Appendix A

UFSAR Supplements

Appendix A1 – McGuire Nuclear Station UFSAR Supplement

Appendix A2 – Catawba Nuclear Station UFSAR Supplement

Appendix A
UFSAR Supplements for
McGuire Nuclear Station and Catawba Nuclear Station

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Appendix A
UFSAR Supplements for
McGuire Nuclear Station and Catawba Nuclear Station

INTRODUCTION

Duke Energy Corporation (Duke) has prepared an Application for Renewed Operating Licenses of McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2 (Application). The complete application includes sufficient information for the NRC to complete their technical and environmental reviews and provides the basis for the NRC to make the findings required by §54.29.

Appendix A of the Application contains the UFSAR Supplements for both stations. Appendix A-1 contains the UFSAR Supplement for McGuire Nuclear Station and Appendix A-2 contains the UFSAR Supplement for Catawba Nuclear Station.

§54.21(d) An FSAR Supplement

> *The FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by paragraphs (a) and (c) of this section, respectively.*

The Application contains the technical information required by §§54.21(a) and (c). Appendix B of the Application provides descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Chapter 4 of the Application contains the evaluations of the time-limited aging analyses for the period of extended operation. Information contained in both of these locations of the Application has been used to prepare the program and activity descriptions that are contained in both of the attached UFSAR Supplements. In addition, the Oconee UFSAR Supplement, which was provided by Duke letter dated March 28, 2000 and accepted by the NRC staff as meeting the requirements of Part 54, § 54.21(d), was used as guidance in the preparation of the McGuire UFSAR Supplement (Appendix A-1) and the Catawba UFSAR Supplement (Appendix A-2).

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Appendix A-1

McGuire Nuclear Station UFSAR Supplement

Changes to Existing Chapters 3 and 5

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Changes to Existing Chapters 3 and 5

Make the following changes to Section 3.5.2.1, Reactor Coolant Pump Flywheel:

Delete the following paragraphs:
~~A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:~~

1. ~~.Maximum tangential stress at an assumed overspeed of 125 percent.~~
2. ~~.A through crack through the thickness of the flywheel at the bore.~~
3. ~~.400 cycles of startup operation in 40 years.~~

~~Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030 in. 0.060 in. per 1000 cycles.~~

Insert the following paragraph:

Evaluation for License Renewal
To estimate the magnitude of fatigue crack growth during plant life, an initial radial crack length of 10 % of the distance through the flywheel (from the keyway to the flywheel outer radius) was conservatively assumed. The analysis assumed 6000 cycles of pump starts and stops for a 60-year plant life. The existing analysis is valid for the period of extended operation.

Changes to Existing Chapters 3 and 5

Add the following paragraph to Section 3.9.2, ASME Code Class 2 and 3 Components:

Evaluation for License Renewal

McGuire has a number of systems that were designed to ASME Code Class 2 and 3. Piping analyses for these systems include stress range reduction factors to provide conservatism in the design to account for thermal cyclic operations. Thermal fatigue of mechanical systems designed to ASME Code Class 2 and 3 is considered to be a time-limited aging analysis because all six of the criteria contained in 10 CFR 54.3 are satisfied. From the license renewal review, it was determined that the analyses of thermal fatigue of these mechanical systems are valid for the period of extended operation.

Changes to Existing Chapters 3 and 5

Add the following paragraph to Section 3.11, Environmental Design of Mechanical and Electrical Equipment:

Evaluation for License Renewal
Some qualification analyses for safety-related equipment identified in Section 3.11.1.1 were found to be time-limited aging analyses for license renewal. The existing EQ process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation.

Changes to Existing Chapters 3 and 5

Add the following paragraphs to Section 5.2.1, Design of Reactor Coolant Pressure Boundary Components:

Fatigue Evaluation for License Renewal

McGuire Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 5.2.1 to assure that components are maintained within design limits. This requirement is managed by the McGuire *Thermal Fatigue Management Program*. For license renewal, continuation of the McGuire *Thermal Fatigue Management Program* into the period of extended operation will provide reasonable assurance that the thermal fatigue analyses, including applicable flaw growth calculations, will remain valid or that appropriate action is taken in a timely manner to assure continued validity of the design.

Leak-Before-Break Evaluation for License Renewal

Leak-before-break analyses evaluate postulated flaw growth in the primary loop piping of the Reactor Coolant System. These analyses consider the thermal aging of the cast austenitic stainless steel material of the piping as well as the fatigue transients that drive the flaw growth over the operating life of the plant. Because all of the criteria contained in §54.3 are met, leak before break is a TLAA for McGuire. The leak before break analyses have been determined to be acceptable for the period of extended operation.

Add the following paragraphs to Section 5.4.3, Evaluation [Reactor Vessel and Appurtenances]:

Pressurized Thermal Shock Evaluation for License Renewal

The requirements of 10 CFR 50.61 are to protect against pressurized thermal shock transients in pressurized-water reactors. The screening criterion established by §50.61 is 270°F for plates, forgings, and axial welds. The screening criterion is 300°F for circumferential welds. According to this regulation, if the calculated RT_{PTS} for the limiting reactor beltline materials is less than the specified screening criterion, then the vessel is acceptable with regard to the risk of vessel failure during postulated pressurized thermal shock transients. The regulations require updating of the pressurized thermal shock assessment upon a request for a change in the expiration date of the facility operating license. The RT_{PTS} calculations are time-limited aging analyses because all six of the criteria contained in 10 CFR 54.3 are met. The RT_{PTS} values have been projected to the end of the period of extended operation using the methods provided in §50.61.

The RT_{PTS} results for all beltline materials are presented in Table 5-W for McGuire Unit 1 and in Table 5-X for McGuire Unit 2. All the beltline materials in the McGuire reactor vessels have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at 54 EFPY. The lower shell plate longitudinal welds 3-442 A and C are the most limiting material for McGuire Unit 1 with a 54 EFPY PTS value of 225°F. The lower shell forging 04 is the most limiting material for McGuire Unit 2 with a 54 EFPY PTS value of 152°F.

Table 5-W RT PTS Calculations for McGuire Unit 1 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY	FF	$RT_{NDT(U)}$	ΔRT_{PTS}	M	RT_{PTS} °F
Intermediate Shell Plate B5012-1	74.2	3.07	1.296	34	96.2	34	164
→ Using Surveillance Capsule Data	62.5	3.07	1.296	34	81.0	17	132
Intermediate Shell Plate B5012-2	100.3	3.07	1.296	0	130.0	34	164
Intermediate Shell Plate B5012-3	74.9	3.07	1.296	-13	97.1	34	118
Lower Shell Plate B5013-1	99.1	3.07	1.296	0	128.4	34	162
Lower Shell Plate B5013-2	65	3.07	1.296	30	84.2	34	148
Lower Shell Plate B5013-3	65	3.07	1.296	15	84.2	34	133
Intermediate Shell Plate Longitudinal Weld Seams 2-442A (0° Azimuth)	201.3	1.89	11.7	-50	235.5	56	242
→ Using Surveillance Capsule Data	156.5	1.89	1.17	-50	183.1	28	161
Intermediate Shell Plate Longitudinal Weld Seams 2-442 B, C (30° Azimuth)	201.3	2.73	1.27	-50	255.7	56	262
→ Using Surveillance Capsule Data	156.5	2.73	1.27	-50	198.8	28	177
Lower Shell Plate Longitudinal Weld Seams 3-442 A, C (30° Azimuth)	208.2	2.73	1.27	-50	264.4	56	270
→ Using Surveillance Capsule Data	194.4	2.73	1.27	-50	246.9	28	225
Lower Shell Plate Longitudinal Weld Seams 3-442 B (0° Azimuth)	208.2	1.89	1.17	-50	243.6	56	250
→ Using Surveillance Capsule Data	194.4	1.89	1.17	-50	227.4	28	205
Intermediate to Lower Shell Plate Circumferential Weld Seam 9-442	37.5	3.07	1.296	-70	48.6	48.6	27

Table 5-X RT PTS Calculations for McGuire Unit 2 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY	FF	$RT_{NDT(U)}$	Δ RT_{PTS}	M	RT_{PTS} °F
Intermediate Shell Forging 05	117	2.88	1.28	-4	149.8	34	180
→ Using Surveillance Capsule Data	84	2.88	1.28	-4	107.5	17	121
Lower Shell Forging 04	115.8	2.88	1.28	-30	148.2	34	152
Circumferential Weld Metal	52.7	2.88	1.28	-68	67.5	56	56
→ Using Surveillance Capsule Data	31.5	2.88	1.28	-68	40.3	28	0

Changes to Existing Chapters 3 and 5

Add the following paragraphs to Section 5.4.3, Evaluation [Reactor Vessel and Appurtenances]:

Upper Shelf Energy Evaluation for License Renewal
Appendix G of 10 CFR Part 50 requires that reactor vessel beltline materials must have a Charpy Upper Shelf Energy (USE) of no less than 75 ft-lb and must maintain a Charpy USE of no less than 50 ft-lb throughout the life of the reactor vessel, unless it is demonstrated, in a manner approved by the Director, Office of Nuclear Reactor Regulation (NRR), that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. The USE calculations are time-limited aging analyses because all six of the criteria contained in 10 CFR 54.3 are met. The USE analyses for each vessel have been projected to the end of the period of extended operation using the guidance provided in Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials.*

The USE values for McGuire Units 1 and 2 reactor vessel beltline materials at 54 EFPY are presented in Table 5-Y for McGuire Unit 1 and in Table 5-Z for McGuire Unit 2. All of the beltline materials in the McGuire reactor vessels have USE above the 50 ft-lb limit. The nozzle shell plate B5011-2 is the most limiting material for McGuire Unit 1 with a 54 EFPY USE value of 53 ft-lbs. The nozzle shell to intermediate shell weld is the most limiting material for McGuire Unit 2 with a 54 EFPY USE value of greater than 55 ft-lbs.

Table 5-Y Evaluation of Upper Shelf Energy for McGuire Unit 1 Beltline Region Materials at 54 EFPY

Material	Weight % of Cu	¼ T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected USE @ 54 EFPY (ft-lb)
Intermediate Shell Plate B5012-1	0.11	1.83	101	11	90
Intermediate Shell Plate B5012-2	0.14	1.83	105	26	78
Intermediate Shell Plate B5012-3	0.11	1.83	112	23	86
Lower Shell Plate B5013-1	0.14	1.83	95	26	70
Lower Shell Plate B5013-2	0.10	1.83	115	22	90
Lower Shell Plate B5013-3	0.10	1.83	103	22	80
Nozzle Shell Plate B5453-2	0.14	1.83	72.4	26	54
Nozzle Shell Plate B5011-2	0.10	1.83	68.3	22	53
Nozzle Shell Plate B5011-3	0.13	1.83	94.7	25	71
Nozzle Shell Longitudinal Weld Seams 1-422A, B, C	0.199	1.63 1.13 1.63	112	38 35 38	69 73 69
Nozzle Shell to Intermediate Shell Circumferential Weld Seam	0.183	1.83	109	40	65
Intermediate Shell Longitudinal Weld Seams 2-442A, B, C	0.199	1.13 1.63 1.63	112	33 36 36	75 72 72
Intermediate Shell to Lower Shell Circumferential Weld Seam	0.051	1.83	143	22	112
Lower Shell Longitudinal Weld Seams 3-442A, B, C	0.213	1.63 1.13 1.63	124	40 37 40	74 78 74

Table 5-Z Evaluation of Upper Shelf Energy for McGuire Unit 2 Beltline Region Materials at 54 EFPY

Material	Weight % of Cu	¼ T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected USE @ 54 EFPY (ft-lb)
Nozzle Shell Forging 06	0.25	1.73	98	38	61
Intermediate Shell Forging 05 (Using Surveillance Capsule Data)	0.153	1.73	94	24	71
Lower Shell Forging 04	0.15	1.73	141	28	102
Intermediate to Lower Shell Circumferential Weld (Using Surveillance Capsule Data)	0.039	1.73	132	3.5	127
Nozzle Shell to Intermediate Shell Weld	0.11	1.73	>71	23	>55

Changes to Existing Chapters 3 and 5

Add the following paragraph to Section 5.4.3, Evaluation [Reactor Vessel and Appurtenances]:

Pressure – Temperature Limits Evaluation for License Renewal
Appendix G of 10 CFR Part 50 requires heatup and cooldown of the reactor pressure vessel be accomplished within established pressure-temperature limits. These limits are established by calculations that utilize the materials and fluence data obtained through the unit specific reactor surveillance capsule program. Normally, the pressure-temperature limits are calculated for several years into the future and remain valid for an established period of time not to exceed the current operating license expiration. Calculations for the pressure-temperature limit curves for McGuire have been performed on each reactor vessel to address projected operation during the period of extended operation. For McGuire Unit 1 and Unit 2, the heatup and cooldown limit curves for normal operation at 50.3 EFPY provide a predicted operating window that is sufficient to conduct heatups and cooldowns.

Changes to Existing Chapters 3 and 5

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New Chapter 18

Insert new UFSAR Chapter 18 to read as follows:

18.0 Aging Management Programs and Activities

18.1 INTRODUCTION

Duke Energy Corporation prepared an Application for Renewed Operating Licenses of McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2 (Application) [Reference 18 - 1]. The application, including information provided in additional correspondence, provides sufficient information for the NRC to complete their technical and environmental reviews and provides the basis for the NRC to make the findings required by §54.29 (Final Safety Evaluation Report – Final SER) [Reference 18 - 2]. Pursuant to the requirements of §54.21(d), the UFSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by §54.21 (a) and (c), respectively.

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

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

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

Table 18-1 Summary Listing of the Programs, Activities and TLAA

Topic	***Application Location***	***UFSAR / ITS Location***
Alloy 600 Aging Management Review	B.3.1	18.2.1
Battery Rack Inspections	B.3.2	ITS SR 3.8.4.3 SLC 16.8.3.3 SLC 16.9.7.12 SLC 16.9.7.17
Boraflex Monitoring Program	B.3.3	SLC 16.9.24
Borated Water Systems Stainless Steel Inspection	B.3.4	18.2.2
Bottom-Mounted Instrumentation Thimble Tube Inspection Program	B.3.5	18.2.3
Chemistry Control Program	B.3.6	18.2.4
Containment Inservice Inspection Plan – IWE	B.3.7	18.2.5
Containment Leak Rate Testing Program	B.3.8	ITS 3.6.1 ITS 5.5.2
Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetrations Inspection Program	B.3.9	18.2.6
Crane Inspection Program	B.3.10	18.2.7
Divider Barrier Seal Inspection and Testing Program	B.3.11	ITS SR 3.6.14.2 ITS SR 3.6.14.4 ITS SR 3.6.14.5
Environmental Qualification	4.4	3.11

Table 18-1 Summary Listing of the Programs, Activities and TLAA (continued)

Topic	***Application Location***	***UFSAR / ITS Location***
Fire Protection Program	B.3.12	SLC 16.9.1 SLC 16.9.2 SLC 16.9.4 SLC 16.9.5 18.2.8
Flood Barrier Inspection	B.3.13	18.2.9
Flow Accelerated Corrosion Program	B.3.14	18.2.10
Fluid Leak Management Program	B.3.15	18.2.11
Galvanic Susceptibility Inspection	B.3.16	18.2.12
Heat Exchanger Activities	B.3.17	18.2.13
Ice Condenser Inspections	B.3.18	ITS 3.6.12 18.2.14
Inaccessible Non-EQ Medium Voltage Cables Aging Management Program	B.3.19	18.2.15
Inservice Inspection Plan	B.3.20	18.2.16
Inspection Program for Civil Engineering Structures and Components	B.3.21	18.2.17
Leak-Before-Break	4.7.2	5.2.1
Liquid Waste System Inspection	B.3.22	18.2.18
Metal Fatigue	4.3	5.2.1 and 3.9.2
Non-EQ Insulated Cables and Connections Aging Management Program	B.3.23	18.2.19

Table 18-1 Summary Listing of the Programs, Activities and TLAA (continued)

Topic	***Application Location***	***UFSAR / ITS Location***
Preventive Maintenance Activities	B.3.24	18.2.20
Reactor Coolant Pump Flywheel	4.7.1	3.5.2.1
Reactor Coolant System Operational Leakage Monitoring Program	B.3.25	ITS 3.4.13 ITS 3.4.15
Reactor Vessel Integrity Program	B.3.26	18.2.21
Reactor Vessel Internals Inspection	B.3.27	18.2.22
Reactor Vessel Neutron Embrittlement	4.2	5.4.3
Selective Leaching Inspection	B.3.28	18.2.23
Service Water Piping Corrosion Program	B.3.29	18.2.24
Standby Nuclear Service Water Pond Dam Inspection	B.3.30	ITS SR 3.7.8.3
Steam Generator Surveillance Program	B.3.31	ITS 5.5.9
Sump Pump Systems Inspection	B.3.32	18.2.25
Technical Specification SR 3.6.16.3 Visual Inspection	B.3.33	ITS SR 3.6.16.3
Treated Water Systems Stainless Steel Inspection	B.3.34	18.2.26
Underwater Inspection of Nuclear Service Water Structures	B.3.35	18.2.27
Waste Gas System Inspection	B.3.36	18.2.28

18.2 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

18.2.1 ALLOY 600 AGING MANAGEMENT REVIEW

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* or other existing programs such as the *Control Rod Drive Mechanism and Other Vessel Head Penetration Program,* the *Reactor Vessel Internals Inspection,* and the *Steam Generator Integrity Program.* 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* or other existing programs such as the *Control Rod Drive Mechanism and Other Vessel Head Penetration Program,* the *Reactor Vessel Internals Inspection, and the Steam Generator Integrity Program.* 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). 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.

18.2.2 BORATED WATER SYSTEMS STAINLESS STEEL INSPECTION

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

- Containment Spray
- Refueling Water

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

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

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

Monitoring & Trending – The *Borated Water Systems Stainless Steel Inspection* will inspect stainless steel components, welds, and heat affected zones, as applicable, in the Containment Spray System in the area of the internal air/water interface. The borated water environment found downstream of valves NS-12, 15, 29, 32, 38, and 43 in the Containment Spray System at McGuire is stagnant and isolated from the remainder of the system, and therefore, not controlled by the Chemistry Control Program. Water from the refueling water storage tank is introduced during valve testing with level in the piping reaching the same elevation as the tank. Since the pipe is open to containment, evaporation occurs and concentration of contaminants could occur at the air/water interface. This concentration of contaminants could lead to loss of material or cracking. Therefore, a one-time inspection around this water line is warranted.

One of twelve possible locations at McGuire will be inspected using a volumetric technique. If no parameters are known that would distinguish the susceptible locations, one of the twelve available at McGuire will be examined based on accessibility and radiological concerns. The results of this inspection 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).

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

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

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

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

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

18.2.3 BOTTOM-MOUNTED INSTRUMENTATION THIMBLE TUBE INSPECTION PROGRAM

Scope – The scope of the *Bottom Mounted Instrumentation Thimble Tube Inspection Program* includes all 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 18 - 4]. 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 18 - 3].

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 18 - 4].

18.2.4 CHEMISTRY CONTROL PROGRAM

The purpose of the *Chemistry Control Program* is to manage loss of material and/or cracking of components exposed to borated water, closed cooling water, fuel oil, and treated water environments. This program manages the relevant conditions that lead to the onset and propagation of loss of material 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.

18.2.5 CONTAINMENT INSERVICE INSPECTION PLAN – IWE

The *Containment Inservice Inspection Plan – IWE* was developed to implement applicable requirements of 10 CFR 50.55a. Section 50.55a(g)(4) requires that throughout the service life of nuclear power plants, components which are classified as either Class MC or Class CC pressure retaining components and their integral attachments must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME Code and Addenda that are incorporated by reference in §50.55a(b). Furthermore, §50.55a(g)(4)(v)(A) requires that metal containment pressure retaining components and their integral attachments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC. These requirements are subject to the limitation listed in paragraph (b)(2)(vi) and the modifications listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of §50.55a, to the extent practical within the limitations of design, geometry and materials of construction of the components [Reference 18 - 5].

18.2.6 CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE PENETRATIONS INSPECTION PROGRAM

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).

Due to length of time in operation, it is expected that Unit 1 results will provide a leading indicator for Unit 2. 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 18 - 6].

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*. 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.

18.2.7 CRANE INSPECTION PROGRAM

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

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

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

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

Monitoring & Trending – The *Crane Inspection Program* detects aging effects through visual examination of the crane rails and girders. No actions are taken as part of this program to trend inspection or test results.

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

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

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

18.2.8 FIRE PROTECTION PROGRAM

Elements of the *Fire Protection Program* that serve to manage aging are implemented in accordance with Selected Licensee Commitments (See Table 18-1). Additional aging management of fouling of sprinkler branch lines that do not receive flow during periodic testing 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).

18.2.9 FLOOD BARRIER INSPECTION

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

18.2.10 FLOW ACCELERATED CORROSION PROGRAM

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

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

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

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

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

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

Monitoring & Trending – The program is consistent with the basic guidelines or recommendations provided by EPRI document NSAC-202L [Reference 18 - 7]. Component wall thickness is measured using volumetric examinations such as ultrasonic testing and radiography. Visual examinations are also employed when access to interior surfaces is allowed by component design. Component wall thickness acceptability is judged in accordance with the McGuire component design code of record.

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

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

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

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

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

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

18.2.11 FLUID LEAK MANAGEMENT PROGRAM

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.

Preventive Actions – The programmatic implementation of the *Fluid Leak Management Program* is accomplished through visual surveillance and systematic trending of findings. Walkdowns of the Auxiliary and Reactor Buildings are conducted at the start of each refueling outage for the purpose of identifying leakage or evidence of leakage from borated water systems. All active leaks are monitored on an appropriate frequency depending on accessibility and rate of leakage.

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

Detection of Aging Effects – 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 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 18 - 8] 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 18 - 9]. 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.

18.2.12 GALVANIC SUSCEPTIBILITY INSPECTION

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

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

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

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

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

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

Monitoring & Trending – The *Galvanic Susceptibility Inspection* will inspect a select set of carbon steel-stainless steel couples at McGuire using a volumetric examination technique. 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).

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

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

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

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

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

18.2.13 HEAT EXCHANGER ACTIVITIES

18.2.13.1 COMPONENT COOLING HEAT EXCHANGERS

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

The purpose of the *Heat Exchanger Preventive Maintenance Activities – Component Cooling* is to manage loss of material for parts of the component cooling heat exchanger exposed to raw water. The *Heat Exchanger Preventive Maintenance Activities– Component Cooling* is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary function. The program is credited with managing loss of material for admiralty brass, carbon steel, and stainless steel materials.

18.2.13.2 CONTAINMENT SPRAY HEAT EXCHANGERS

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

The purpose of the *Heat Exchanger Preventive Maintenance Activities – Containment Spray* is to manage loss of material for parts of the containment spray heat exchanger exposed to raw water. The *Heat Exchanger Preventive Maintenance Activities – Containment Spray* is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary function. The program is credited with managing loss of material for stainless steel and titanium materials.

18.2.13.3 DIESEL GENERATOR ENGINE COOLING WATER HEAT EXCHANGERS

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

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

18.2.13.4 CONTROL AREA CHILLED WATER

The purpose of the *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* is to manage fouling and loss of material of parts of the control room area chillers exposed to raw water. The *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* is a condition monitoring program that monitors specific component parameters to detect the presence and assess the extent of material loss that can affect the pressure boundary functions and periodically cleans the chiller tubes to manage fouling. The *Heat Exchanger Preventive Maintenance Activities – Control Area Chilled Water* is credited with managing loss of material or fouling for admiralty brass, carbon steel, and stainless steel materials.

18.2.13.5 DIESEL GENERATOR ENGINE STARTING AIR

The purpose of the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* is to manage loss of material for parts of the diesel generator engine starting air aftercoolers exposed to 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 than can affect the pressure boundary function. The program is credited with managing loss of material for carbon steel, Monel, and stainless steel materials.

18.2.13.6 PUMP MOTOR AIR HANDLING UNITS

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

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

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).

18.2.13.7 PUMP OIL COOLERS

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

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

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).

18.2.14 ICE CONDENSER ENGINEERING INSPECTION

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

18.2.15 INACCESSIBLE NON-EQ MEDIUM-VOLTAGE CABLES AGING MANAGEMENT PROGRAM

Scope – The scope of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* includes inaccessible (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 that last 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).

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.

18.2.16 INSERVICE INSPECTION PLAN

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

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

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

18.2.16.1 MCGUIRE UNIT 1 COLD LEG ELBOW

Reduction in fracture toughness due to thermal embrittlement can be an aging effect for certain types of cast austenitic stainless steel in locations where temperatures continuously exceed 482ºF. In a May 19, 2000 letter to NEI, Christopher I. Grimes, Chief License Renewal and Standardization Branch clarified that not all cast austenitic stainless steels are subject to thermal embrittlement [Reference 18 - 10]. The piping components and reactor coolant pumps fabricated from cast austenitic stainless steel were evaluated using the acceptance criteria set forth in the above letter. For those components requiring evaluation, only the McGuire 1, 27 ½-inch ID Loop B cold leg elbow exceeds the NRC-established threshold and is susceptible to thermal embrittlement which requires aging management for license renewal.

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

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

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

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

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

18.2.16.2 SMALL BORE PIPING

Small bore piping is defined as piping less than 4-inch NPS. This piping does not receive volumetric inspection in accordance with ASME Section XI, 1989 Edition, Examination Category B-J or B-F. Cracking has been identified as an aging effect requiring programmatic management for Reactor Coolant System small bore piping for the period of extended operation. A risk-informed method to select Class 1 piping welds for inspection in lieu of the requirements of 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.

The risk-informed approach is based on WCAP 14572 Revision 1 - NP-A [Reference 18 - 11] 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 safety in the event of a piping failure. As is required by WCAP 14572 Revision 1 - NP-A, the McGuire risk-informed submittals will provide equivalent or better risk coverage for the Risk Informed Inservice Inspection scope.

The results from these two independent 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 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 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 18 - 12] 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 McGuire.

18.2.17 INSPECTION PROGRAM FOR CIVIL ENGINEERING STRUCTURES AND COMPONENTS

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

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

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

NEI 96-03, *Industry Guideline for Monitoring the Condition of Structures at Nuclear Power Plants*, has been used as guidance in the preparation of the *Inspection Program for Civil Engineering Structures and Components*.

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

18.2.18 LIQUID WASTE SYSTEM INSPECTION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

18.2.19 NON-EQ INSULATED CABLES AND CONNECTIONS AGING MANAGEMENT PROGRAM

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

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

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

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

Monitoring & Trending – Accessible non-EQ insulated cables and connections installed in the Reactor Buildings, Auxiliary Building and Turbine Building are visually inspected per the *Non-EQ Insulated Cables and Connections Aging Management Program* at least once every 10 years. EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments* [Reference 18 - 13], is used as guidance in performing the inspections.

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

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

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

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

18.2.20 PREVENTIVE MAINTENANCE ACTIVITIES

18.2.20.1 CONDENSER CIRCULATING WATER SYSTEM INTERNAL COATING INSPECTION

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

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

18.2.20.2 REFUELING WATER STORAGE TANK INTERNAL COATING INSPECTION

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

18.2.21 REACTOR VESSEL INTEGRITY PROGRAM

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

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

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

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

Monitoring & Trending – Each reactor vessel had six specimen capsules located in guide baskets welded to the outside of the neutron shield pads and were positioned directly opposite the center portion of the core. McGuire Unit 1 capsules contain reactor vessel steel specimens oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting shell plate located in the core region. McGuire Unit 2 reactor vessel specimens are oriented both parallel and normal to the major working direction of the limiting core region shell forging. Associated weld metal and weld heat affected zone metal specimens are also included in each capsule. Capsule withdrawal schedules for the McGuire Units are provided in Table 18.0-2. The limiting weld material is not contained in a McGuire Unit 1 surveillance capsule, but is contained in a sister plant surveillance capsule and integrated into the McGuire Unit 1 surveillance program.

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

Fluence Received by the Specimens – Dosimeters such as Ni, Cu, Fe, Co-Al, shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are contained in the capsules. The dosimeters permit evaluation of the flux seen by the specimens. In addition, thermal monitors made of low melting point alloys are included to monitor the temperature of the specimens. A description of the methodology used to evaluate fluence received by the specimens using dosimetry measurements and fluence calculations, assuming the same neutron spectrum at the specimens and the vessel inner wall, is described in McGuire UFSAR, Sections 5.4.3.7.1 and 5.4.3.7.2 [Reference 18 - 14]. 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 18 - 15]. 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.

Table 18.0-2

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

18.2.22 REACTOR VESSEL INTERNALS INSPECTION

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

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

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

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

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

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

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

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

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

For the items comprised of plates, forgings, and welds, critical crack size will be determined by analysis 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.

18.2.23 SELECTIVE LEACHING INSPECTION

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

- Conventional Wastewater Treatment
- Diesel Generator Room Sump Pump
- Exterior Fire Protection
- Groundwater Drainage
- Interior Fire Protection
- Nuclear Service 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 parameter inspected by the *Selective Leaching Inspection* is the hardness of the wetted surface of cast iron pump casings and brass valve bodies. Selective leaching (a form of galvanic corrosion) is the dissolution of one metal in an alloy at the metal surface which leaves a weakened network of corrosion products that is revealed by a Brinnell Hardness check or equivalent as reduction in material hardness.

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

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

The *Selective Leaching Inspection* will also perform a Brinnell Hardness Test or equivalent test on a sample of brass valves at McGuire in the Interior Fire Protection System. Valves selected for inspection should be continuously exposed to stagnant or low flow raw water environments. If no parameters are known that would distinguish the susceptible locations at McGuire, a select set of susceptible locations will be examined based on accessibility, operational, and radiological concerns. The results of this inspection will be applied to the brass components exposed to raw water environments in the systems listed above.

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

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

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

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

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

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

18.2.24 SERVICE WATER PIPING CORROSION PROGRAM

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

- Containment Ventilation Cooling Water
- 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

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 microbiologically-influenced corrosion (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 18 - 16] in order to assure structural integrity. Several factors are used to determine structural integrity at an inspection location. These factors include consideration of actual as-found wall thickness, calculated rate of material loss, use of the piping stress analyses to determine a minimum required thickness and projected time to reach the minimum wall thickness which, in turn, will establish the re-inspection interval or component replacement schedule.

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

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

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

18.2.25 SUMP PUMP SYSTEMS INSPECTION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

18.2.26 TREATED WATER SYSTEMS STAINLESS STEEL INSPECTION

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

- Nuclear Solid Waste Disposal

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

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

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

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

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

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

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

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

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

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

18.2.27 UNDERWATER INSPECTION OF NUCLEAR SERVICE WATER STRUCTURES

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

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

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

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

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

Monitoring & Trending – The *Underwater Inspection of Nuclear Service Water Structures* detects aging effects through visual examination. The inspection is performed every five years at McGuire. No actions are taken as part of this program to trend inspection or test results.

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.

18.2.28 WASTE GAS SYSTEM INSPECTION

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

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

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

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

Monitoring & Trending – The *Waste Gas System Inspection* will use a volumetric technique to inspect 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 carbon steel components exposed to unmonitored treated water environments at McGuire, inspections will be performed on the lower portions of decay tanks and associated drain lines where condensate is likely to accumulate. One of eight possible locations at McGuire will be examined. If no parameters are known that would distinguish the susceptible locations at McGuire, one of the eight available at McGuire will be examined based on accessibility and radiological concerns. The results of this

inspection will be applied to the remainder of the Waste Gas System carbon steel components within the scope of license renewal exposed to unmonitored treated water environment.

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

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

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

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

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

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

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

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

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

18.3 REFERENCES FOR CHAPTER 18

18 - 1. M. S. Tuckman (Duke) letter dated June 13, 2001, to Document Control Desk (NRC), *Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2*, Docket Nos. 50-369, 50-370, 50-413, and 50-414.

18 - 2. SER (later)

18 - 3. M. S. Tuckman (Duke) letter dated July 30, 1991, *NRC Bulletin 88-09, Thimble Tube Thinning in Westinghouse Reactors,* McGuire Nuclear Station, Docket Nos. 50-369 and 50-370; Catawba Nuclear Station, Docket Nos. 50-413 and 50-414.

18 - 4. WCAP-12866, *Bottom Mounted Instrumentation Flux Thimble Wear*, January 1991.

18 - 5. 10 CFR Part 50, §50.55a, *Codes and Standards.*

18 - 6. W. T. Russell (NRC) letter dated November 19,1993 to William Rasin, (NUMARC), *Safety Evaluation for Potential Reactor Vessel Head Adapter Tube Cracking*.

18 - 7. EPRI NSAC-202L-R1, *Recommendations for an Effective Flow Accelerated Corrosion Program*, Revision 2, April 1999.

18 - 8. Nuclear System Directive 104, *Housekeeping, Materiel Condition and Foreign Material Exclusion*, Revision 19.

18 - 9. Nuclear System Directive 413, *Fluid Leak Management Program*, Revision 0.

18 - 10. C. I. Grimes (NRC) letter dated May 19, 2000 to D. J. Walters (NEI), *License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,"* Project No. 690.

18 - 11. WCAP 14572 Revision 1, NP-A, *Westinghouse Owners Group Application of Risk-Based Methods to Piping Inservice Inspection Topical Report*.

18 - 12. *Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One*, Unit 1, April 2001.

18 - 13. *Guideline for the Management of Adverse Localized Equipment Environments*, EPRI, Palo Alto, CA: 1999. EPRI TR-109619.

18 - 14. *McGuire Nuclear Station Updated Final Safety Analysis Report*, as revised.

18 - 15. WCAP-14040, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*, June 1994.

18 - 16. ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components, Subsection ND Class 3 Components, 1971 edition.