe sections. Overall the program continues to successfully manage loss of material in the raw water systems of McGuire and Catawba.

***<u>Conclusion</u>***

The *Service Water Piping Corrosion Program* has been demonstrated to be capable of managing loss of material in components exposed to a raw water environment. The program described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.2.13 [Reference B - 5]. Based on the above review, the continued implementation of the *Service Water Piping Corrosion Program* provides reasonable assurance that the aging effect will be managed and that the structure(s) or component(s) will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

## B.3.30 STANDBY NUCLEAR SERVICE WATER POND DAM INSPECTION

*Note: The STANDBY NUCLEAR SERVICE WATER POND DAM INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Loss of material and cracking of earthen embankments have been identified as aging effects requiring management for the Standby Nuclear Service Water Pond Dam for the extended period of operation. The *Standby Nuclear Service Water Pond Dam Inspection* is credited with managing these aging effects. The *Standby Nuclear Service Water Pond Dam Inspection* is implemented per Technical Specification (SR) 3.7.8.3 for McGuire and Technical Specification (SR) 3.7.9.3 for Catawba. The *Standby Nuclear Service Water Pond Dam Inspection* is a condition monitoring program.

**Scope –** The scope of the *Standby Nuclear Service Water Pond Dam Inspection* includes the upstream and downstream slopes, the spillway overflow/outlet works, the area near the right and left abutments, and the toe of the dam.

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

**Parameters Monitored or Inspected –** Parameters monitored during the *Standby Nuclear Service Water Pond Dam Inspection* are cracking and loss of material. The examination guidelines for the inspection are in accordance with Regulatory Guide 1.127.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Standby Nuclear Service Water Pond Dam Inspection* will detect loss of material and cracking of the earthen embankment prior to loss of structure or component intended functions.

**Monitoring & Trending –** Aging effects are detected through visual examination of the dam. The *Standby Nuclear Service Water Pond Dam Inspection* visually examines the SNSW Pond Dam for erosion, settlement, slope stability, seepage, drainage systems, integrity of rip-rap, and environmental conditions. The *Standby Nuclear Service Water Pond Dam Inspection* is performed on an annual basis as required by McGuire Technical Specification (SR) 3.7.8.3 and Catawba Technical Specification (SR) 3.7.9.3. In addition, the results of the piezometric readings and settlement monitoring are reviewed. Piezometers are located on the dam to monitor foundation pore pressure. The piezometers are read quarterly. Survey monuments are located on the crest along the entire length of the dam to provide information on settlement. Surveys of the monuments are performed annually.

Inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. In particular, these records identify the inspection team and provide review of past inspection results, the results of the current inspection, whether the results were acceptable, discrepancies and their cause, and any corrective action resulting from the inspection.

**Acceptance Criteria –** Acceptance criteria are no visual indications of abnormal degradation, vegetation growth, erosion, or excessive seepage that would affect the Standby Nuclear Service Water Pond Dam operability.

**Corrective Action & Confirmation Process –** Structures and components which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repaired as required. Each inspection notes recommendations concerning repairs or studies. 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 *Standby Nuclear Service Water Pond Dam Inspection* is governed by the McGuire Technical Specification 3.7.8.3 and Catawba Technical Specification 3.7.9.3.

**Operating Experience –**

***McGuire Operating Experience***
*Standby Nuclear Service Water Pond Dam Inspections* have been performed for the Standby Nuclear Service Water (SNSW) Pond Dam and appurtenances since 1980. Previous inspections have revealed the dam to be in good condition. No conditions were identified that represent immediate danger to the safety and permanence of the SNSW Pond Dam and appurtenances. The inspection also includes a review of the instrumentation (piezometers, seepage monitor) for any abnormal conditions or trends. Available data show no significant changes in the trends for both the piezometers and observation wells. Each inspection performed a review of the previous inspection recommendations and documents the current status such as repair implemented or continual monitoring. The majority of the inspection recommendations were to spray the riprap on the upstream face and downstream toe of the dam to kill any vegetation, repair ruts, and re-seed. Structurally, cracks found in the vicinity of the concrete drainage ditch have been cleaned out and sealed with appropriate sealer.

In addition to the *Standby Nuclear Service Water Dam Inspection*, previous dam safety audits conducted by the NRC in 1994 and 1998 have concluded that there were no conditions observed that would indicate an immediate or adverse threat to the safety and permanence of the Standby Nuclear Service Water Pond Dam [References B - 49 and B - 50]. An independent consultant also performs inspections every five years per NCUC Docket No. E-100, Subpart 23. These inspections have also revealed similar findings.

***Catawba Operating Experience***

The Standby Nuclear Service Water Pond Dam is inspected annually in compliance with Technical Specification Surveillance Requirement 3.7.9.3. Based upon review of the results of previous inspections, no conditions were observed which have adverse effects on the intended function of the Standby Nuclear Service Water Pond Dam. Most common recurring recommendations are to clear vegetation from the concrete drainage ditches, pack soil and gravel along the sides of the concrete drainage ditch, and monitor any signs of erosion along the sides of the concrete drainage ditch. Most corrective items are routine and are addressed by regularly scheduled maintenance activities.

In addition to the credited *Standby Nuclear Service Water Pond Dam Inspection*, dam safety audits conducted by the NRC in 1997 and 1999 have concluded that there were no conditions that would indicate an immediate or adverse threat to the safety and permanence of the Standby Nuclear Service Water Pond Dam [References 51 and 52].

***Conclusion***

The *Standby Nuclear Service Water Pond Dam Inspection* has been demonstrated to be capable of detecting and managing loss of material and cracking of the dams. Based on the above review, the continued implementation of the *Standby Nuclear Service Water Pond Dam Inspection* provides reasonable assurance that loss of material and cracking will be managed such that the intended functions of the dams will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.31 STEAM GENERATOR SURVEILLANCE PROGRAM

*Note: The STEAM GENERATOR SURVEILLANCE PROGRAM is generically applicable to both McGuire and Catawba Nuclear Stations, except as otherwise noted.*

The purpose of the *Steam Generator Surveillance Program* is to provide comprehensive examinations of the steam generator tubes to ensure that degradation is identified and corrective actions are taken prior to exceeding allowable limits. The *Steam Generator Surveillance Program* is a condition monitoring program that is credited with managing loss of material and cracking of Alloy 600 and 690 steam generator tubes.

**Scope –** The scope of the *Steam Generator Surveillance Program* includes all steam generator tubes (including plugs and sleeves) in each steam generator and internal support structures.

**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 *Steam Generator Surveillance Program* monitors steam generator tube wall degradation and support plate locations.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Steam Generator Surveillance Program* will detect loss of material and cracking prior to loss of component intended functions.

**Monitoring & Trending –** The minimum frequency of inspection for the steam generator tubes is specified in Technical Specification 5.5.9.3, Inspection Frequencies.

The minimum sample size of steam generator tubes to be inspected is specified in Technical Specification 5.5.9.2, Steam Generator Tube Sample Selection and Inspection.

In addition to the Technical Specification requirements, inspection of the steam generators follow the recommendations of the NEI 97-06 [Reference B - 53] and EPRI PWR Steam Generator Examination Guidelines [Reference B - 54].

Inspection of tubes and rolled plugs are done by eddy current examination. Tube plugs that cannot be examined by eddy current are visually inspected.

**Acceptance Criteria –** The acceptance criteria for the *Steam Generator Surveillance Program* is provided in Technical Specification 5.5.9.4, Acceptance Criteria. In addition, data are evaluated to determine that all structural and leakage criteria were met during the past operating cycle and a projection is made to determine that all tubes left in service will continue to meet operability requirements until the next examination.

**Corrective Action & Confirmation Process –** Corrective actions for the *Steam Generator Surveillance Program* are specified in Table 5.5-2, Steam Generator Tube Inspection of the Technical Specifications. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

**Administrative Controls –** The *Steam Generator Surveillance Program* is implemented using approved engineering programs and procedures and meets the minimum requirements of Technical Specification 5.5.9 for both McGuire and Catawba Nuclear Stations.

**Operating Experience –**

***McGuire Operating Experience***
The McGuire Steam Generators were replaced in May, 1997 for Unit 1 and December, 1998 for Unit 2. Since the replacement of the steam generators the only mechanism of tube degradation that has been identified in the McGuire Units 1 and 2 steam generators is wear at the secondary side U-bend fan bar and lattice grid supports.

***Catawba Operating Experience***
The Catawba Unit 1 steam generators were replaced in October, 1996. Wear has been identified at the secondary side U-bend fan bar supports.

The Catawba Unit 2 steam generators have not been replaced. The degradation mechanism that has been identified in the Unit 2 steam generators is wear. Wear has occurred at the edge of anti vibration bars and in the preheater section. Wear at the secondary side structures is a slow process and is readily detectable by eddy current testing before it is severe enough to detect tube structural integrity. Current wear rate is <5% per cycle for anti vibration bars and pre-heater tubes based on a review of eddy current data.

***Conclusion***

The *Steam Generator Surveillance Program* has been demonstrated to be capable of ensuring that steam generator degradation is identified and corrective actions are taken prior to exceeding allowable limits. The *Steam Generator Surveillance Program* described above is equivalent to the *Steam Generator Tube Surveillance Program* described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *Steam Generator Surveillance Program* provides reasonable assurance that the aging effects will be managed and that the steam generator will continue to perform its intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

### B.3.32 SUMP PUMP SYSTEMS INSPECTION

*Note: The SUMP PUMP SYSTEMS INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Sump Pump Systems Inspection* is to characterize any loss of material of the internal and external surfaces of a limited set of mechanical components exposed to sump environments. Sump environments may contain leakage from a variety of systems but are considered to be raw water environments with alternate wetting and drying as sump levels change. Uncertainty exists as to whether long term exposure to these sump environments could cause loss of material of system components such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect components constructed of various materials to detect the presence and extent of any loss of material from exposure to raw water, including alternate wetting and drying. The *Sump Pump Systems Inspection* is a one-time inspection.

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

- Diesel Generator Room Sump Pump System
- Conventional Waste Water Treatment System (MNS Only)
- Groundwater Drainage System
- Turbine Building Sump Pump System (CNS Only)

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

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

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

**Operating Experience –** The *Sump Pump Systems Inspection* is a new one-time inspection for which there is no operating experience.

***Conclusion***

Based on the above review, the implementation of the *Sump Pump Systems Inspection* will adequately verify that no need exists to manage the aging effects on the components or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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## B.3.33 TECHNICAL SPECIFICATION SR 3.6.16.3 VISUAL INSPECTION

*Note: The TECHNICAL SPECIFICATION SR 3.6.16.3 VISUAL INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Change in material property due to leaching has been identified as an aging effect requiring programmatic management for the walls and dome of the concrete Reactor Building for the extended period of operation. Technical Specification SR 3.6.16.3 requires that a visual inspection be performed on the exposed interior and exterior surfaces of the Reactor Building three times every ten years. The purpose of such visual inspections is to uncover evidence of deterioration which could affect the Reactor Building structural integrity. The *Technical Specification SR 3.6.16.3 Visual Inspection* is a condition monitoring program.

**Scope –** The scope of the *Technical Specification SR 3.6.16.3 Visual Inspection* includes accessible surface areas of the walls and dome of the concrete Reactor Building.

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

**Parameters Monitored or Inspected –** The parameter monitored by the *Technical Specification SR 3.6.16.3 Visual Inspection* is change in material property due to leaching.

**Detection of Aging Effects –** In accordance with information provided in **Monitoring & Trending**, the *Technical Specification SR 3.6.16.3 Visual Inspection* will detect change in material properties due to leaching prior to loss of structure or component intended function.

**Monitoring & Trending –** Loss of material due to leaching is detected through visual examination. From the Technical Specification bases, this surveillance requirement provides advance indication of deterioration of the concrete structural integrity of the Reactor Building. The frequency of the inspection is three times every ten years. *Technical Specification SR 3.6.16.3 Visual Inspection* does not include a requirement to monitor or trend degradation. Unacceptable conditions are noted for correction or further evaluation.

**Acceptance Criteria –** Acceptance criteria are based on visual indication of structural damage or degradation.

For concrete, the acceptance criterion is no unacceptable indication of change in material property due to leaching.

**Corrective Action & Confirmation Process –** Structures and components which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repaired as required. Structures and components which are deemed unacceptable are documented under the corrective action program. Specific corrective actions are implemented in accordance with the corrective action program. Confirmatory actions, as needed, are implemented as part of the corrective action program.

**Administrative Controls –** McGuire and Catawba Technical Specifications SR 3.6.16.3 govern these inspections. The *Technical Specification SR 3.6.16.3 Visual Inspection* is implemented by plant procedures as required by Technical Specification 5.4.

**Operating Experience –** *Technical Specification SR 3.6.16.3 Visual Inspections* have been performed at the specified frequencies since initial operation. The inspection results are documented in station procedures. These inspections have revealed only minor degradation of concrete at McGuire and Catawba. Observations include minor hairline surface cracking and minor leaching. Leaching has been observed on the interior of the reactor building domes at McGuire near the dome-to-shell interface, and maintenance has been planned for the dome exterior to minimize water intrusion. Adverse conditions are reinspected during subsequent inspections. The observed aging effects are relatively minor and have no impact on the ability of the concrete Reactor Building to perform its intended functions.

***Conclusion***

The *Technical Specification SR 3.6.16.3 Visual Inspection* has been demonstrated to be effective in managing aging of the concrete Reactor Building during the period of extended operation. Based on the above review, the continued implementation of the *Technical Specification SR 3.6.16.3 Visual Inspection* provides reasonable assurance that change in material property due to leaching of Reactor Building concrete will be managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

### B.3.34 TREATED WATER SYSTEMS STAINLESS STEEL INSPECTION

*Note: The TREATED WATER SYSTEMS STAINLESS STEEL INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Treated Water Systems Stainless Steel Inspection* is to characterize any loss of material or cracking of stainless steel components resulting from exposure to unmonitored treated water environments. An unmonitored treated water environment is one that may contain conditions that can concentrate existing levels of contaminants or that may simply start with a higher level of contaminants than those systems routinely monitored by the Chemistry Control Program. Examples of contaminants are halogens, sulfates, and dissolved oxygen. Uncertainty exists as to whether exposure of stainless steel components located in an unmonitored treated water environment could lead to loss of material or cracking such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect stainless steel components to detect the presence and extent of any loss of material or cracking. The *Treated Water Systems Stainless Steel Inspection* is a one-time inspection.

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

- Containment Valve Injection Water (CNS only)
- Drinking Water (CNS only)
- Nuclear Solid Waste Disposal (MNS only)
- Solid Radwaste (CNS only)

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

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

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

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

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

**Operating Experience –** The *Treated Water Systems Stainless Steel Inspection* is a one-time inspection activity for which there is no operating experience. However, a similar inspection activity was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

***Conclusion***

The *Treated Water Systems Stainless Steel Inspection* described above is similar to the corresponding activity described and evaluated in NUREG-1723, Section 3.2.11 [Reference B - 5]. Based on the above review, implementation of the *Treated Water Systems Stainless Steel Inspection* will adequately verify that no need exists to manage the aging effects on the component or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.35 UNDERWATER INSPECTION OF NUCLEAR SERVICE WATER STRUCTURES

*Note: The UNDERWATER INSPECTION OF NUCLEAR SERVICE WATER STRUCTURES is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

Loss of material due to corrosion of steel components in fluid environments and loss of material and cracking of concrete components have been identified as aging effects requiring management for Nuclear Service Water Structures and the Low Pressure Service Water Intake Structure at Catawba. The structures at McGuire which are exposed to pond water are the Standby Nuclear Service Water Intake and Discharge structures. The structures at Catawba which are exposed to lake or pond water are the Standby Nuclear Service Water Intake Structure, the Standby Nuclear Service Water Discharge Structures, the Standby Nuclear Service Water Pond Outlet, the Nuclear Service Water Pump Structure, the Nuclear Service Water Intake Structure, and the Low Pressure Service Water Intake Structure. The *Underwater Inspection of Nuclear Service Water Structures* is credited with managing loss of material of steel and loss of material and cracking for concrete for the period of extended operation. The *Underwater Inspection of Nuclear Service Water Structures* is a condition monitoring program.

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

McGuire Nuclear Station
- Standby Nuclear Service Water Discharge Structures
- Standby Nuclear Service Water Intake Structure

Catawba Nuclear Station
- 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 performed every five years at McGuire and every Unit 1 refueling outage for Catawba Nuclear Service Water and Standby Nuclear Service Water Intake structures and five years for other Catawba structures. No actions are taken as part of this program to trend inspection or test results.

The underwater inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. The diver report includes the results of the current inspection or findings. The accountable engineer is responsible for reviewing the findings and determining whether or not the results are acceptable.

**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.

**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.

**Operating Experience –**

***McGuire Operating Experience***
The *Underwater Inspection of Nuclear Service Water Structures* has been performed for the Standby Nuclear Service Water (SNSW) Intake and Discharge structures since 1989.

In 1992 the SNSW Intake trash racks were replaced. An inspection of the SNSW Intake Structure at that time had revealed deterioration of the trash racks and fasteners. The old trash racks and fasteners were made of galvanized steel. The new trash racks are made of stainless steel.

Review of previous *Underwater Inspection of Nuclear Service Water Structures* reports indicates the SNSW Intake and Discharge structures are in good working condition. Observations included good concrete condition, acceptable silt level, minor biofouling, and intact trash rack and fasteners.

***Catawba Operating Experience***

Previous *Underwater Inspection of Nuclear Service Water Structures* have revealed only minor degradation. No deterioration that could cause loss of intended function has been identified from the previous inspections.

***<u>Conclusion</u>***

The *Underwater Inspection of Nuclear Service Water Structures* has been demonstrated to be capable of detecting and managing loss of material for steel components and loss of material and cracking for concrete components. The *Underwater Inspection of Nuclear Service Water Structures* described above is equivalent to the Duke Power 5-Year Underwater Inspection of Hydroelectric Dams and Appurtenances described and evaluated in NUREG-1723, Section 3.2.4 [Reference B - 5]. Based on the above review, the continued implementation of the *Underwater Inspection of Nuclear Service Water Structures* provides reasonable assurance that the aging effects will be managed such that the intended functions of the structures and components will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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### B.3.36 WASTE GAS SYSTEM INSPECTION

*Note: The WASTE GAS SYSTEM INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.*

The purpose of the *Waste Gas System Inspection* is to characterize loss of material and cracking in Waste Gas System components resulting from exposure to unmonitored treated water and gas environments. Unmonitored treated water is condensation of the water vapor contained in the waste gas stream and effluent from the recombiners and separators. The gas environment is a combination of nitrogen, hydrogen, oxygen, and fission product gases. Uncertainty exists as to whether exposure to these environments could cause loss of material or cracking of the Waste Gas System components such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect Waste Gas System components to detect the presence and extent of loss of material and cracking from exposure to unmonitored treated water and gas environments. The *Waste Gas System Inspection* is a one-time 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 McGuire and 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 system engineer.

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

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

**Operating Experience –** The *Waste Gas System Inspection* is a one-time inspection activity for which there is no operating experience.

***Conclusion***

Based on the above review, implementation of the *Waste Gas System Inspection* will adequately verify that no need exists to manage the aging effects on the components or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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