Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2

Technical Information

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1.0 ADMINISTRATIVE INFORMATION

In compliance with 10 CFR §54.19(a) and (b), the following general information (derived from 10 CFR §50.33(a) through (e), (h), and (i)) is provided:

1.1 NAME OF APPLICANT

For McGuire Nuclear Station, Units 1 and 2, Duke Energy Corporation is the applicant.

For Catawba Unit 1, Duke Energy Corporation is the applicant and is authorized to act as agent for the North Carolina Electric Membership Corporation and the Saluda River Electric Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility (NPF-35, License Condition 1.E).

For Catawba Unit 2, Duke Energy Corporation is the applicant and is authorized to act as agent for the North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility (NPF-52, License Condition 1.E).

1.2 ADDRESS OF APPLICANT

Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28202-1904

1.3 DESCRIPTION OF BUSINESS OF APPLICANT

Duke Energy Corporation (collectively with its subsidiaries, "Duke Energy") is an integrated energy and energy services provider with the ability to offer physical delivery and management of both electricity and natural gas throughout the United States and abroad. Duke Energy provides these and other services through seven business segments.

Franchised Electric generates, transmits, distributes and sells electric energy in central and western North Carolina and the western portion of South Carolina. Electric operations are conducted primarily through Duke Power and Nantahala Power and Light. These electric operations are subject to the rules and regulations of the Federal Energy Regulatory Commission, the North Carolina Utilities Commission and the Public Service Commission of South Carolina.

Natural Gas Transmission provides interstate transportation and storage of natural gas for customers primarily in the Mid-Atlantic, New England and southeastern states. Gas operations are conducted primarily through Duke Energy Gas Transmission Corporation. The interstate natural gas transmission and storage operations are subject to the rules and regulations of the FERC.

Field Services gathers, processes, transports, markets and stores natural gas and produces, transports, markets and stores natural gas liquids. Gas operations are conducted primarily through Duke Energy Field Services, LLC, a limited liability company that is approximately 30% owned by Phillips Petroleum. Field Services operates gathering systems in western Canada and 11 contiguous states that serve major natural gas-producing regions in the Rocky Mountain, Permian Basin, Mid-Continent, East Texas-Austin Chalk-North Louisiana, as well as onshore and offshore Gulf Coast areas.

North American Wholesale Energy's (NAWE's) activities include asset development, operation and management, primarily through Duke Energy North America, LLC, and commodity sales and services related to natural gas and power, primarily through Duke Energy Trading and Marketing, LLC (DETM). DETM is a limited liability company that is approximately 40% owned by Exxon Mobil Corporation. NAWE also includes Duke Energy Merchants (DEM), which develops new business lines in the evolving energy commodity markets. NAWE conducts its business throughout the U.S. and Canada.

International Energy conducts its operations through Duke Energy International, LLC. International Energy's activities include asset development, operation and management of natural gas and power facilities and energy trading and marketing of natural gas and electric power. This activity is targeted in the Latin American, Asia-Pacific and European regions.

Energy Services is a combination of businesses that provide engineering, consulting, construction and integrated energy solutions worldwide, primarily through Duke Engineering & Services, Inc., Duke/Fluor Daniel (D/FD) and DukeSolutions, Inc. D/FD is a 50/50 partnership between Duke Energy and Fluor Enterprises, Inc.

Duke Ventures is comprised of other diverse businesses, primarily operating through Crescent Resources, Inc. (Crescent), DukeNet Communications, LLC (DukeNet) and Duke Capital Partners (DCP). Crescent develops high-quality commercial, residential and multi-family real estate projects and manages land holdings primarily in the southeastern U.S. DukeNet provides fiber optic networks for industrial, commercial and residential customers. DCP, a newly formed, wholly owned merchant finance company, provides financing, investment banking and asset management services to wholesale and commercial energy markets.

1.4 LEGAL STATUS AND ORGANIZATION

Duke Energy Corporation is a public utility incorporated under the laws of the State of North Carolina. Duke's principal office is located in Charlotte, North Carolina at the address stated in Section 1.2 above. The principal location where Duke does business is North Carolina.

Duke Energy Corporation is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

Directors:

The names and business addresses of Duke Energy Corporation's directors and principal officers, all of whom are citizens of the United States, are as follows:

G. Alex Bernhardt, Sr.
Bernhardt Furniture Company
P.O. Box 740
Lenoir, North Carolina 28645

Robert J. Brown
B & C Associates, Inc.
P.O. Box 2636
High Point, North Carolina 27261

William A. Coley
Duke Energy Corporation
P.O. Box 1006, EC 3XD
Charlotte, North Carolina 28201-1006

William T. Esrey
Sprint Corporation
P.O. Box 11315
Kansas City, Missouri 64112

Ann Maynard Gray
1262 Rockrimmon
Stamford, Connecticut 06903

Dennis R. Hendrix
Duke Energy Corporation
P.O. Box 1642
Houston, Texas 77251-1642

Harold S. Hook
American General Corporation/Wortham Tower
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Houston, Texas 77019-2125

George Dean Johnson, Jr.
Extended Stay America
961 East Main Street
Spartanburg, South Carolina 29302

Dr. Max Lennon, President
Mars Hill College, Blackwell Hall
124 Cascade Street
Mars Hill, North Carolina 28754

Leo E. Linbeck, Jr.
Linbeck Corporation
P.O. Box 22500
Houston, Texas 77227

James G. Martin
Carolinas Healthcare System
P.O. Box 32861
Charlotte, North Carolina 28232

Richard B. Priory
Duke Energy Corporation
P.O. Box 1006, EC3XB
Charlotte, North Carolina 28201-1006

Principal Officers:

Richard W. Blackburn, Executive Vice President, General Counsel and Secretary
Duke Energy Corporation
P.O. Box 1006
Charlotte, North Carolina 28201-1006

Robert P. Brace, Executive Vice President and Chief Financial Officer
Duke Energy Corporation
P.O. Box 1006
Charlotte, North Carolina 28201-1006

William A. Coley,
Group President, Duke Power
Duke Energy Corporation
P.O. Box 1006
Charlotte, North Carolina 28201-1006

Fred J. Fowler, Group President, Energy Transmission
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P.O. Box 1642
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David L. Hauser
Senior Vice President and Treasurer
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P. O. Box 1244
Charlotte, North Carolina 28201-1244

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P.O. Box 1244
Charlotte, North Carolina 28201-1244

Richard J. Osborne, Executive Vice President and Chief Risk Officer
Duke Energy Corporation
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Charlotte, North Carolina 28201-1006

Harvey J. Padewer,
Group President, Energy Services
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Houston, Texas 77251-1642

Ruth G. Shaw, Executive Vice President and Chief Administrative Officer
Duke Energy Corporation
P.O. Box 1006
Charlotte, North Carolina 28201-1006

Richard B. Priory, Chairman of the Board, President and Chief Executive Officer
Duke Energy Corporation
P.O. Box 1006
Charlotte, North Carolina 28201-1006

Michael S. Tuckman, Executive Vice
President, Nuclear Generation
Duke Energy Corporation
P.O. Box 1006
Charlotte, North Carolina 28201-1006

1.5 CLASS AND PERIOD OF LICENSE SOUGHT

Duke requests renewal of the NRC Section 103 operating licenses for McGuire Nuclear Station, Units 1 and 2 (license numbers NPF-9 and NPF-17, respectively). For McGuire Unit 1 (NPF-9), renewal would revise the existing license expiration date from midnight June 12, 2021, until midnight June 12, 2041.

For McGuire Unit 2 (NPF-17), renewal would revise the existing license expiration date from midnight March 3, 2023, until either midnight March 3, 2043 or midnight 40 years from the date of the issuance of the renewed operating license for Unit 2, whichever is earlier.

Duke requests renewal of the NRC Section 103 operating licenses for Catawba Nuclear Station, Units 1 and 2 (license numbers NPF-35 and NPF-52, respectively). For Catawba Unit 1 (NPF-35), renewal would revise the existing license expiration date from midnight December 6, 2024, until either midnight December 6, 2044 or midnight 40 years from the date of the issuance of the renewed operating license for Unit 1, whichever is earlier.

For Catawba Unit 2 (NPF-52), renewal would revise the existing license expiration date from midnight February 24, 2026, until either midnight February 24, 2046 or midnight 40 years from the date of the issuance of the renewed operating license for Unit 2, whichever is earlier. The use to which both the McGuire and Catawba facilities will be put during the renewal period is the continued generation of electric power.

As reflected in these proposed revisions to the license expiration dates, Duke recognizes the legal limits associated with the term of renewed operating licenses. We also note that the technical and environmental reviews performed in connection with this Application cover operation for a period of sixty years. Duke therefore requests that the NRC complete its safety and environmental reviews such that 60-years of operation are evaluated—even though the renewed licenses issued may actually provide somewhat less than an additional 20-years of operation beyond the end of the current operating licenses of one or more of the McGuire or Catawba units.

This Application includes a request for renewal of those NRC source material, special nuclear material, and byproduct material licenses that are currently subsumed or combined with the current operating licenses.

Duke does not propose to construct or alter any production or utilization facility in connection with this renewal Application.

This Application contains no Restricted Data or other defense information.

1.6 CONFORMING CHANGES TO STANDARD INDEMNITY AGREEMENT

10 CFR §54.19(b) requires that license renewal applications include "conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license."

The current indemnity agreement for McGuire Nuclear Station (B-83) states in Article VII that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the Attachment to the agreement. Item 3 of the Attachment to the indemnity agreement, as revised through Amendment No. 10, lists NPF-9 and NPF-17, the license numbers for McGuire Nuclear Station Units 1 and 2, respectively. Should the license numbers be changed upon issuance of the renewed licenses, Duke requests that conforming changes be made to Item 3 of the Attachment to Indemnity Agreement B-83, and any other sections of the indemnity agreement as appropriate.

The current indemnity agreement for Catawba Nuclear Station (B-100) states in Article VII that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the Attachment to the agreement. Item 3 of the Attachment to the indemnity agreement, as revised through Amendment No. 9, lists NPF-35 and NPF-52, the license numbers for Catawba Nuclear Station Units 1 and 2, respectively. Should the license numbers be changed upon issuance of the renewed licenses, Duke requests that conforming changes be made to Item 3 of the Attachment to Indemnity Agreement B-100, and any other sections of the indemnity agreement as appropriate.

1.7 REGULATORY AGENCIES WITH JURISDICTION

The North Carolina Utilities Commission and the Public Service Commission of South Carolina currently have jurisdiction over the rates and services provided by Duke's utility operations at McGuire and at Catawba. The addresses of these state commissions are as follows:

North Carolina Utilities Commission
4325 Mail Service Center
Raleigh, North Carolina 27699-4325

Public Service Commission of
South Carolina
P.O. Drawer 1169
Columbia, South Carolina 29211

1.8 LOCAL NEWS PUBLICATIONS

The trade and news publications which circulate in the area surrounding McGuire Nuclear Station and Catawba Nuclear Station, and which are considered appropriate to give reasonable notice of the renewal application to those municipalities, private utilities, public bodies, and cooperatives that might have a potential interest in the facility, include the following:

The Charlotte Observer
600 South Tryon Street
Charlotte, North Carolina 28202

Lake Norman Times
Huntersville Times
147 East Center Avenue
Mooresville, North Carolina 28115

Lincoln Times News
P. O. Box 40
Lincolnton, North Carolina 28093-0040

Mt. Holly News
Belmont Banner
611 Central Avenue
Belmont, North Carolina 28012

Moorseville Tribune
Davidson Gazette
147 East Center Avenue
Moorseville, North Carolina 28115

Lake Wylie Magazine
P. O. Box 5181
Lake Wylie, South Carolina 29710

The Clover Herald
The Yorkville Inquirer
P. O. Box 38
Clover, South Carolina 29710

Fort Mill Times
P. O. Box 250
Fort Mill, South Carolina 29716-0250

The Herald
P. O. Box 11701
Rock Hill, South Carolina 29731

1.9 COMMUNICATIONS

All communications to the applicant pertaining to the Application to Renew the Operating Licenses of McGuire Nuclear Station and Catawba Nuclear Station should be sent to the following:

Addressee:
Michael S. Tuckman, Executive Vice President, Nuclear Generation
Duke Energy Corporation
Mail Stop EC 07H
P.O. Box 1006
Charlotte, North Carolina 28201-1006

Copies to:

Brew Baron
Site Vice President
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, North Carolina 28078-8985

Gary Petersen
Site Vice President
Catawba Nuclear Station
4800 Concord Road
York, South Carolina 29745-9635

Robert L. Gill, Jr.
Duke Energy Corporation
Mail Stop EC 12R
P.O. Box 1006
Charlotte, North Carolina 28201-1006

Gregory D. Robison
Duke Energy Corporation
Mail Stop EC 12R
P.O. Box 1006
Charlotte, North Carolina 28201-1006

North Carolina Municipal Power Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626

North Carolina Electric Membership Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Saluda River Electric Cooperative, Inc.
P. O. Box 929
Laurens, South Carolina 29360

In addition, it is requested that copies of correspondence related to the renewal of the McGuire and Catawba Operating Licenses be sent to Duke's Office of General Counsel and Washington, DC counsel as follows:

Lisa Vaughn, Esq.
Duke Energy Corporation
P. O. Box 1244
Mail Stop PB 05E
Charlotte, North Carolina 28201-1244

J. Michael McGarry, III, Esq.
Anne W. Cottingham, Esq.
Winston & Strawn
1400 L Street, NW
Washington, DC 20005

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2.0 SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW, AND IMPLEMENTATION RESULTS

2.1 SCOPING AND SCREENING METHODOLOGY

Note: The Scoping and Screening Methodology described in Section 2.1 is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

The criteria that determine the systems, structures and components within the scope of license renewal are provided in 10 CFR Part 54, §54.4 [Reference 2.1-1]. Section 2.1.1 provides a description of the scoping methodology that has been implemented to address these criteria for both McGuire and Catawba. This methodology is consistent with the guidance provided in NEI 95-10 [Reference 2.1-2].

The criteria that determine which structures and components are subject to aging management review are provided in 10 CFR Part 54, §54.21(a)(1) [Reference 2.1-1]. Section 2.1.2 provides a description of the screening methodology that has been implemented to address these criteria for both McGuire and Catawba. This methodology is consistent with the guidance provided in NEI 95-10 [Reference 2.1-2].

2.1.1 SCOPING METHODOLOGY

As background and in contrast with mechanical and structural scoping, electrical scoping is not performed globally on all electrical systems or components. The scoping criteria are applied only to specific electrical systems and components in order to demonstrate that they are not within the scope of license renewal. Electrical systems and components for which there is no detailed scoping evaluation are within the scope of license renewal. The result of the electrical system and component scoping is a broad set of components that are included within the scope of license renewal, and only a few systems and components are excluded from further license renewal consideration. This limited scoping does not preclude performing a scoping evaluation for other electrical systems or components at any time to demonstrate that they are not within the scope of license renewal. The electrical scoping methodology, by including within scope a set of components larger than the set of components that actually meet the §54.4(a) criteria, is consistent with the requirements of §54.4 and §54.21(a)(1) and provides reasonable assurance that there has been no omission of electrical components that are within the scope of license renewal.

2.1.1.1 Safety Related Systems, Structures and Components

Systems, structures and components which are relied upon to remain functional during and following design bases events to ensure the functions specified in §54.4(a)(1), *Scope*, are within the scope of license renewal.

Provided below are the methodologies for identifying systems, structures and components that satisfy the scoping criteria in §54.4(a)(1).

§54.4 (a)(1)

(a) Plant systems, structures, and components within the scope of this part are—

(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design bases events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions:*

(i) The integrity of the reactor coolant pressure boundary,
(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition, or
(iii)The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposure comparable to those referred to in §50.34(a)(1), §50.67(b)(2), or §100.11 of this chapter, as applicable.

** 10 CFR 50.49(b)(1)(ii)*
Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (b)(1)(i) (A) through (C) of this section. [Essentially the same as functions (a)(1)(i) through (iii) above].

2.1.1.1.1 SAFETY RELATED MECHANICAL SYSTEMS

Guidance contained in Regulatory Guides 1.26 and 1.29 has been used to establish those McGuire and Catawba systems, structures and components that satisfy the scoping criteria in §54.4(a)(1).

Regulatory Guide (RG) 1.26, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants* [Reference 2.1-3] describes a method acceptable to the NRC for assigning quality classifications to safety-related components containing water, steam or radioactive material in water-cooled nuclear power plants. The classification system consists of four different quality groups (A through D), includes methods for assigning components to these quality groups, and describes specific quality standards applicable to each quality group. The applicability of the requirements in RG 1.26 is summarized in the McGuire UFSAR Section 1.7, *Division 1 Regulatory Guides*, and in the Catawba UFSAR Section 1.7, *Regulatory Guides*. The UFSARs state how the systems, structures and components meet the guidance contained in RG 1.26. Duke System Piping Classifications A, B and C and Duke QA Condition 1 are assigned to safety-related systems, structures and components. The basis for these safety classifications is consistent with the scoping criteria of Part 54. In addition, these three piping classifications correspond to NRC Quality Group A, B and C, and ANS Safety Class 1, 2 and 3, respectively.

Regulatory Guide (RG) 1.29, *Seismic Design Classification* [Reference 2.1-4] describes a method acceptable to the NRC for identifying and classifying those plant systems, structures and components, including their foundations and supports, that should be designed to withstand the effects of a safe shutdown earthquake and remain functional. These plant systems, structures and components are designated as Seismic Category I. The applicability of the requirements in RG 1.29 is summarized in the McGuire UFSAR Section 1.7, *Division 1 Regulatory Guides*, and in the Catawba UFSAR Section 1.7, *Regulatory Guides*. The UFSARs state how the systems, structures and components meet the guidance contained in RG 1.29. Quality Assurance (QA) requirements of QA Condition 1 are applied to plant systems, structures and components identified in Regulatory Positions C.1 and C.3 of RG 1.29, while the QA requirements of QA Condition 4 are applied (with noted clarifications) to those items specified in Regulatory Position C.2.

Plant mechanical systems and components are categorized by safety classification. Within a system, components or portions of systems may have different classifications. System piping classifications are shown on mechanical system flow diagrams. Categories A, B, C, E, F, G and H have been established by Duke for the classification of components. These categories are based on NRC-defined Quality Group Standards and ANS Safety Classes, and they reflect both safety-related and non safety-related classifications. Table 2.1-1 shows the relationship between Duke System Piping Classes, NRC Quality Groups and ANS Safety Classes. This information is consistent with the descriptions of the classification system given in Table 3-5 of the McGuire UFSAR and Table 3-5 of the Catawba UFSAR.

Table 2.1-1 Duke System Piping Classifications

Duke System Piping Class	Safety Related ?	NRC Quality Group	Duke QA Condition	ANS Safety Class	Code Design Criteria	Designed for Seismic Loading?
A	Yes	A	1	1	Class 1, ASME Section III	Yes - Cat. I
B	Yes	B	1	2	Class 2, ASME Section III	Yes - Cat. I
C	Yes	C	1	3	Class 3, ASME Section III	Yes - Cat. I
E	No	D	2*	NNS	ANSI B31.1.0	No
F	No	D	4	NNS	ANSI B31.1.0	Yes - Cat. II
G	No	-	-	-	ANSI B31.1.0	No
H	No	-	-	-	Duke Power Company Specification	No

* QA CONDITION 2 is applied to systems designed to normally carry a radioactive fluid; however, they are considered non-nuclear safety (NNS) systems, since a component failure would not result in a calculated potential exposure in excess of the limits established in 10 CFR 20.

Duke System Piping Classes A, B, and C are within the scope of license renewal because they satisfy the criteria of §54.4(a)(1). A detailed description of the Duke System Piping Classes A, B and C is given in Section 3.2.3 of the McGuire UFSAR and Section 3.2.2.3 of the Catawba UFSAR. Mechanical systems and components not classified as Duke Class A, B or C are nonsafety-related for both McGuire and Catawba.

The results of the review to identify McGuire and Catawba safety related mechanical systems are included with the results of the other scoping reviews. The composite list of systems within the scope of license renewal is provided in Section 2.2 of the Application.

2.1.1.1.2 SAFETY RELATED STRUCTURES

All structures at both McGuire Nuclear Station and Catawba Nuclear Station are classified according to their design function and the degree of structural integrity required to ensure the health and safety of the public. Appendix A to 10 CFR Part 100, *Seismic and Geologic Siting Criteria for Nuclear Power Plants* [Reference 2.1-5] requires that all nuclear power plants be designed so that, if a Safe Shutdown Earthquake occurs, certain structures, systems, and components remain functional. These structures, systems and components are those necessary to ensure:

(1) the integrity of the reactor coolant pressure boundary,

(2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures in 10 CFR Part 100.

These three functions meet the intent of those specified in the scoping criteria in 10 CFR 54.4(a)(1). The specific structures that are required to ensure these functions are satisfactorily implemented are identified in RG 1.29, as Seismic Category I. All safety-related structures are designated as Seismic Category I and are within the scope of license renewal. The classification of each structure has been previously determined and documented in the McGuire Nuclear Station and the Catawba Nuclear Station UFSARs. Category I structures are identified through a review of the plant UFSAR.

The results of the review to identify McGuire and Catawba safety-related structures are included with the results of the other scoping reviews. The composite list of structures within the scope of license renewal is provided in Section 2.2 of the Application.

2.1.1.1.3 SAFETY-RELATED ELECTRICAL SYSTEMS AND COMPONENTS

Electrical scoping was performed at the system and component level. Specific electrical systems and components were reviewed against the safety-related scoping criteria of §54.4(a)(1). Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet safety-related scoping criteria of §54.4(a)(1). Refer to Section 2.1.2.3.1 for passive electrical commodities. No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

2.1.1.2 Nonsafety-Related Systems, Structures and Components

All nonsafety-related systems, structures and components, whose failure could prevent satisfactory accomplishment of any of the functions identified in §54.4 (a)(1)(i), (ii), and (iii) are within the scope of license renewal.

§54.4 (a)(2)

> *All nonsafety-related systems, structures and components, whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (i), (ii), and (iii) of this section.*

Two types of systems, structures, and components must be considered for inclusion within the scope of license renewal per §54.4(a)(2):

1. Nonsafety-related systems and structures, and nonsafety-related portions of safety-related systems and structures, whose physical failure could damage equipment that is performing a safety function, and prevent it from performing that function. This includes:

   - Failure of structural supports such that piping and/or components might fall onto equipment that is performing a safety function, and prevent it from performing that function. In this case, only the structural supports are required to remain intact in order to ensure the safety function is fulfilled.

   - Fluid leakage from piping and/or components onto nearby equipment that is performing a safety function, and which could prevent it from performing that function. In this case, both the pressure boundary and structural integrity of the piping and/or components are of concern.

2. Nonsafety-related systems, structures and components whose failure could prevent satisfactory accomplishment of any of the functions identified in §54(a)(1)(i),(ii) and (iii).

Provided below are the methodologies for identifying mechanical systems, structures, and electrical systems and components that satisfy the scoping criterion in §54.4(a)(2).

2.1.1.2.1 NONSAFETY-RELATED MECHANICAL SYSTEMS

Duke System Piping Class F, also designated QA Condition 4, is the nonsafety-related piping assigned to those systems and components whose pressure boundary loss may adversely affect essential systems or equipment. These systems and components have therefore been designed for seismic loading.

Duke Class F applies to those portions of systems and components whose failure:

- could result in flooding of nuclear safety-related equipment, or

- could jeopardize nuclear safety-related piping or equipment during a safe shutdown earthquake.

Components and materials classified as Duke Class F also meet the code and standard requirements of NRC Quality Group D. Mechanical systems containing Class F piping and/or components are within the scope of license renewal.

QA Condition 4 has also been applied to nonsafety-related portions of piping and HVAC duct systems that must be seismically-qualified to preclude adversely impacting the function of safety-related equipment during a safe shutdown earthquake. In this situation, only the structural supports are required to remain intact in order to ensure the safety function is fulfilled.

In addition, a review of design and licensing basis documentation indicates that some additional nonsafety-related mechanical systems, which would be designated as Duke Class E, G, or H, may be relied upon to remain functional during and following design basis events. These nonsafety-related systems are also included in the scope of license renewal.

2.1.1.2.2 NONSAFETY-RELATED STRUCTURES

Structures whose continued function is not required, but whose failure could impair the function of safety-related SSCs, or could injure control room occupants, are designated as Duke Power Seismic Category II in accordance with Regulatory Guide 1.29 Position C.2. They are nonsafety-related, but are designed to prevent detrimental effects to safety-related SSCs or injury to control room occupants. Category II structures meet the intent of §54.4(a)(2) and are within the scope of license renewal.

Structures at McGuire and Catawba that are not identified as either Category I or II are classified as Category III structures. Category III structures are those whose functions are not related to nuclear safety and whose collapse under earthquake loading will not impair the integrity of Seismic Category I or II items. Category III structures are not within the scope of license renewal unless they are determined to meet the criteria of §54.4(a)(3).
The classification of each structure has been previously determined and documented in the McGuire UFSAR and the Catawba UFSAR. Category II structures are identified through a review of the plant UFSAR.

2.1.1.2.3 NONSAFETY RELATED ELECTRICAL SYSTEMS AND COMPONENTS

Electrical scoping was performed at the system and component level. Specific electrical systems and components were reviewed against the nonsafety-related scoping criterion of §54.4(a)(2). Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet nonsafety-related scoping criterion of §54.4(a)(2). Refer to Section 2.1.2.3.1 for passive electrical commodities. No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

2.1.1.3 Regulated Events

In addition to those systems, structures and components relied upon to mitigate design basis events (§54.4(a)(1)), or whose failure could prevent mitigation of design basis events (§54.4(a)(2)), the systems and plant structures previously committed to support certain specific NRC regulations must be identified for license renewal. The methodology for identifying the systems and structures that are required to demonstrate compliance with each of these regulations is provided in this section.

Part 54 defines the scoping requirements for these specific regulations as follows:

10 CFR 54.4(a)(3)

> *(a) Plant systems, structures, and components within the scope of this part are--*
>
> *[...]*
>
> *(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).*

The systems, structures and components required to demonstrate compliance with the specific regulations identified in §54.4(a)(3) were determined through an extensive review of licensing correspondence files, safety evaluation reports, the UFSAR, and other appropriate design documents for each station. The following sections include discussions of the regulated events and the results of the review.

2.1.1.3.1 FIRE PROTECTION

Systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with §50.48, *Fire protection*, are within the scope of license renewal.

§50.48 (b)

> *Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues licensed to operate prior to January 1, 1979. Except for the requirements of Sections III.G, III.J, and III.O, the provision of Appendix R to this part shall not be applicable to nuclear power plants licensed to operate prior to January 1, 1979, to the extent that fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position BTP APCSB 9.5-1 reflected in staff fire protection safety evaluation reports issued prior to the effective date of this rule, or to the extent that fire protection features were accepted by the staff in comprehensive fire protection safety evaluation reports issued before Appendix A to Branch Technical Position BTP APCSB 9.5-1 was published in August 1976.*

Note: The licensing basis with regard to fire protection differs at McGuire and Catawba. The following paragraphs discuss the fire protection systems at each station.

McGuire is licensed to 10 CFR §50.48 (b) as specifically stated in Safety Evaluation Reports and Facility Operating License. The McGuire fire protection system is designed to provide automatic and manual means to control and extinguish fires that may occur within building, yard, and transformer areas. The fire protection program at McGuire is based upon an evaluation of potential fire hazards throughout the Auxiliary and Reactor Buildings and areas adjacent to these facilities. This evaluation demonstrates that the plant will maintain the ability to perform safe shutdown functions and minimize radioactive releases to the environment in the event of a fire. The NRC Safety Evaluation Report for McGuire, NUREG-0422 [Reference 2.1-6], provides the staff evaluation of the McGuire response to BTP APCSB 9.5-1 and modifications to comply with BTP APCSB 9.5-1. The following license conditions apply for McGuire Nuclear Station: 2.C.(3) for Unit 1 and 2.C.(7) for Unit 2.

Catawba is licensed to 10 CFR §50.48 (b) as specifically stated in Safety Evaluation Reports and Facility Operating License. The Catawba fire protection system is designed to provide automatic and manual means to control and extinguish fires that may occur within building, yard and transformer areas. The fire protection program at Catawba Nuclear Station is based upon an evaluation of potential fire hazards throughout the Auxiliary, Diesel Generator and Reactor Buildings, and Nuclear Service Water Pump Structure, and those portions of the Turbine and Service Buildings adjacent to these facilities. This evaluation demonstrates that the plant will maintain the ability to perform safe shutdown functions and minimize radioactive releases to the environment in the event of a fire. The NRC Safety Evaluation Report for Catawba, NUREG-0954 [Reference 2.1-7], provides the staff evaluation of the Catawba response to the applicable fire protection requirements. The following license conditions apply for Catawba Nuclear Station: 2.C.(8) for Unit 1 and 2.C.(6) for Unit 2.

As part of the response to Appendix A to BTP APCSB 9.5-1 [Reference 2.1-8], Duke committed to install a dedicated standby shutdown system (SSS) at each station that uses some existing plant safety-related systems, as well as certain equipment that would be used only in the event of a fire or plant security emergency. The purpose of the SSS during such events is to bring the plant to a hot standby condition and maintain the hot standby mode for up to 3 days without recourse to damage control measures. The capability to achieve and maintain hot standby conditions utilizing the SSS is assured in most fire areas by virtue of location of SSS-required equipment and cabling outside the fire area and breaker coordination on non-required SSS loads. In cases where the SSS is unavailable because of the fire, normal shutdown capability would be available.

Each station uses a quality condition designation, Duke QA Condition 3, that applies uniquely to fire protection systems, structures, components, and services. Systems designated as QA Condition 3 are those systems that promptly detect, control and extinguish fires to limit their damage and provide protection for systems, structures, components, and services so that a fire will not prevent the safe shutdown of the plant.

The results of the review to Identify McGuire and Catawba mechanical systems and structures that are relied upon to demonstrate compliance with §50.48 are included with the results of the other scoping reviews. The composite list of mechanical systems and structures within the scope of license renewal is provided in Section 2.2 of this Application.

Electrical scoping was performed at the system and component level. Specific electrical systems and components were reviewed against the fire protection regulated event scoping criterion of §54.4(a)(3). Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet the fire protection regulated event scoping criterion of §54.4(a)(3). No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

2.1.1.3.2 ENVIRONMENTAL QUALIFICATION

Systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with §50.49, *Environmental qualification of electric equipment important to safety for nuclear power plants*, are within the scope of license renewal.

§50.49

(a) Each holder of or an applicant for a license for a nuclear power plant, other than a nuclear power plant for which the certification required under §50.82(a)(1) have been submitted, shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.

(b) Electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment…

(2) Non-safety-related electric equipment…

(3) Certain post-accident monitoring equipment…

The Environmental Qualification master list identifying environmentally qualified electrical equipment is used to scope electrical components for this criterion.

The results of the review to identify McGuire and Catawba mechanical systems, containing mechanical components associated with electrical components that are relied upon to demonstrate compliance with §50.49, are included with the results of the other scoping reviews. The composite list of mechanical systems within the scope of license renewal is provided in Section 2.2. of this Application. No structures are relied upon to demonstrate compliance with §50.49.

Electrical scoping was performed at the system and component level. Specific electrical systems and components were reviewed against the environmental qualification regulated event scoping criterion of §54.4(a)(3). Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet the environmental qualification regulated event scoping criterion of §54.4(a)(3). No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

2.1.1.3.3 PRESSURIZED THERMAL SHOCK

Systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with §50.61, *Fracture toughness requirements for protection against pressurized thermal shock events,* are within the scope of license renewal.

§50.61(a)(2)

> *Pressurized Thermal Shock Event means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.*

Pressurized thermal shock has also been identified as a time-limited aging analysis (TLAA) for both McGuire Nuclear Station and Catawba Nuclear Station. The discussion of TLAA is provided in Chapter 4 of the Application. As discussed in Section 4.2.2 of the Application, the screening criteria contained in §50.61 will not be exceeded during the 20-year period of extended operation for any one of the McGuire or Catawba reactor vessels.

The reactor vessels are the only components required to demonstrate compliance with 10 CFR 50.61.

Pressurized thermal shock is a phenomenon limited to the Reactor Coolant System and the reactor vessel and does not directly involve any structures or structural components. Therefore, no structures are required to demonstrate compliance with §50.61. Likewise, no electrical systems or components are required to demonstrate compliance with §50.61.

The results of the review to identify McGuire and Catawba mechanical systems that are relied upon to demonstrate compliance with §50.61 are included with the results of the other scoping reviews. The composite list of systems within the scope of license renewal is provided in Section 2.2 of this Application.

2.1.1.3.4 ANTICIPATED TRANSIENTS WITHOUT SCRAM

Systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with §50.62, *Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water cooled nuclear power plants*, are within the scope of license renewal.

§50.62 (b)

> *Anticipated Transient Without Scram (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Appendix A of this part.*

§50.62 (c)(1)

> *Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.*

To satisfy the regulatory requirements in §50.62, ATWS Mitigation System Actuation Circuitry (AMSAC) systems were installed at McGuire Nuclear Station and Catawba Nuclear Station. The design of these AMSAC systems is consistent with the generic design proposed by the Westinghouse Owners Group and is based on conditions that indicate a loss of main feedwater event, which if accompanied by a failure of the Reactor Protection System to SCRAM, the reactor leads to overpressurization of the Reactor Coolant System.

AMSAC actuation will occur if both main feedwater pumps trip or when main feedwater flow to the steam generators (3 out of 4 logic) is blocked due to valve closure in the feedwater lines. When an actuation occurs, the AMSAC circuitry will perform the following:

- Trip the main turbine,
- Start both motor-driven, auxiliary feedwater pumps, and
- Close the steam generator blowdown and sampling valves.

Electrical scoping was performed at the system and component level. Specific electrical systems and components were reviewed against the anticipated transients without SCRAM regulated event scoping criterion of §54.4(a)(3). Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet the anticipated transients without SCRAM regulated event scoping criterion of §54.4(a)(3). No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

The results of the review to identify McGuire and Catawba mechanical system components that either provide input to or are actuated by the AMSAC system are included with the results of the other scoping reviews. The composite list of systems within the scope of license renewal is provided in Section 2.2 of the Application. No structures are relied upon to demonstrate compliance with §50.62.

2.1.1.3.5 STATION BLACKOUT

Systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with §50.63, *Loss of all alternating current power*, are within the scope of license renewal.

§50.63 (a)(1)

> *Each light-water cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in §50.2. The specified station blackout duration shall be based on the following factors:*
>
> *(i) The redundancy of the onsite emergency ac power sources;*
>
> *(ii) The reliability of the onsite emergency ac power sources;*
>
> *(iii) The expected frequency of loss of offsite power; and*
>
> *(iv) The probable time needed to restore offsite power.*

McGuire Nuclear Station is a four (4) hour coping duration plant. The station utilizes an alternate AC power source as the method of coping with the station blackout. The alternate AC source is the Standby Shutdown Facility diesel. The equipment that would be used to achieve a safe shutdown condition is the equipment associated with the Standby Shutdown System.

Likewise, Catawba Nuclear Station is a four (4) hour coping duration plant. The station utilizes an alternate AC power source as the method of coping with the station blackout. The alternate AC source is the Standby Shutdown Facility diesel. The equipment that would be used to achieve a safe shutdown condition is the equipment associated with the Standby Shutdown System.

For both McGuire and Catawba, certain structures are relied upon to demonstrate compliance with §50.63. Certain areas containing safe shutdown equipment may heat up due to a loss of ventilation and cooling. Heatup calculations that were performed as a part of the station blackout review take credit for walls and ceilings of structures housing this safe shutdown equipment as a heat sink. The structures that contain these walls and ceilings are within the scope of license renewal.

Electrical scoping was performed at the system and component level. Specific electrical systems and components were reviewed against the station blackout regulated event scoping criterion of §54.4(a)(3). Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet the station blackout regulated event scoping criterion of §54.4(a)(3). No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

The results of the review to identify McGuire and Catawba mechanical systems and structures that are relied upon to demonstrate compliance with §50.63 are included with the results of the other scoping reviews. The composite list of systems and structures within the scope of license renewal is provided in Section 2.2 of the Application.

2.1.2 Screening Methodology

The identification of those structures and components that are subject to an aging management review is defined as screening. The criteria that determine the structures and components subject to aging management review for license renewal are provided in §54.21(a)(1).

> **§54.21(a)**
> *(1) For those systems, structures and, components within the scope of this part, as delineated in §54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components –*
>
> *(i) That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties…; and*
>
> *(ii) That are not subject to replacement based on a qualified life or specified time period.*

NEI 95-10 [Reference 2.1-2] contains guidance that has been used for determining structures and components subject to an aging management review in this Application. Areas of NEI 95-10 that have been specifically used in this Application are Section 4.1.2, "Determining Structures and Components Subject to Aging Management Review and Their Intended Functions," and Appendix B, "Typical Structure, Component, and Commodity Groupings and Active/Passive Determinations for the Integrated Plant Assessment." The following sections provide additional descriptions of the screening methodologies used to identify the mechanical, structural and electrical components that are subject to an aging management reviews for McGuire Nuclear Station and Catawba Nuclear Station.

2.1.2.1 Screening Methodology for Mechanical Components

Section 2.1.2.1 provides the methodology to determine the mechanical components subject to an aging management review in accordance with the requirements of §54.21(a)(1). This component screening methodology for McGuire and Catawba involves the following steps:

1. Establishment of the license renewal evaluation boundaries.

2. Identification of mechanical components subject to aging management review.

3. Identification of the intended function(s) of each mechanical component.

These activities are discussed further in the following sections.

2.1.2.1.1 ESTABLISHMENT OF LICENSE RENEWAL SYSTEM EVALUATION BOUNDARIES

The result of implementing the methodology described in this section is a set of highlighted system flow diagrams for each plant establishing the license renewal evaluation boundaries corresponding to the scoping criteria of 10 CFR 54.4(a). The mechanical components found within the highlighted portions of these flow diagrams comprise the complete set of mechanical components within the scope of license renewal.

The evaluation boundaries are defined by marking system flow diagrams in accordance with the following guidelines:

Safety Related Boundaries

By definition, Duke Piping Class A, B, and C portions of a system are intended to perform safety-related functions to ensure that the criteria of §54.4(a)(1)(i)-(iii) are met. All flowpaths designated as Duke Piping Class A, B, or C, as identified by the Design Parameters table on the system flow diagrams, are highlighted.

Duke Piping Class A, B, and C do not apply to ventilation systems. Safety-related ventilation systems are identified as QA Condition 1, as indicated on system flow diagrams. For McGuire, the flow diagrams contain triangular flags that contain the number “1” to designate flowpaths or components as QA Condition 1. For Catawba, a review of system design basis specifications and applicable portions of the UFSAR is used to identify the safety-related flowpaths or components of each ventilation system. The QA Condition 1 flowpaths or components of ventilation systems are highlighted.

Non-Safety Related Boundaries

Duke System Piping Class F is the non safety-related piping classification for piping and components whose pressure boundary loss may adversely affect safety-related systems and components due to physical interactions. All Class F piping and components meet the criteria of §54.4(a)(2). All flowpaths designated as Duke Piping Class F, as identified by the Design Parameters table on the system flow diagrams, are highlighted.

A review of design and licensing documents indicates that some nonsafety-related components’ functional failure may prevent a safety-related function from being performed. These flowpaths are not designated as Duke Piping Class A, B, C, or F. The flowpaths associated with these functions are highlighted.

Regulated Event Boundaries

Fire Protection
The following three aspects of plant design and documentation define the flowpaths and components that establish compliance with §50.48 and thus the evaluation boundaries to meet the fire protection criteria of §54.4(a)(3):

(1) Quality designation QA Condition 3 applies uniquely to fire protection systems, components, and services. Flow diagrams indicate with a note those flowpaths required to comply the with QA Condition 3 program. Those flowpaths and components designated on flow diagrams as QA Condition 3 are highlighted.
(2) As part of the response to Branch Technical Position APCSB 9.5-1, McGuire and Catawba provided a dedicated standby shutdown system to provide an alternate and independent means of achieving and maintaining a hot standby condition for one or both units following a postulated fire, sabotage, or station blackout event. For McGuire, the flow diagrams contain triangular flags that contain the letter "S" to designate flowpaths as part of the standby shutdown system. For Catawba, a review of applicable design and licensing documents related to the fire protection program provide information regarding flowpaths designated as part of the standby shutdown system. Those flowpaths designated as part of the standby shutdown system are highlighted.
(3) Components required to reach cold shutdown are required to establish compliance with §50.48 and thus the evaluation boundaries for the fire protection criteria of §54.4(a)(3). A review of applicable design and licensing documents provide information regarding functions required for cold shutdown. The flowpaths associated with these functions are highlighted.

Environmental Qualification
Environmental Qualification is only applicable at the component level. The Environmental Qualification equipment list established for 10 CFR 50.49(d) meets the environmental qualification criteria of §54.4(a)(3). Using this established equipment list, Environmental Qualification-related electrical components associated with mechanical components are included in the evaluation boundaries and are highlighted.

Pressurized Thermal Shock
The reactor vessels, components of the Reactor Coolant System, are the only components required to demonstrate compliance with 10 CFR 50.61, as described in Section 2.1.1.3.3 of this Application. The evaluation boundaries for pressurized thermal shock are bounded by the established safety-related boundaries.

Anticipated Transients without Scram

The Anticipated Transients without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) electrical system was installed at McGuire and Catawba to meet the requirements of 10 CFR 50.62. A review of applicable design and licensing documents provide information regarding mechanical components that either provide input to or are actuated by the AMSAC system. These flowpaths or components are highlighted.

Station Blackout

McGuire and Catawba both take credit for the Standby Shutdown Facility diesel generator as an alternate AC source for coping with a station blackout event, and for the standby shutdown system-related equipment to achieve and maintain a hot standby condition for one or both units following the station blackout. As a result, the regulated event boundaries for station blackout are coincident with or bounded by those established for the safe shutdown portion of Fire Protection.

2.1.2.1.2 IDENTIFICATION OF COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW

A menu that lists all passive, long-lived mechanical components and component groupings that are subject to an aging management review was developed for each plant. This menu was developed using plant system flow diagrams, equipment databases, and the guidance contained in NEI 95-10, Appendix B. The components within the marked areas of the flow diagrams are compared to the menu to determine the components that are subject to aging management review.

All instruments are exempt from an aging management review because they perform their function(s) with moving parts and/or a change of configuration or properties except for those mechanical indicating devices and electrical components that form an integral part of the pressure boundary. The pressure boundary of mechanical indicating devices and electrical components such as level glasses, flow glasses, in-line flow switches, elements, resistance temperature detectors (RTDs), sensors, thermocouples and transducers are subject to aging management review.

In addition, the definition of cooling fans includes ventilation fans, exhaust fans, purge fans, and blowers, and the definition of ventilation dampers is expanded to include ventilation louvers because the functions of these components are performed with moving parts.

Filter mediums such as paper filters, charcoal filters, and resins are within the scope of license renewal, but are replaced on condition and are not subject to aging management review. Periodic testing and inspection programs are in place to monitor filter performance, degradation of which may be indicated by an increase in differential pressure or a change in absorption efficiency. The filter mediums are replaced as conditions warrant; therefore, an aging management review is not required.

Portable equipment is within the scope of license renewal but is not subject to aging management review because it is replaced on condition. Such equipment is routinely inspected for degradation. For example, fire extinguishers, self-contained breathing air packs, fire hoses and portable ductwork, credited for compliance with the Fire Protection rule, are inspected in accordance with National Fire Protection Association (NFPA) standards. These standards require replacement of portable equipment based on their condition or performance during testing and inspection. These portable components are not long-lived and are subject to replacement per NFPA standards, therefore an aging management review is not required.

2.1.2.1.3 IDENTIFICATION OF THE INTENDED FUNCTION(S) OF EACH SUBJECT COMPONENT

As described in NEI 95-10, the intended functions define the plant process, condition, or action that must be accomplished in order to perform or support a safety function for responding to a design basis event or to perform or support a specific requirement of one of the five regulated events in §54.4(a)(3). At a system level, the intended functions may be thought of as the functions of the system that are the bases for including this system within the scope of license renewal as specified in §54.4(a)(1)-(3). For the purposes of component screening and aging management review, a component intended function is defined as a function performed by a component in support of an intended function. The component intended functions of a subject component include only those functions that are required to enable the system to perform its intended function(s). A single component may provide multiple intended functions.

2.1.2.1.4 FLOW DIAGRAM HIGHLIGHTING AND COMPONENT IDENTIFICATION CONVENTIONS

The following items define the specific flow diagram highlighting conventions and are provided as an aid to the reviewer:

- For vents, drains, sample and test connections, the line is marked at the outlet of the isolation valve.

- All instrument lines normally open to the process system, through and including the instrument itself, are included within the scope of license renewal. These lines, however, are not marked, except for Containment penetrations.

- When the in-scope portion of the system is continued on a flow diagram for another system, the evaluation boundary includes the system boundary valve for the interfacing system.

- When the system boundary is not a physical boundary, the evaluation boundary extends to include the first physical boundary (closeable valve(s)) in the interfacing system.

- Each heat exchanger within the marked evaluation boundaries is evaluated with the system in its component tag number.

2.1.2.1.5 RESULTS
The tables contained in Sections 3.1, 3.2, 3.3, and 3.4 of this Application list all mechanical components that are subject to aging management review along with their intended functions.

2.1.2.2 Screening Methodology for Structural Components

Section 2.1.2.2 provides the methodology to determine the structural components subject to an aging management review in accordance with the requirements of §54.21(a)(1) of the license renewal rule. This component screening methodology for McGuire and Catawba involves the following steps:

1. Generation of a list of structural component types.

2. Identification of the intended function(s) of each structural component.

3. Identification of structural components subject to aging management review.

Following the identification of structures within the scope of license renewal, the structural components within each of the structures are identified. A generic list of structural components was developed by using the lists of components provided in NUMARC Containment and Class I Structures Industry Reports and Appendix B of NEI 95-10, [References 2.1-9, 2.1-10, and 2.1-2 respectively]. Additional components were added following the review of commitments made for compliance with the following regulated events: fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram and station blackout. Finally, several McGuire specific and Catawba specific documents were reviewed to determine any other structural components not previously identified. The components for each structure were identified using this generic list.

The functions of the structures were determined from a review of information contained in the plant specific UFSARs, plant specific engineering specifications, and plant specific regulated events documentation. The functions of structural components were determined from a review of the commitments made in response to design basis events and regulated events.

The structural component function(s) may support the intended function(s) of the structure or may have a unique function that does not support the intended function of the structure. A case in point is the spent fuel storage racks that are located in the Auxiliary Building. A unique function of the spent fuel storage racks is to maintain separation of the fuel assemblies to prevent criticality, which is not considered to be an intended function of the Auxiliary Building itself.

Structural steel, anchor bolts, base plates, etc. that are required to support nonsafety-related components to prevent physical interaction with safety-related equipment are subject to aging management reviews. These components must remain in place such that they do not impact equipment that is required to perform a safety function in such a way as to prevent the equipment from performing its safety function. This interaction is commonly referred to as Category II over Category I and is addressed in Regulatory Guide 1.29, Position C.2. [Reference 2.1-4] Components that address Position C.2 are identified as Duke QA Condition 4 and are within the scope of license renewal.

Architectural and structural features such as stairs, building siding, panels, platforms, and grating that do not perform a license renewal intended function are not subject to an aging management review.

Materials such as caulking and waterstops are not identified as structures or components. However, limited situations may exist where these materials are important to maintaining the integrity of the components to which they are connected. The license renewal structure or component intended functions supported by these materials are limited to two functions. These functions are:

(1) Providing a rated fire barrier.
(2) Providing a flood barrier.

Sealants and caulking that support the fire barrier function are addressed as part of the fire barrier penetration seals. Caulking and waterstops that support the flood barrier function are addressed with the wall or floor within which the sealant/waterstop is contained. Flood seals are identified as components and are addresed with the Auxiliary Building. Seals associated with maintaining pressure boundary are limited to the divider barrier seals in the Reactor Building.

Consistent with the guidance provided in NEI 95-10, structures and structural components within the scope of license renewal are long-lived and passive; therefore they require an aging management review. The tables contained in Section 3.5 of this Application list structural components that are subject to aging management review along with their intended functions.

2.1.2.3 Screening Methodology for Electrical Components

Section 2.1.2.3 provides the methodology of applying the passive and long-lived criteria of §54.21(a)(1) to electrical components. The electrical component integrated plant assessment is a commodity review and the electrical and instrumentation and control (I&C) commodity groupings are identified in Appendix B of NEI 95-10 [Reference 2.1-2]. The electrical screening methodology involves passive screening and long-lived screening that are discussed in the following sections.

2.1.2.3.1 PASSIVE SCREENING

NEI 95-10, Appendix B identifies the passive electrical and I&C commodity groupings (components that perform an intended function without moving parts or without a change in configuration) which are shown in Table 2.1-2 below. Other electrical and I&C commodities are active.

Table 2.1-2 Identification of Passive Electrical and I&C Commodities

Passive Electrical and I&C Commodities
Electrical portions of electrical and I&C penetration assemblies
High-voltage insulators
Insulated cables and connections (power, instrumentation and control applications; connections include plug-in connectors, splices and terminal blocks)
Phase bus (e.g., isolated-phase bus, nonsegregated-phase bus, bus duct)
Switchyard bus
Transmission conductors
Uninsulated ground conductors

Electrical and I&C commodities that may have a function to maintain pressure boundary (elements, resistance temperature detectors [RTDs], sensors, thermocouples, transducers, and heaters) are electrically active and would be subject to an aging management review only for the pressure boundary function. Components performing a pressure boundary function are included in the mechanical component screening process (See Section 2.1.2.1).

2.1.2.3.2 LONG-LIVED SCREENING

Electrical components included in the McGuire and Catawba *Environmental Qualification Program* (per 10 CFR 50.49) are replaced based on a qualified life, do not meet the criterion of §54.21(a)(1)(ii) and are not subject to an aging management review.

Some insulated cables and connections are included in the McGuire and Catawba *Environmental Qualification Program*. Insulated cables and connections included in the McGuire and Catawba *Environmental Qualification Program* do not meet this long-lived screening criterion and are not subject to an aging management review.

All electrical and I&C penetration assemblies are included in the McGuire and Catawba *Environmental Qualification Program*. None of the electrical and I&C penetration assemblies meet this long-lived screening criterion and none are subject to an aging management review. The portion of the penetrations which support the essentially leak-tight barrier of Containment are subject to an aging management review and are included with Reactor Building structural components.

No other electrical components are screened out per this long-lived screening criterion. The result is that the remainder of the integrated plant assessment involves only non-EQ electrical and I&C components.

2.1.3 REFERENCES FOR SECTION 2.1

2.1-1. Code of Federal Regulations, Title 10, Part 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*, 60 FR 22461, May 8, 1995.

2.1-2. *NEI 95-10*, *Revision 2*, *Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule,* Nuclear Energy Institute, August 2000.

2.1-3. Regulatory Guide (RG) 1.26, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.*

2.1-4. Regulatory Guide (RG) 1.29, *Seismic Design Classification.*

2.1-5. Code of Federal Regulations, Title 10, Part 100, *Reactor Site Criteria.*

2.1-6. NUREG-0422, *Safety Evaluation Report Related to the Operation of the McGuire Nuclear Station, Units 1 and 2*, March 1978, as supplemented, Docket Nos. 50-369 and 50-370.

2.1-7. NUREG-0954, *Safety Evaluation Report Related to the Operation of the Catawba Nuclear Station, Units 1 and 2,* February 1983, as supplemented, Docket Nos. 50-413 and 50-414.

2.1-8. BTP APCSB 9.5-1 Appendix A, Branch Technical Position, *Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976,* August 1976.

2.1-9. *Pressurized Water Reactor Containment Structures License Renewal Industry Report*, NUMARC Report Number 90-01, Nuclear Management and Resources Council, Revision 1, September 1991.

2.1-10. *Class I Structures License Renewal Industry Report,* NUMARC Report Number 90-06, Nuclear Management and Resources Council, Revision 1, December 1991.

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2.2 PLANT LEVEL SCOPING RESULTS

2.2.1 SYSTEMS, STRUCTURES AND COMPONENTS WITHIN THE SCOPE OF LICENSE RENEWAL

As described in Section 2.1 of this Application, the criteria that determine the systems, structures and components within the scope of license renewal are provided in 10 CFR §54.4. Guidance contained in Regulatory Guides 1.26 and 1.29 has been used to establish which McGuire and Catawba mechanical systems, structures and components that satisfy the scoping criteria in §54.4(a)(1) as described in Section 2.1.1.1 of this Application. All nonsafety-related systems, structures and components, whose failure could prevent satisfactory accomplishment of any of the functions identified in §54.4 (a)(1)(i), (ii), and (iii) are within the scope of license renewal and have been identified by the methodology described in Section 2.1.1.2 of this Application. Finally, in addition to those systems, structures and components relied upon to mitigate design basis events (§54.4(a)(1)), or whose failure could prevent mitigation of design basis events (§54.4(a)(2)), the systems, structures and components previously committed to support certain specific NRC regulations (§54.4(a)(3)) are within the scope of license renewal and have been identified by the methodology described in Section 2.1.1.3 of this Application.

The McGuire systems, structures and components that are within the scope of license renewal (§54.4) are listed in Table 2.2-1. The mechanical systems listed in this table are described in Section 2.3, System Scoping and Screening Results: Mechanical. Specifically, the Reactor Coolant System is described in Section 2.3.1, engineered safety features systems are described in Section 2.3.2, auxiliary systems are described in Section 2.3.3 and steam and power conversion systems are described in Section 2.3.4. The structures listed in Table 2.2-1 and their structural components are described in Section 2.4. The electrical and I&C components that are subject to an aging management review are described in Section 2.5.

The Catawba systems, structures and components that are within the scope of license renewal (§54.4) are listed in Table 2.2-2. The mechanical systems listed in this table are described in Section 2.3, System Scoping and Screening Results: Mechanical. Specifically, the Reactor Coolant System is described in Section 2.3.1, engineered safety features systems are described in Section 2.3.2, auxiliary systems are described in Section 2.3.3 and steam and power conversion systems are described in Section 2.3.4. The structures listed in Table 2.2-2 and their structural components are described in Section 2.4. The electrical I&C components that are subject to an aging management review are described in Section 2.5.

2.2.2 Systems and Structures Not within the Scope of License Renewal

To assist the staff is its review of scoping, tables are provided that identify systems and structures that are not within the scope of license renewal. None of the systems and structures listed in these tables meet any of the criteria contained in §54.4.

The McGuire systems and structures that are not within the scope of license renewal are listed in Table 2.2-3, McGuire Systems and Structures Not within the Scope of License Renewal.

The Catawba systems and structures that are not within the scope of license renewal are listed in Table 2.2-4, Catawba Systems and Structures Not within the Scope of License Renewal.

Table 2.2-1 McGuire Systems and Structures within the Scope of License Renewal

McGuire Mechanical Systems within the Scope of License Renewal

Annulus Ventilation
Auxiliary Building Ventilation
Auxiliary Feedwater
Auxiliary Steam
Boron Recycle
Chemical & Volume Control
Component Cooling
Condenser Circulating Water
Containment Air Return Exchange & Hydrogen Skimmer
Containment Spray
Control Area Chilled Water
Control Area Ventilation
Conventional Waste Water Treatment
Diesel Building Ventilation
Diesel Generator Air Intake And Exhaust
Diesel Generator Cooling Water
Diesel Generator Crankcase Vacuum
Diesel Generator Fuel Oil
Diesel Generator Lube Oil
Diesel Generator Room Sump Pump
Diesel Generator Starting Air
Feedwater
Feedwater Pump Turbine Hydraulic Oil
Fire Protection
Fuel Handling Building Ventilation
Groundwater Drainage
Heating Water
Hydrogen Bulk Storage
Instrument Air
Liquid Waste Monitor And Disposal
Liquid Waste Recycle
Lower Containment Ventilation
Main Steam
Main Steam Supply To Auxiliary Equipment
Main Steam Vent To Atmosphere
Main Turbine Hydraulic Oil
Main Turbine Lube Oil and Purification
Nitrogen
Nuclear Sampling
Nuclear Service Water
Nuclear Solid Waste Disposal
Reactor Coolant
Refueling Water
Residual Heat Removal
Safety Injection
Spent Fuel Cooling
Standby Shutdown Diesel
Turbine Building Ventilation
Turbine Exhaust
Upper Containment Ventilation
Waste Gas

Table 2.2-1 McGuire Systems and Structures within the Scope of License Renewal (continued)

The following McGuire mechanical systems are within the scope of license renewal only because a portion of each system provides valves and piping for containment isolation purposes. For convenience, these systems are described collectively in Section 2.3.2.2, Containment Isolation System:

Breathing Air	Ice Condenser Refrigeration
Containment Air Release & Addition System	Makeup Demineralized Water
Containment Purge Ventilation	Station Air
Containment Ventilation Cooling Water	Steam Generator Blowdown Recycle
Conventional Chemical Addition	Steam Generator Wet Lay-Up Recirculation
Equipment Decontamination	

Table 2.2-1 McGuire Systems and Structures within the Scope of License Renewal (continued)

McGuire Structures within the Scope of License Renewal

Auxiliary Building	Reactor Building Unit 2
Condenser Cooling Water Intake Structure (Fire Pump Room only)	Reactor Makeup Water Storage Tank Foundation
Control Building	Refueling Water Storage Tank Foundation
Diesel Generator Building Unit 1	Refueling Water Storage Tank Missile Wall
Diesel Generator Building Unit 2	Service Building
Groundwater Drainage System	Standby Nuclear Service Water Pond Dam
Main Steam Doghouses Unit 1	Standby Nuclear Service Water Pond Discharge Structure (Nuclear Service Water Discharge Structure)
Main Steam Doghouses Unit 2	Standby Nuclear Service Water Pond Intake Structure (Nuclear Service Water Intake Structure)
New Fuel Storage Building Unit 1	Spent Fuel Building Unit 1
New Fuel Storage Building Unit 2	Spent Fuel Building Unit 2
New Fuel Storage Vault Unit 1	Standby Shutdown Facility
New Fuel Storage Vault Unit 2	Turbine Building Unit 1
Trenches	Turbine Building Unit 2
Reactor Building Unit 1	Unit Vent Stack

Table 2.2-1 McGuire Systems and Structures within the Scope of License Renewal (continued)

McGuire Electrical Systems within the Scope of License Renewal

Electrical scoping was performed at the system and component level. Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet any of the scoping criteria of §54.4(a). Other electrical and instrumentation and control systems and components are within the scope of license renewal as part of a bounding scope. No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

Table 2.2-2 Catawba Systems and Structures within the Scope of License Renewal

Catawba Mechanical Systems within the Scope of License Renewal

Annulus Ventilation
Auxiliary Building Ventilation
Auxiliary Feedwater
Auxiliary Steam
Boron Recycle
Building Heating Water
Chemical & Volume Control
Component Cooling
Condensate
Condensate Storage
Condenser Circulating Water
Containment Air Return Exchange & Hydrogen Skimmer
Containment Spray
Containment Valve Injection Water
Control Area Chilled Water
Control Room Area Ventilation
Diesel Building Ventilation
Diesel Generator Engine Air Intake & Exhaust
Diesel Generator Engine Cooling Water
Diesel Generator Engine Crankcase Vacuum
Diesel Generator Engine Fuel Oil
Diesel Generator Engine Lube Oil
Diesel Generator Engine Starting Air
Diesel Generator Room Sump Pump
Drinking Water
Feedwater
Feedwater Pump Turbine Exhaust
Feedwater Pump Turbine Hydraulic Oil
Fire Protection
Fuel Handling Area Ventilation
Groundwater Drainage
Hydrogen Bulk Storage
Instrument Air
Liquid Radwaste
Main Steam
Main Steam Auxiliary Equipment
Main Steam Vent to Atmosphere
Main Turbine Hydraulic Oil
Main Turbine Lube Oil and Purification
Miscellaneous Structures Ventilation
Nitrogen
Nuclear Sampling
Nuclear Service Water
Nuclear Service Water Pump Structure Ventilation
Reactor Coolant
Recirculated Cooling Water
Refueling Water
Residual Heat Removal
Safety Injection
Solid Radwaste
Spent Fuel Cooling

Table 2.2-2 Catawba Systems and Structures within the Scope of License Renewal

(continued)

Catawba Mechanical Systems within the Scope of License Renewal

Standby Shutdown Diesel	Waste Gas
Turbine Building Sump Pump	

The following Catawba mechanical systems are within the scope of license renewal only because a portion of each system provides valves and piping for containment isolation purposes. For convenience, these systems are described collectively in Section 2.3.2.2, Containment Isolation System:

Breathing Air	Ice Condenser Refrigeration
Containment Air Release & Addition	Make-up Demineralized Water
Containment Hydrogen Sample & Purge	Station Air
Containment Purge	Steam Generator Blowdown
Equipment Decontamination	Steam Generator Wet Layup Recirculation

Table 2.2-2 Catawba Systems and Structures within the Scope of License Renewal (continued)

Catawba Structures within the Scope of License Renewal

Auxiliary Building	Reactor Building Unit 2
Control Complex	Refueling Water Storage Tank Foundation
Diesel Generator Building Unit 1	Refueling Water Storage Tank Missile Shield
Diesel Generator Building Unit 2	Refueling Water Storage Tank Pipe Trench
Doghouse Unit 1	Service Building
Doghouse Unit 2	Standby Nuclear Service Water Pond Dam
Fuel Building	Standby Nuclear Service Water Discharge Structure
Fuel Pool	Standby Nuclear Service Water Intake Structure
Groundwater Drainage System	Standby Nuclear Service Water Pond Outlet
Low Pressure Service Water Intake Structure	Standby Shutdown Facility
Nuclear Service Water and Standby Nuclear Service Water Pump Structure	Trenches
Nuclear Service Water Conduit Manholes	Turbine Building Unit 1
Nuclear Service Water Intake Structure	Turbine Building Unit 2
Reactor Building Unit 1	Unit Vent Stack
	Upper Head Injection Tank Building

Table 2.2-2 Catawba Systems and Structures within the Scope of License Renewal (continued)

Catawba Electrical Systems within the Scope of License Renewal

Electrical scoping was performed at the system and component level. Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet any of the scoping criteria of §54.4(a). Other electrical and instrumentation and control systems and components are within the scope of license renewal as part of a bounding scope. No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

Table 2.2-3 McGuire Systems and Structures Not within the Scope of License Renewal

McGuire Mechanical Systems Not within the Scope of License Renewal

Administration Building HVAC	Heater Vents
Auxiliary Fuel Oil	Heating Boiler Feedwater
Boron Thermal Regeneration	Heating Boiler Fuel Gas
CO_2 Generator Purge	Hydrogen Blanket
Condensate	Incore Instrumentation Area Ventilation
Condensate Storage	Main Steam Bypass To Condenser
Condenser Circulating Water Intake Screen Backwash	Main Steam To Feedwater Pump Turbine
Condenser Cleaning	Main Turbine Leakoff & Steam Seal
Condenser Steam Air Ejector	Main Vacuum
Conventional Low Pressure Service Water	Miscellaneous Outside Structures HVAC
Conventional Sampling	Moisture Separator-Reheater Bleed Steam
Drinking Water	Moisture Separator-Reheater Drains
Equipment Staging Building Ventilation	On-Site Technical Support Center Ventilation
Feedwater Pump Condensate Seal	Oxygen
Feedwater Pump Turbine Lube Oil	Reactor Building / Control Rod Drive Ventilation
Feedwater Pump Turbine Steam Seal	Recirculated Cooling Water
Filtered Water	Sanitation And Waste Treatment
Generator Hydrogen System	Service Building And Warehouse Ventilation
Generator Seal Oil	Turbine Crossover
Generator Stator Cooling	Turbine Room Sump Pump
Heater Bleed Steam A, B, C, D, E, F, G	Unwatering Pump System
Heater Drains	Vacuum Priming
Heater Relief Valves	Waste Oil

Table 2.2-3 McGuire Systems and Structures Not within the Scope of License Renewal

(continued)

McGuire Structures Not within the Scope of License Renewal

230 kV Switch Station Steel and Foundation	Earthen Dam and Dike Extension of Cowans Ford Dam North of McGuire
525 kV Switch Station	Equipment Staging Building
Administration Building	Final Holdup Pond
Advanced Training Facility	Guardhouse
Bahnson Building	Hazardous Waste Storage Facility / Bottle Gas Storage
Ballistics Vault	Hydrogen Storage Facility
BBQ Shed	Initial Holding Pond
Blower House	Landfill Area
Canteen, Office & Warehouse	McGuire Garage
Condenser Circulating Water Low Level Intake Structure	McGuire Office Complex
Condenser Circulating Water Low Level Pump Structure	Meteorological Instrument House
Chemical Mixing	Mulsifyer Houses (7)
CO_2 Cylinder Storage House	Nitrogen Storage Facility
Communications Back-up Diesel Building	North Personnel Access Portal
Communications Buildings (3)	Office Building (2)
Compacted Waste Storage Building	Office Shop Building
Condenser Cooling Water Intake Structure (except the Fire Pump Rooms)	Oil Storage House
Condenser Discharge Structure	Oil Water Separator Building
Contract Services Building	Operations Office Addition
Corporate Medical Facility	Oxygen Storage Facility
Cowans Ford Auxiliary Intake Structure	Paint Shed
Cowans Ford Dam	Picnic Shelter
Demineralized Water Processing Building & Equipment	Pumphouse
Diesel Building (#7434)	Pumphouse Fish Tank

Table 2.2-3 McGuire Systems and Structures Not within the Scope of License Renewal

(continued)

McGuire Structures Not within the Scope of License Renewal

(continued)

Radiographics Shooting Vault	Training Stand & Storage
Radwaste Facility	Transmitter Shelters (4)
Recirculation Pump House	Upper McGuire Hydro Office
Relay House	Warehouses (2)
Restroom	Waste Solidification Building
Retired Steam Generator Storage Facility	Waste Treatment Pumphouse / Chemical Mixing Station
Settling Pond A,B	Waste Water Collection Basin Dam (Non-Nuclear Waste)
Shed	Water Treatment Supply
Storage Buildings (5)	Wellhouse
Target Storage	Yard Equipment Storage Building

McGuire Electrical Systems Not within the Scope of License Renewal

Electrical scoping was performed at the system and component level. Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet any of the scoping criteria of §54.4(a). Other electrical and instrumentation and control systems and components are within the scope of license renewal as part of a bounding scope. No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

Table 2.2-4 Catawba Systems and Structures Not within the Scope of License Renewal

Catawba Mechanical Systems Not within the Scope of License Renewal

Administration Building Chilled Water
Administration Building Ventilation
Auxiliary Boiler Feedwater
Auxiliary Building Cooling Water
Boron Thermal Regeneration
Catawba Steam Production Office Chilled Water
Chemical Cleaning
Computer Area Ventilation
Computer Room Area Chilled Water
Condenser Cleaning
Condenser Steam Air Ejector
Containment Chilled Water
Containment Ventilation
Conventional Chemical Addition
Conventional Low Pressure Service Water
Conventional Sampling
Conventional Waste Water Treatment
Cooling Tower Water Treatment
Feedwater Pump Condensate Seal
Feedwater Pump Turbine Lube Oil
Feedwater Pump Turbine Steam Seal
Filtered Water
Generator CO_2 Purge
Generator Hydrogen
Generator Seal Oil
Generator Stator Cooling Water

Heater Bleed Steam A, B, C, D, E, F And G
Heater Drain
Heater Relief Valve
Heater Vent
Hydrogen Blanket
Low Pressure Service Water Intake Screen Backwash
Main Steam Bypass To Condenser
Main Turbine Leakoff And Steam Seal
Main Vacuum
Moisture Separator Reheater Bleed Steam
Moisture Separator Reheater Drain
Monitor Tank Building HVAC
On-Site Technical Support Center Ventilation
Oxygen
Radwaste Area Chilled Water
Sanitation And Waste Treatment
Service Building Chilled Water
Service Building Sump Pump
Service Building Ventilation
Steam Supply To Feedwater Pump Turbine
Turbine Building Ventilation
Turbine Crossover
Turbine Exhaust Hood Spray
Unwatering Pump
Vacuum Priming

Table 2.2-4 Catawba Systems and Structures Not within the Scope of License Renewal

(continued)

Catawba Structures Not within the Scope of License Renewal

230 kV Switching Station
Administration Building
Bottled Gas Storage House
Carpenter Shop
Chlorination House
Communication Building
Compressor Shed (2)
Containment Mechanical Equipment Building Unit 1
Containment Mechanical Equipment Building Unit 2
Contaminated Material Storage & Warehouse
Cooling Towers
Discharge Canal and Dike
Electrical Cable Trenches
Energyquest
Gang Box Storage Building
Garage
Guardhouse
Hazardous Waste Storage Building
Hot Machine Shop
HVAC/Chemical Storage
Hydrogen Storage House
Low Pressure Service Water Canal and Dike
Low Pressure Service Water Discharge Structure
Lube Oil Storage House
Maintenance Facility
Maintenance Training Facility
Medical Facility
Metal Fabrication Shop
Microwave & Environmental Building
Monitor Tank Building
Motor Control Center House
MTU Communication Building
Nitrogen Storage Building
Office Buildings (3)
Office Complex
Oxygen Storage Building
Paint Drying Shed
Paint Mix/Storage Building
Paint Storage Building
Picnic Shed
Pipe Cut Shed
Pole Shed
Pole Storage
Pumphouse
Relay House
Restrooms
Restrooms @ CNS Park
Retired Steam Generator Facility
Radiation Protection Storage Tent
Sandblast Building

Table 2.2-4 Catawba Systems and Structures Not within the Scope of License Renewal

(continued)

Catawba Structures Not within the Scope of License Renewal

(continued)

Shed	Training Building
Spare Diesel Generator	Truck Tent
Standby Nuclear Service Water Pond Instrument Pier	Warehouses (4)
Steam Generator Drain Tank Building	Water Chemistry Building
Target Range House	Wellhouses (2)

Catawba Electrical Systems Not within the Scope of License Renewal

Electrical scoping was performed at the system and component level. Switchyard Systems, Unit Main Power System, Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System and uninsulated ground conductors were found not to meet any of the scoping criteria of §54.4(a). Other electrical and instrumentation and control systems and components are within the scope of license renewal as part of a bounding scope. No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope.

2.3 SYSTEM SCOPING AND SCREENING RESULTS: MECHANICAL

2.3.1 REACTOR COOLANT SYSTEM

Note: The Reactor Coolant System description and component descriptions are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

The methodology to identify the mechanical systems within the scope of license renewal is described in Section 2.1 of this Application. Section 2.1.2.1 contains a description of the methodology to identify the mechanical components that are subject to aging management review.

Systems, structures and components are located in this section consistent with their locations in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (July 1981) except where the internal operating environment or function suggests a more appropriate location within the Application. In these situations, the system, structure or component has been located in the more appropriate section. For example, Class 1 Component Supports which are normally described within Chapter 5 of NUREG-0800, are described in Section 2.4 of this Application because they are functionally structural components and are included in Section 2.4 of the draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (August 2000).

This section of the Application applies to the Class 1 portions of the Reactor Coolant System and ancillary piping from adjoining systems that form the Reactor Coolant pressure boundary. The license renewal evaluation boundaries for the Class 1 portions of the Reactor Coolant System extend onto the flow diagrams of the following systems:

- Safety Injection System
- Chemical and Volume Control System
- Residual Heat Removal System

Non-class 1 portions of the Reactor Coolant System are described in Section 2.3.3 of this Application. The Reactor Coolant Pump Motor Oil Collection Sub-System is described in Section 2.3.3.31 and the Reactor Coolant System (Non-Class Components) are described in Section 2.3.3.32. The following Reactor Coolant System Class 1 components are described in Section 2.3.1, Reactor Coolant System:

- Class 1 Piping, Valves, and Pumps
- Pressurizer
- Reactor Vessel and Control Rod Drive Mechanism Pressure Boundary

- Reactor Vessel Internals
- Steam Generator

2.3.1.1 Reactor Coolant System Description

The Reactor Coolant System consists of four similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator and associated piping and valves. In addition, the system includes a pressurizer, a pressurizer relief tank (Class F), interconnecting piping and instrumentation necessary for operational control. All major components are located in the Reactor Building.

During operation, the Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine generator. Borated demineralized water is circulated in the Reactor Coolant System at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The Reactor Coolant System pressure boundary provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the unit. Reactor Coolant System pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters or water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring-loaded safety valves and power operated relief valves are mounted on the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

The McGuire UFSAR Chapter 5, Reactor Coolant System, provides additional information concerning the McGuire Reactor Coolant System. The mechanical components, component functions, and materials of construction for the McGuire Reactor Coolant System are listed in Table 3.1-1. The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Reactor Coolant System:

MCFD-1553-01.00	MCFD-1562-02.00	MCFD-2554-01.01
MCFD-1553-02.00	MCFD-1562-02.01	MCFD-2554-01.02
MCFD-1553-02.01	MCFD-1562-03.00	MCFD-2561-01.00
MCFD-1554-01.00	MCFD-1562-03.01	MCFD-2562-01.00
MCFD-1554-01.01	MCFD-2553-01.00	MCFD-2562-02.00
MCFD-1554-01.02	MCFD-2553-02.00	MCFD-2562-02.01
MCFD-1561-01.00	MCFD-2553-02.01	MCFD-2562-03.00
MCFD-1562-01.00	MCFD-2554-01.00	MCFD-2562-03.01

These flow diagrams are contained in Reference [2.3.1-1]

The Catawba UFSAR Chapter 5, Reactor Coolant System, provides additional information concerning the Catawba Reactor Coolant System. The mechanical components, component functions, and materials of construction for the Catawba Reactor Coolant System are listed in Table 3.1-1. The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Reactor Coolant System:

CN-1553-1.0	CN-1562-1.1	CN-2561-1.0
CN-1553-1.1	CN-1562-1.2	CN-2561-1.1
CN-1554-1.0	CN-1562-1.3	CN-2562-1.0
CN-1554-1.5	CN-2553-1.0	CN-2562-1.1
CN-1561-1.0	CN-2553-1.1	CN-2562-1.2
CN-1561-1.1	CN-2554-1.0	CN-2562-1.3
CN-1562-1.0	CN-2554-1.5	

These flow diagrams are contained in Reference [2.3.1-1].

2.3.1.2 Class 1 Piping, Valves and Pumps

The Reactor Coolant System Class 1 piping and associated pressure boundary components consist of:

- Westinghouse supplied primary loop piping which interconnects the reactor vessel, steam generators, and reactor coolant pumps;
- Duke-designed Class 1 piping;
- Pressure boundary portion of Class 1 valves (bodies and bonnets, bolting); and
- Pressure boundary portion of the reactor coolant pump (casing, main closure flange, thermal barrier heat exchanger and bolting).

The Westinghouse supplied primary loop piping consists of four loops of piping interconnecting the reactor vessel, steam generator and reactor coolant pump in each loop. This piping includes branch connection nozzles and special items such as the RTD scoop elements, pressurizer spray scoop, sample connection scoop, reactor coolant temperature element installation boss, and the temperature element well itself.

Class 1 branch piping consists of piping connected at the Westinghouse supplied primary loop piping out to and including (1) the outermost containment isolation valve in piping which penetrates primary containment, or (2) the second of two valves normally closed during normal reactor operation in piping which does not penetrate primary containment. Some Class 1 branch lines and instrument connections in the Reactor Coolant System are equipped with 3/8 inch ID flow restricting orifices that limit the maximum flow from a break downstream of the flow restrictor to below the makeup capability of the Reactor Coolant System. This orifice is used to make the break from Class 1 to Class 2 instead of double isolation valves.

For Class 1 valves, the pressure-retaining portion of the component consists of the valve body, bonnet and closure bolting. The valves are welded in place with the exception of the pressurizer safety valves that have flanged connections.

For the reactor coolant pumps, the pressure-retaining portion of the component includes the pump casing, the main closure flange, the thermal barrier heat exchanger within the reactor coolant pump, the reactor coolant pump seals and the pressure retaining bolting. The reactor coolant pump seals are excluded from aging management review because they are periodically replaced. Preventive maintenance is currently scheduled every three cycles for the reactor coolant pump seals unless data indicates that the inspection must be done more frequently.

The McGuire UFSAR Section 5.5, Component and Subsystem Design, provides additional information concerning the McGuire Reactor Coolant System Class 1 piping and associated pressure boundary components. The Catawba UFSAR Section 5.4, Component and Subsystem Design, provides additional information concerning the Catawba Reactor Coolant System Class 1 piping and associated pressure boundary components. The mechanical components, component functions, and materials of construction for the McGuire and Catawba Reactor Coolant System Class 1 piping and associated pressure boundary components are listed in Table 3.1-1.

2.3.1.3 Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads that is connected to the Reactor Coolant System on one of the hot legs of a coolant loop. Electrical heaters are installed through the bottom head of the pressurizer while the spray nozzle, relief and safety valve connections are located in the top head of the pressurizer. The McGuire UFSAR Section 5.5.10, Pressurizer, provides additional information concerning the McGuire pressurizer. The Catawba UFSAR Section 5.4.10, Pressurizer, provides additional information concerning the Catawba pressurizer. The mechanical components, component functions, and materials of construction for all four of the McGuire and Catawba pressurizers are listed in Table 3.1-1.

2.3.1.4 Reactor Vessel and Control Rod Drive Mechanism Pressure Boundary

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the core, core supporting structures, control rods and other parts directly associated with the core. The upper (closure) head contains 82 penetrations (78 for control rod drive mechanism penetrations and 4 auxiliary head adapters). The vessel has inlet and outlet nozzles located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

The bottom head of the vessel contains 58 penetrations for connection and entry of the nuclear incore instrumentation. Each penetration consists of a tubular member made of Inconel. Each tube is attached inside of the bottom head by a partial penetration weld. Stainless steel conduits extend from the Inconel penetration in the bottom head of the reactor vessel down through the concrete shield area and up to a thimble shield table. The retractable thimble tubes, which travel within the conduit, are closed at the leading ends, are dry inside, and serve as the pressure barrier between the reactor water pressure and the Reactor Building atmosphere. Mechanical seals between the retractable thimbles and the conduits are provided at the seal table.

The McGuire UFSAR Section 5.4, Reactor Vessel, provides additional information concerning the McGuire reactor vessel. The Catawba UFSAR Section 5.3, Reactor Vessel, provides additional information concerning the Catawba reactor vessel. The mechanical components, component functions, and materials of construction for all four of the McGuire and Catawba reactor vessels are listed in Table 3.1-1.

2.3.1.5 Reactor Vessel Internals

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the in-core instrumentation support structure. The reactor vessel internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements and to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the in-core instrumentation.

The McGuire UFSAR Section 4.2.2, Reactor Vessel Internals, provides additional information concerning the McGuire reactor vessel internals. The Catawba UFSAR Section 3.9.5, Reactor Vessel Internals, provides additional information concerning the Catawba reactor vessel internals. The mechanical components, component functions, and materials of construction for all four of the McGuire and Catawba reactor vessel internals are listed in Table 3.1-1.

2.3.1.6 Steam Generator

The replacement steam generators at McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Unit 1 were manufactured by Babcock & Wilcox International in Cambridge, Ontario, Canada. The McGuire Unit 1 steam generators were replaced in May 1997, and the Unit 2 steam generators were replaced in December 1998. The Catawba Unit 1 steam generators were replaced in October 1996. For Catawba Nuclear Station, Unit 2, the steam generators that were installed during original construction have not been replaced.

All steam generators at both stations are vertical shell and U-tube evaporators with integral moisture separating equipment. Reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles equipped with stainless steel safe ends located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side of the tubes and flows upward through the moisture separators to the outlet nozzle at the top of the steam generator. Feedwater flows directly into a downcomer section and is mixed with saturated recirculation flow before entering the tube bundle for the replacement steam generators. The Catawba Unit 2 steam generators are equipped with a preheater and feedwater flow restrictor with main feedwater delivered just above the tube sheet. Subsequently, the water-steam mixture flows upward through the tube bundle and into the steam drum section. Centrifugal moisture separators, located above the tube bundle, remove most of the entrained water from the steam.

The McGuire UFSAR Section 5.5.2, Steam Generator, provides additional information concerning the McGuire steam generators. The Catawba UFSAR Section 5.4.2, Steam Generator, provides additional information concerning the Catawba steam generators. The mechanical components, component functions, and materials of construction for all sixteen of the McGuire and Catawba steam generators are listed in Table 3.1-1.

2.3.1.7 References for Section 2.3.1

2.3.1-1. M. S. Tuckman (Duke) letter dated June 13, 2001 to Document Control Desk (NRC), *License Renewal Evaluation Boundary Drawings.* McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2, Docket Nos. 50-369, 50-370, 50-413, and 50-414.

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2.3.2 ENGINEERED SAFETY FEATURES

The methodology to identify the mechanical systems within the scope of license renewal is described in Section 2.1.1of this Application. Section 2.1.2.1 contains a description of the methodology to identify the mechanical components that are subject to aging management review.

Systems, structures and components are located in this section consistent with their locations in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (July 1981) except where the internal operating environment or function suggests a more appropriate location within the Application. In these situations, the system, structure or component has been located in the more appropriate section. For example, the Refueling Water System which is normally described within Chapter 9 of NUREG-0800, is described in this Section of the Application because of its function to support Engineered Safety Features and its internal operating environment is the same as many of the other systems contained in this section of the Application.

The following mechanical systems are described in Section 2.3.2, Engineered Safety Features in the section indicated:

- Annulus Ventilation System (Section 2.3.2.1)
- Containment Isolation System (Section 2.3.2.2)
- Containment Air Return Exchange & Hydrogen Skimmer System (Section 2.3.2.3)
- Containment Spray System (Section 2.3.2.4)
- Containment Valve Injection Water System (Section 2.3.2.5)
- Refueling Water System (Section 2.3.2.6)
- Residual Heat Removal System (Section 2.3.2.7)
- Safety Injection System (Section 2.3.2.8)

2.3.2.1 Annulus Ventilation System

McGuire Nuclear Station – The Annulus Ventilation System is an engineered safety feature that creates and maintains a negative pressure zone in the annular space between the steel Primary Containment and Reactor Building (Secondary Containment), to prevent the leakage of radioisotopes through the Reactor Building and into the environment, following a loss-of-coolant-accident (LOCA). The Annulus Ventilation System is also designed to maintain Containment isolation integrity. The McGuire UFSAR Section 6.2 , Containment Systems, provides additional information concerning the McGuire Annulus Ventilation System. The mechanical components, component functions, and materials of construction for the McGuire Annulus Ventilation System are listed in Table 3.2-1.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Annulus Ventilation System:

MC-1564-1.0	MC-2564-1.0
MC-1577-1.0	MC-2577-1.0

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Annulus Ventilation System is an engineered safety feature used in conjunction with the Secondary Containment to limit operator and site boundary doses, following a Design Basis Accident, to within the guidelines specified in 10 CFR 100 and provides long-term, fission product removal capability within the annulus through holdup and filtration. The Catawba UFSAR Section 6.2, Containment Systems, provides additional information concerning the Catawba Annulus Ventilation System. The mechanical components, component functions, and materials of construction for the Catawba Annulus Ventilation System are listed in Table 3.2-1.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Annulus Ventilation System:

CN-1564-1.0	CN-2564-1.0
CN-1577-3.0	CN-2577-3.0

These flow diagrams are contained in Reference [2.3.2-1]

2.3.2.2 Containment Isolation System

Note: The Containment Isolation System is an engineered safety feature that provides for the closure of all fluid penetrations not required for operation of the Engineered Safeguards System to prevent the leakage of uncontrolled or unmonitored radioactive materials to the environment. Where specific differences exist between McGuire and Catawba, a brief explanation is provided. Section 6.2.4, Containment Isolation Systems, of each station's UFSAR provide additional information on the containment isolation feature of the systems described in Section 2.3.2.2.

2.3.2.2.1 BREATHING AIR SYSTEM

McGuire Nuclear Station – The Breathing Air System provides an adequate capacity of air to meet appropriate American National Standards Institute (ANSI) specifications. Two full capacity compressors are provided to furnish the total average breathing air requirements. The Breathing Air System is also relied upon to provide and maintain Containment isolation and closure. The Breathing Air System contains Duke Class B piping and components for Containment isolation. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The McGuire UFSAR Section 9.3.1, Compressed Air Systems, provides additional information concerning the McGuire Breathing Air System. The mechanical components, component functions, and materials of construction for the McGuire Breathing Air System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Breathing Air System:

MCFD-1605-03.01 MCFD-2605-03.01

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Breathing Air System supplies clean, oil free, compressed air to various locations in the Auxiliary Building, Monitor Tank Building, and Containment for breathing protection against airborne contamination while performing certain maintenance and cleaning operations. The Breathing Air System is relied upon to provide and maintain Containment isolation and closure. The Breathing Air System contains Duke Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The Catawba UFSAR Section 9.3.1, Compressed Air Systems, provides additional information concerning the Breathing Air System. The mechanical components, component functions, and materials of construction for the Catawba Breathing Air System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Breathing Air System:

CN-1605-3.1	CN-1605-3.2	CN-2605-3.2

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.2 CONTAINMENT AIR RELEASE AND ADDITION SYSTEM

McGuire Nuclear Station – The Containment Air Release and Addition System maintains containment pressure between the McGuire Technical Specification limits of -0.3 to +0.3 psig. Increases in pressure during normal operation are controlled by venting the containment through the Containment Air Release and Addition filters. The Containment Air Release and Addition System contains Duke Class B piping and components for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The McGuire UFSAR Section 9.5.12, Containment Air Release and Addition System, provides additional information concerning the McGuire Containment Air Release and Addition System. The mechanical components, component functions, and materials of construction for the McGuire Containment Air Release and Addition System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Containment Air Release and Addition System:

MCFD-1585-01.00	MCFD-2585-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Containment Air Release and Addition System maintains containment pressure between the Catawba Technical Specification limits of -0.1 to +0.3 psig during normal plant operation. An increase in pressure during normal operation is controlled by the containment air release fans taking suction from the containment and passing through the containment air release filters. The Containment Air Release and Addition System contains Duke Class B piping and components for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The Catawba UFSAR Section 9.5.10, Containment Air Release and Addition System, provides additional information concerning the Containment Air Release and

Addition System. The mechanical components, component functions, and materials of construction for the Catawba Containment Air Release and Addition System are listed in Table 3.2-2.

The following is a list of the flow diagrams and instrument details that have been marked to indicate the license renewal evaluation boundary for the Catawba Containment Air Release and Addition System:

CN-1585-1.0	CN-1499-VQ-04.00
CN-2585-1.0	CN-2499-VQ-04.00

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.3 CONTAINMENT HYDROGEN SAMPLE AND PURGE SYSTEM

McGuire Nuclear Station – No system corresponding to the Catawba Containment Hydrogen Sample and Purge System exists at McGuire.

Catawba Nuclear Station – The Containment Hydrogen Sample and Purge System is used after a loss-of-coolant accident (LOCA) to monitor the hydrogen concentration inside Containment, and if necessary, reduce the levels of hydrogen by manually purging the hydrogen from Containment into the annulus. The Containment Hydrogen Sample and Purge System contains Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The Catawba UFSAR Section 6.2.5.3.2, Containment Hydrogen Sample and Purge System, provides additional information concerning the Containment Hydrogen Sample and Purge System. The mechanical components, component functions, and materials of construction for the Catawba Containment Hydrogen Sample and Purge System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Containment Hydrogen Sample and Purge System:

CN-1559-1.0	CN-2559-1.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.4 CONTAINMENT PURGE VENTILATION SYSTEM

McGuire Nuclear Station – The Containment Purge Ventilation System reduces the airborne radioactivity levels in containment by purging the upper containment atmosphere to the environment via the unit vent stack during periods of sustained personnel access (including refueling). The Containment Purge Ventilation System contains Duke Class B piping for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The McGuire UFSAR Section 9.4.5, Containment, provides additional information concerning the McGuire Containment Purge Ventilation System. The mechanical components, component functions, and materials of construction for the McGuire Containment Purge Ventilation System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Containment Purge Ventilation System:

MC-1576-01.00	MC-2576-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Containment Purge System reduces the airborne radioactivity levels in containment by purging the upper containment, lower containment, and the incore instrumentation room atmosphere to the unit vent stack during refueling when periods of personnel access are required. The Containment Purge System contains Duke Class B piping for the containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The Catawba UFSAR Section 9.4.5, Containment Purge Ventilation System, provides additional information concerning the Containment Purge System. The mechanical components, component functions, and materials of construction for the Catawba Containment Purge System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Containment Purge System:

CN-1576-1.0	CN-2576-1.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.5 CONTAINMENT VENTILATION COOLING WATER SYSTEM

McGuire Nuclear Station – The Containment Ventilation Cooling Water System operates in conjunction with the Nuclear Service Water System to supply cooling water to ventilation units located in the Reactor and Auxiliary Buildings. The Containment Ventilation Cooling Water System contains Duke Class B and C piping for the containment isolation function. All components within the Duke Class B and C boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The McGuire UFSAR Section 6.2.4, Containment Isolation Systems, provides additional information concerning the McGuire Containment Ventilation Cooling Water System. The mechanical components, component functions, and materials of construction for the McGuire Containment Ventilation Cooling Water System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Containment Ventilation Cooling Water System:

MCFD-1574-01.00	MCFD-1604-03.03	MCFD-2604-03.01
MCFD-1604-03.00	MCFD-2604-03.00	

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – This system does not exist at Catawba. The comparable components cooled by the McGuire Containment Ventilation Cooling Water System are cooled by the Catawba Nuclear Service Water System.

2.3.2.2.6 CONVENTIONAL CHEMICAL ADDITION SYSTEM

McGuire Nuclear Station – The Conventional Chemical Addition System uses the Auxiliary Feedwater supply headers to provide chemical addition to the steam generators. The Conventional Chemical Addition System contains Duke Class B piping for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The McGuire UFSAR Section 6.2.4, Containment Isolation Systems, provides additional information concerning the McGuire Conventional Chemical Addition System. The mechanical components, component functions, and materials of construction for the McGuire Conventional Chemical Addition System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Conventional Chemical Addition System:

MCFD-1617-01.00	MCFD-2617-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – This system does not exist at Catawba. The comparable components to the McGuire Conventional Chemical Addition System are contained in the Catawba Auxiliary Feedwater System.

2.3.2.2.7 EQUIPMENT DECONTAMINATION

McGuire Nuclear Station – The Equipment Decontamination System provides decontamination of station equipment before handling by personnel. The original design of McGuire included Containment isolation capability. This design is now different than the design of Catawba. The original design has been modified by the installation of a sleeve cap on the annulus side of the penetration. Associated with the capped penetration is some remaining piping and components designated as Class F. These piping and components have been determined to have no component intended function. Therefore, no mechanical components in the Equipment Decontamination System are subject to aging management review. The sleeve cap and penetration and Class F supports are addressed as part of the Structural review discussed in Section 2.4 of the Application.

The following is a list of the flow diagrams that show the design of the McGuire Equipment Decontamination System:

MCFD-1568-01.00	MCFD-2568-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Equipment Decontamination System provides cleaning and decontamination of radioactive equipment prior to handling, maintenance or shipping. The Equipment Decontamination System and its components are not safety related, with the exception of the portions associated with Containment Isolation. The Equipment Decontamination System is relied upon to maintain two trains of containment isolation and maintain containment closure for shutdown. The Equipment Decontamination System contains Duke Class B pipe for the containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The Catawba UFSAR Section 6.2.4, Containment Isolation System, provides additional information concerning the Containment Isolation System. The mechanical components, component functions, and materials of construction for the Catawba Equipment Decontamination System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Equipment Decontamination System:

CN-1568-1.0	CN-2568-1.0
CN-1570-1.0	CN-2570-1.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.8 ICE CONDENSER REFRIGERATION

McGuire Nuclear Station – The primary safety function of the Ice Condenser Refrigeration System is to rapidly reduce, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptable levels. The safety-related function of the mechanical systems portion of the Ice Condenser Refrigeration System is Containment isolation. The Ice Condenser Refrigeration System contains Duke Class B piping and components for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The McGuire UFSAR Section 6.2.2, Ice Condenser System, provides additional information concerning the McGuire Ice Condenser Refrigeration System. The mechanical components, component functions, and materials of construction for the McGuire Ice Condenser Refrigeration System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Ice Condenser Refrigeration System:

MCFD-1558-04.00 MCFD-2558-04.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The primary safety function of the Ice Condenser Refrigeration System is to rapidly reduce, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptable levels. The safety-related function of the mechanical systems portion of the Ice Condenser Refrigeration System is Containment isolation. The Ice Condenser Refrigeration System contains Duke Class B piping and components for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The Catawba UFSAR Section 6.7.6, Refrigeration System, provides additional information concerning the Ice Condenser Refrigeration System. The mechanical components, component functions, and materials of construction for the Catawba Ice Condenser Refrigeration System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Ice Condenser Refrigeration System:

CN-1558-1.0 CN-1558-2.0 CN-2558-2.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.9 MAKEUP DEMINERALIZED WATER SYSTEM

McGuire Nuclear Station – The Makeup Demineralized Water System provides treated and demineralized water to various plant systems and components. The Demineralized Water System contains Duke Class B piping and components for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The McGuire UFSAR Section 9.2.6, Treated Water Systems, provides additional information concerning the McGuire Demineralized Water System. The mechanical components, component functions, and materials of construction for the McGuire Demineralized Water System are listed in Table 3.2-2.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the McGuire Demineralized Water System:

MCFD-1601-02.04

This flow diagram is contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Makeup Demineralized Water System provides treated and demineralized water to various plant systems and components. The Makeup Demineralized Water System contains Duke Class B piping and components for the Containment isolation function. All components within the Duke Class B boundaries are within the scope of license renewal per §54.4(a)(1). The Catawba UFSAR Section 9.2.3, Makeup Demineralized Water System, provides additional information concerning the Makeup Demineralized Water System. The mechanical components, component functions, and materials of construction for the Catawba Makeup Demineralized Water System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Makeup Demineralized Water System:

CN-1601-3.1	CN-1601-3.5	CN-2556-2.0
CN-1601-3.2	CN-1556-2.0	

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.10 STATION AIR SYSTEM

McGuire Nuclear Station – The Station Air System provides an adequate capacity for general station service air requirements. Normally, the Instrument Air System provides the station air requirements through system cross-connect valves. However, if needed, one Station Air System compressor is provided to furnish the station air requirements if the Instrument Air System is not available or desired. The Station Air System is also relied upon to provide and maintain Containment isolation and closure. The Station Air System contains Duke Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The McGuire UFSAR Section 9.3.1, Compressed Air Systems, provides additional information concerning the McGuire Station Air System. The mechanical components, component functions, and materials of construction for the McGuire Station Air System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Station Air System:

MCFD-1605-02.02 MCFD-2605-02.02

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Station Air System supplies low pressure compressed air for air operated tools, miscellaneous equipment, and various maintenance purposes. The Station Air System, if required, is available to act as a backup supply of compressed air for the Instrument Air System. The Station Air System is relied upon to provide and maintain Containment isolation and closure. The Station Air System contains Duke Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The Catawba UFSAR Section 9.3.1, Compressed Air Systems, provides additional information concerning the Catawba Station Air System. The mechanical components, component functions, and materials of construction for the Catawba Station Air System are listed in Table 3.2-2.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the Catawba Station Air System:

CN-1605-2.1

This flow diagram is contained in Reference [2.3.2-1].

2.3.2.2.11 STEAM GENERATOR BLOWDOWN RECYCLE SYSTEM

McGuire Nuclear Station – The Steam Generator Blowdown Recycle System is used in conjunction with the Condensate System to maintain acceptable secondary side water chemistry and control corrosion product buildup. The Steam Generator Blowdown Recycle System is designed to maintain containment isolation integrity. The system automatically isolates the blowdown lines penetrating the containment following receipt of a containment isolation signal and also following a start signal of the Auxiliary Feedwater System. The Steam Generator Blowdown Recycle System contains Duke Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The McGuire UFSAR Section 10.4.8, Steam Generator Blowdown System, provides additional information concerning the McGuire Steam Generator Blowdown Recycle System. The mechanical components, component functions, and materials of construction for the McGuire Steam Generator Blowdown Recycle System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Steam Generator Blowdown Recycle System:

MCFD-1580-01.00	MCFD-2580-01.00
MCFD-1580-01.01	MCFD-2580-01.01
MCFD-1584-01.00	MCFD-2584-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Steam Generator Blowdown System is used in conjunction with the Condensate System to maintain acceptable secondary side water chemistry and control corrosion product buildup. The Steam Generator Blowdown System is designed to maintain containment isolation integrity. The system automatically isolates the blowdown lines penetrating the containment following the receipt of a containment isolation signal, and also, following a start signal of the Auxiliary Feedwater System. The Steam Generator Blowdown System contains Duke Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The Catawba UFSAR

Section 10.4.8, Steam Generator Blowdown System, provides additional information concerning the Steam Generator Blowdown System. The mechanical components, component functions, and materials of construction for the Catawba Steam Generator Blowdown System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Steam Generator Blowdown System:

CN-1565-2.6	CN-2565-2.6
CN-1580-1.0	CN-2580-1.0
CN-1584-1.0	CN-2584-1.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.2.12 STEAM GENERATOR WET LAY-UP RECIRCULATION SYSTEM

McGuire Nuclear Station – The Steam Generator Wet Lay-Up Recirculation System maintains containment isolation integrity. The Steam Generator Wet Lay-Up Recirculation System contains Duke Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The McGuire UFSAR Section 6.2.4, Containment Isolation System, provides additional information concerning the Steam Generator Wet Lay-Up Recirculation System. The mechanical components, component functions, and materials of construction for the McGuire Steam Generator Wet Lay-Up Recirculation System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Steam Generator Wet Lay-Up Recirculation System:

MCFD-1584-01.00	MCFD-2584-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Steam Generator Wet Lay-Up Recirculation System maintains containment isolation integrity. The Steam Generator Wet Lay-Up Recirculation System contains Duke Class B piping and components for the Containment isolation function. All piping and components within the Duke Class B boundaries are within the scope of license renewal in accordance with §54.4(a)(1). The Catawba UFSAR Section 10.4.8, Steam Generator Blowdown System, provides additional information concerning the Steam Generator Blowdown System. The mechanical components, component functions, and materials of construction for the Catawba Steam Generator Wet Lay-Up Recirculation System are listed in Table 3.2-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Steam Generator Wet Lay-Up Recirculation System:

CN-1584-1.0	CN-2584-1.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.3 Containment Air Return Exchange & Hydrogen Skimmer System

McGuire Nuclear Station – The Containment Air Return Exchange and Hydrogen Skimmer System (1) maintains Containment pressure less than the design pressure during any high energy line break (HELB), (2) ensures hydrogen concentration remains less than the flammability limit during a loss-of-coolant-accident (LOCA), and (3) maintains Containment isolation integrity for the system piping penetrating the Containment. The McGuire UFSAR Section 6.2, Containment Systems, provides additional information concerning the McGuire Containment Air Return Exchange and Hydrogen Skimmer System. The mechanical components, component functions, and materials of construction for the McGuire Containment Air Return Exchange and Hydrogen Skimmer System are listed in Table 3.2-3.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Containment Air Return Exchange and Hydrogen Skimmer System:

MC-1557-1.0	MC-2557-1.0

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Containment Air Return Exchange and Hydrogen Skimmer System (1) maintains Containment pressure less than the design pressure during any high energy line break (HELB), (2) ensures hydrogen concentration remains less than the flammability limit during a loss-of-coolant-accident (LOCA), and (3) maintains Containment isolation integrity for the system piping penetrating the Containment. The Catawba UFSAR Section 6.2, Containment Systems, provides additional information concerning the Catawba Containment Air Return Exchange and Hydrogen Skimmer System. The mechanical components, component functions, and materials of construction for the Catawba Containment Air Return Exchange and Hydrogen Skimmer System are listed in Table 3.2-3.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Containment Air Return Exchange and Hydrogen Skimmer System:

CN-1557-1.0	CN-2557-1.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.4 Containment Spray System

McGuire Nuclear Station – The Containment Spray System is an engineered safety feature that serves to remove thermal energy from the Containment in the event of a loss-of-coolant-accident or a Main Steam Line Break. It performs this function in conjunction with the Emergency Core Cooling System, which subcools the reactor by direct injection. After all the ice in the ice condenser has melted, the heat removal capability of the spray system will keep the Containment pressure below the design pressure. The Containment Spray System also serves to remove fission product iodine from the post-accident Containment atmosphere. The McGuire UFSAR Section 6.5, Containment Spray System, provides additional information concerning the McGuire Containment Spray System. The mechanical components, component functions, and materials of construction for the McGuire Containment Spray System are listed in Table 3.2-4.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Containment Spray System:

MCFD-1563-01.00 MCFD-2563-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Catawba Containment Spray System is an engineered safeguard feature that serves to remove thermal energy from the Containment atmosphere in the event of a loss-of-coolant-accident. It performs this function in conjunction with the Emergency Core Cooling System, which cools the reactor during injection and recirculation modes of operation. The heat removal capability of the spray system maintains containment pressure below the design pressure value after the ice in the Ice Condenser has been depleted, and steam generated in the core continues to enter Containment. The Containment Spray System also serves to remove fission product iodine from the post-accident Containment atmosphere. In addition, the Containment Spray System is designed for the suppression of steam partial pressure in the upper Containment volume due to operating deck leakage from a LOCA. The Catawba UFSAR Section 6.2.2, Containment Heat Removal Systems, provides additional information concerning the Catawba Containment Spray System. The mechanical components, component functions, and materials of construction for the Catawba Containment Spray System are listed in Table 3.2-4.

The following is a list of the flow diagrams and instrument details that have been marked to indicate the license renewal evaluation boundary for the Catawba Containment Spray System:

CN-1499-NS.02-00	CN-2499-NS.02-00
CN-1499-VQ.04-00	CN-2499-VQ.04-00
CN-1563-1.0	CN-2563-1.0

These flow diagrams and instrument details are contained in Reference [2.3.2-1].

2.3.2.5 Containment Valve Injection Water System

McGuire Nuclear Station – The design and licensing basis of McGuire Nuclear Station does not contain any system that is functionally equivalent to the Catawba Containment Valve Injection Water System.

Catawba Nuclear Station – The Containment Valve Injection Water System is designed to inject water between the two seating surfaces of double disc gate valves used for Containment isolation. The injection pressure is higher than Containment design peak pressure during a LOCA. This will prevent leakage of the Containment atmosphere through the gate valves, thereby reducing potential offsite dose below the values specified by 10 CFR 100 limits following the postulated accident. The Catawba UFSAR Section 6.2.4, Containment Isolation System, provides additional information concerning the Catawba Containment Valve Injection Water System. The mechanical components, component functions, and materials of construction for the Catawba Containment Valve Injection Water System are listed in Table 3.2-5.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Containment Valve Injection Water System:

CN-1569-1.0	CN-1573-1.3	CN-2563-1.0
CN-2569-1.0	CN-1574-2.0	CN-2565-2.0
CN-1553-1.1	CN-1574-2.2	CN-2565-2.1
CN-1554-1.0	CN-1574-2.4	CN-2565-2.4
CN-1562-1.2	CN-1574-2.8	CN-2565-2.6
CN-1562-1.3	CN-1599-2.1	CN-2573-1.3
CN-1563-1.0	CN-1599-2.2	CN-2574-2.0
CN-1565-2.0	CN-2553-1.1	CN-2574-2.2
CN-1565-2.1	CN-2554-1.0	CN-2574-2.4
CN-1565-2.4	CN-2562-1.2	CN-2574-2.7
CN-1565-2.6	CN-2562-1.3	

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.6 Refueling Water System

McGuire Nuclear Station – The Refueling Water System provides a source of borated water to be used during refueling, for the Emergency Core Cooling System to mitigate the consequences of a FSAR Chapter 15 accident or as borated makeup water for the spent fuel pool. The system can remove impurities from the refueling cavity and transfer canal during refueling, and it can clean up the Refueling Water Storage Tank water following refueling. This can be accomplished by routing flow through the purification loop of the Spent Fuel Pool Cooling System. The Refueling Water System provides a means of transferring the final 30% of the refueling water between the refueling cavity and the Refueling Water Storage Tank. It also provides a secondary means of filling the refueling cavity from the Refueling Water Storage Tank. The McGuire UFSAR Section 9.2.5, Refueling Water System, provides additional information concerning the McGuire Refueling Water System. The mechanical components, component Table 3.2-6.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Refueling Water System:

MCFD-1554-03.00	MCFD-1563-01.00	MCFD-2561-01.00
MCFD-1554-03.01	MCFD-1571-01.00	MCFD-2562-03.00
MCFD-1561-01.00	MCFD-2554-03.00	MCFD-2563-01.00
MCFD-1562-03.00	MCFD-2554-03.01	MCFD-2571-01.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Refueling Water System provides an adequate supply of borated water to the Emergency Core Cooling System and Containment Spray System in order to mitigate the consequences of a Design Basis Event. The Refueling Water System, along with the Safety Injection System, Residual Heat Removal System, and Chemical and Volume Control System function together to form the Emergency Core Cooling System. The Catawba UFSAR Section 9.2.7, Refueling Water System, provides additional information concerning the Catawba Refueling Water System. The mechanical components, component functions, and materials of construction for the Catawba Refueling Water System are listed in Table 3.2-6.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Refueling Water System:

CN-1554-1.2	CN-1570-1.0	CN-2562-1.2
CN-1554-1.7	CN-1571-1.0	CN-2563-1.0
CN-1561-1.0	CN-2554-1.2	CN-2570-1.0
CN-1562-1.2	CN-2554-1.7	CN-2571-1.0
CN-1563-1.0	CN-2561-1.0	

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.7 Residual Heat Removal System

McGuire Nuclear Station – The Residual Heat Removal System transfers heat from the Reactor Coolant System to the Component Cooling System to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of unit cooldown and maintains this temperature until the unit is started up. The Residual Heat Removal System also serves as part of the Emergency Core Cooling System during the injection and recirculation phases of Small-Break and Large-Break Loss of Coolant Accidents. The McGuire UFSAR Section 6.3, Emergency Core Cooling System, provides additional information concerning the McGuire Residual Heat Removal System. The mechanical components, component functions, and materials of construction for the McGuire Residual Heat Removal System are listed in Table 3.2-7.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Residual Heat Removal System:

MCFD-1561-01.00	MCFD-1571-01.00	MCFD-2562-03.00
MCFD-1554-01.02	MCFD-1572-01.00	MCFD-2562-03.01
MCFD-1554-02.00	MCFD-2561-01.00	MCFD-2563-01.00
MCFD-1562-03.00	MCFD-2554-01.02	MCFD-2571-01.00
MCFD-1562-03.01	MCFD-2554-02.00	MCFD-2572-01.00
MCFD-1563-01.00		

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Residual Heat Removal System transfers heat from the Reactor Coolant System to the Component Cooling System to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second phase of unit cooldown and maintains this temperature until the unit is started up. The Residual Heat Removal System also serves as part of the Emergency Core Cooling System during the injection and recirculation phases of Design Basis Events. The Residual Heat Removal System has several secondary functions which includes transferring refueling water between the Refueling Water Storage Tank and the Refueling Cavity before and after refueling operations, providing overpressure protection to the Reactor Coolant System, providing Reactor Coolant letdown flow for pressure control and purification during shutdown and refueling, and providing Residual Heat Removal Auxiliary Pressurizer Spray. The Catawba UFSAR Section 6.3, Emergency Core Cooling System, provides additional information concerning the Catawba Residual Heat Removal System. The mechanical components, component functions, and materials of construction for the Catawba Residual Heat Removal System are listed in Table 3.2-7.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Residual Heat Removal System:

CN-1561-1.0	CN-1563-1.0	CN-2554-1.6
CN-1561-1.1	CN-1571-1.0	CN-2562-1.2
CN-1554-1.0	CN-1572-1.0	CN-2562-1.3
CN-1554-1.6	CN-2561-1.0	CN-2563-1.0
CN-1562-1.2	CN-2561-1.1	CN-2571-1.0
CN-1562-1.3	CN-2554-1.0	CN-2572-1.0

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.8 Safety Injection System

McGuire Nuclear Station – The Safety Injection System constitutes a major portion of the Emergency Core Cooling System. Along with the Residual Heat Removal, Chemical and Volume Control and Refueling Water Systems, the Safety Injection System provides emergency cooling to the reactor core in the event of a break in either the primary (reactor coolant) or secondary (steam) systems. The three primary functions of the Emergency Core Cooling System are: (1) removing stored (sensible) and fission product decay heat, (2) controlling reactivity, and (3) precluding reactor vessel boron precipitation. The Safety Injection System supports each of these functions. The McGuire UFSAR Section 6.3, Emergency Core Cooling System, provides additional information concerning the McGuire Safety Injection System. The mechanical components, component functions, and materials of construction for the McGuire Safety Injection System are listed in Table 3.2-8.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Safety Injection System:

MCFD-1562-01.00	MCFD-1562-03.01	MCFD-2562-02.01
MCFD-1562-02.00	MCFD-1562-04.00	MCFD-2562-03.00
MCFD-1562-02.01	MCFD-2562-01.00	MCFD-2562-03.01
MCFD-1562-03.00	MCFD-2562-02.00	MCFD-2562-04.00

These flow diagrams are contained in Reference [2.3.2-1].

Catawba Nuclear Station – The Safety Injection System constitutes a major portion of the Emergency Core Cooling System (ECCS). Along with the Residual Heat Removal, Chemical and Volume Control and Refueling Water Systems, the Safety Injection System provides emergency cooling to the reactor core in the event of a break in either the primary (reactor coolant) or secondary (steam) systems. The three primary functions of the ECCS are: (1) removing stored (sensible) and fission product decay heat, (2) controlling reactivity, and (3) precluding reactor vessel boron precipitation. The Safety Injection System supports each of these functions. The Catawba UFSAR Section 6.3, Emergency Core Cooling System, provides additional information concerning the Catawba Safety Injection System. The mechanical components, component functions, and materials of construction for the Catawba Safety Injection System are listed in Table 3.2-8.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Safety Injection System:

CN-1562-1.0	CN-1562-1.3	CN-2562-1.2
CN-1562-1.1	CN-2562-1.0	CN-2562-1.3
CN-1562-1.2	CN-2562-1.1	

These flow diagrams are contained in Reference [2.3.2-1].

2.3.2.9 References for Section 2.3.2

2.3.2-1. M. S. Tuckman (Duke) letter dated June 13, 2001 to Document Control Desk (NRC), *License Renewal Evaluation Boundary Drawings.* McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2, Docket Nos. 50-369, 50-370, 50-413, and 50-414.

2.3.3 AUXILIARY SYSTEMS

The methodology to identify the mechanical systems within the scope of license renewal is described in Section 2.1.1 of this Application. Section 2.1.2.1 contains a description of the methodology to identify the mechanical components that are subject to aging management review.

Systems, structures and components are located in this section consistent with their locations in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (July 1981) except where the internal operating environment or function suggests a more appropriate location within the Application. In these situations, the system, structure or component has been located in the more appropriate section. For example, spent fuel storage racks, cranes, and fire barriers which are normally described within the Chapter 9 of NUREG-0800, are described in Section 2.4, Structures and Structural Components, of this Application because their functions are structural rather than mechanical.

The following mechanical systems are described in Section 2.3.3, Auxiliary Systems, in the sections as indicated:

- Auxiliary Building Ventilation System (Section 2.3.3.1)
- Boron Recycle System (Section 2.3.3.2)
- Building Heating Water System (Section 2.3.3.3)
- Chemical & Volume Control System (Section 2.3.3.4)
- Component Cooling System (Section 2.3.3.5)
- Condenser Circulating Water System (Section 2.3.3.6)
- Containment Ventilation System (Section 2.3.3.7)
- Control Area Ventilation System and Chilled Water Systems (Section 2.3.3.8)
- Conventional Waste Water Treatment (Section 2.3.3.9)
- Diesel Building Ventilation (Section 2.3.3.10)
- Diesel Generator Air Intake and Exhaust System [Footnote 2.3.3-1] (Section 2.3.3.11)
- Diesel Generator Cooling Water System [Footnote 2.3.3-1] (Section 2.3.3.12)
- Diesel Generator Crankcase Vacuum System [Footnote 2.3.3-1] (Section 2.3.3.13)
- Diesel Generator Fuel Oil System [Footnote 2.3.3-1] (Section 2.3.3.14)
- Diesel Generator Lube Oil System [Footnote 2.3.3-1] (Section 2.3.3.15)
- Diesel Generator Room Sump Pump (Section 2.3.3.16)
- Diesel Generator Starting Air System [Footnote 2.3.3-1] (Section 2.3.3.17)
- Drinking Water System (Section 2.3.3.18)

2.3.3-1. Catawba Nuclear Station system designations include "Engine" in these system names (See Table 2.2-2). Hereafter, either system name may be used for Catawba.

- Fire Protection System (Section 2.3.3.19)
- Fuel Handling Area Ventilation System (Section 2.3.3.20)
- Fuel Handling Building Ventilation System (Section 2.3.3.20)
- Groundwater Drainage System (Section 2.3.3.21)
- Heating Water System (Section 2.3.3.3)
- Hydrogen Bulk Storage System (Section 2.3.3.22)
- Instrument Air System (Section 2.3.3.23)
- Liquid Radwaste System (Section 2.3.3.24)
- Liquid Waste Recycle System (Section 2.3.3.24)
- Liquid Waste Monitor and Disposal System (Section 2.3.3.24)
- Miscellaneous Structures Ventilation System (Section 2.3.3.25)
- Nitrogen System (Section 2.3.3.26)
- Nuclear Sampling System (Section 2.3.3.27)
- Nuclear Service Water System (Section 2.3.3.28)
- Nuclear Service Water Pump Structure Ventilation System (Section 2.3.3.29)
- Nuclear Solid Waste Disposal System (Section 2.3.3.30)
- Reactor Coolant Pump Motor Oil Collection Sub-System (Section 2.3.3.31)
- Reactor Coolant System (Non-Class 1 Components) (Section 2.3.3.32)
- Recirculated Cooling Water System (Section 2.3.3.33)
- Solid Radwaste System (Section 2.3.3.30)
- Spent Fuel Cooling System (Section 2.3.3.34)
- Standby Shutdown Diesel (Section 2.3.3.35)
- Turbine Building Sump Pump System (Section 2.3.3.36)
- Turbine Building Ventilation System (Section 2.3.3.37)
- Waste Gas System (Section 2.3.3.38)

2.3.3.1 Auxiliary Building Ventilation System

McGuire Nuclear Station – The Auxiliary Building Ventilation System automatically aligns to maintain the Emergency Core Cooling System (ECCS) Pump Rooms at a negative pressure so that air exhausted from these rooms is filtered prior to being released following a Design Basis Accident (DBA). The ECCS Pump Rooms include the Safety Injection Pumps, Residual Heat Removal Pumps, Centrifugal Charging Pumps and Containment Spray Pumps. The McGuire UFSAR Section 9.4.2, Auxiliary Building, provides additional information concerning the McGuire Auxiliary Building Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Auxiliary Building Ventilation System are listed in Table 3.3-1.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Auxiliary Building Ventilation System:

MC-1577-1	MC-1577-5	MC-1577-9
MC-1577-2	MC-1577-8	MC-2577-1
MC-1577-4		

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Auxiliary Building Ventilation System automatically aligns to maintain the Emergency Core Cooling System (ECCS) Pump Rooms at a negative pressure so that air exhausted from these rooms is filtered prior to being released following a Design Basis Accident (DBA). The ECCS Pump Rooms include the Safety Injection Pumps, Residual Heat Removal Pumps, Centrifugal Charging Pumps and Containment Spray Pumps. The Catawba UFSAR Section 9.4.3, Auxiliary Building Ventilation System provides additional information concerning the Catawba Auxiliary Building Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Auxiliary Building Ventilation System are listed in Table 3.3-1.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Auxiliary Building Ventilation System:

CN-1577-1.0	CN-1577-1.3	CN-1577-3.0
CN-1577-1.2	CN-1577-1.8	CN-2577-3.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.2 Boron Recycle System

McGuire Nuclear Station – The Boron Recycle System receives borated effluent from the Reactor Coolant System and associated support systems. This borated effluent is demineralized, filtered, and separated into 4 weight percent boric acid and reactor makeup water for reuse. The Boron Recycle System also provides reactor grade flush water for components in the Auxiliary and Reactor buildings. The McGuire UFSAR Section 9.3.6, Boron Recycle System, provides additional information concerning the McGuire Boron Recycle System. The mechanical components, component functions, and materials of construction for the McGuire Boron Recycle System are listed in Table 3.3-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Boron Recycle System:

MCFD-1554-02.00	MCFD-1556-02.01	MCFD-1567-01.00
MCFD-1556-01.00	MCFD-1556-03.00	MCFD-1567-04.00
MCFD-1556-01.01	MCFD-1565-02.00	MCFD-1595-01.02
MCFD-1556-02.00	MCFD-1565-02.01	MCFD-2556-03.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Boron Recycle System receives and recycles reactor coolant effluent for reuse of the boric acid and makeup water. The system decontaminates the effluent by means of demineralization and gas stripping and uses evaporation to separate and recover the boric acid and makeup water. Portions of the Boron Recycle System are shared between both reactor units, while other portions are unit specific. The Catawba UFSAR Section 9.3.5, Boron Recycle System, provides additional information concerning the Catawba Boron Recycle System. The mechanical components, component functions, and materials of construction for the Catawba Boron Recycle System are listed in Table 3.3-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Boron Recycle System:

CN-1554-1.1
CN-1554-1.7
CN-1556-1.0
CN-1556-1.1
CN-1556-1.2
CN-1556-1.3
CN-1556-1.4
CN-1556-1.5
CN-1556-1.6
CN-1556-2.0
CN-1561-1.0
CN-1561-1.1
CN-1562-1.2
CN-1562-1.3
CN-1563-1.0
CN-1565-1.4
CN-1567-1.0
CN-1567-1.3
CN-1595-1.0
CN-2554-1.1
CN-2554-1.2
CN-2554-1.7
CN-2556-2.0
CN-2561-1.0
CN-2561-1.1
CN-2562-1.2
CN-2562-1.3
CN-2563-1.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.3 Building Heating Water System

McGuire Nuclear Station – The Heating Water System provides normal heating requirements of the Auxiliary Building Ventilation System, Fuel Pool Ventilation System, Containment & Incore Instrumentation Room Purge System, Service Building Ventilation System, and the Turbine Building Heating System. The Heating Water System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Heating Water System are listed in Table 3.3-3.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the McGuire Heating Water System:

MCFD-1606-03.02

This flow diagram is contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Building Heating Water System supplies hot water to the heating coils of various HVAC units throughout the plant. The Building Heating Water System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Building Heating Water System are listed in Table 3.3-3.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Building Heating Water System:

CN-1606-1.0	CN-1606-1.7	CN-1606-1.9
CN-1606-1.6	CN-1606-1.8	CN-1606-1.10

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.4 Chemical & Volume Control System

McGuire Nuclear Station – The Chemical and Volume Control System is an integral part of the Emergency Core Cooling System (ECCS) and provides high pressure injection and recirculation of borated water to the Reactor Coolant System cold legs following Small-Break and Large-Break Loss of Coolant Accidents (SB and LBLOCAs) and Main Steam Line Breaks. The Chemical and Volume Control System is also used to provide negative reactivity, by boron injection, to the core. The McGuire UFSAR Section 9.3.4, Chemical and Volume Control System, provides additional information concerning the McGuire Chemical and Volume Control System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Chemical and Volume Control System are listed in Table 3.3-4.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Chemical and Volume Control System:

MCFD-1554-01.00	MCFD-1562-01.00	MCFD-2554-03.01
MCFD-1554-01.01	MCFD-1562-03.00	MCFD-2554-04.00
MCFD-1554-01.02	MCFD-1561-01.00	MCFD-2554-05.00
MCFD-1554-01.03	MCFD-1567-02.01	MC-2555-1.0
MCFD-1554-02.00	MCFD-1565-01.00	MC-2555-2.0
MCFD-1554-02.01	MCFD-1572-02.00	MCFD-2562-01.00
MCFD-1554-03.00	MCFD-2554-01.00	MCFD-2562-03.00
MCFD-1554-03.01	MCFD-2554-01.01	MCFD-2561-01.00
MCFD-1554-04.00	MCFD-2554-01.02	MCFD-2565-01.00
MCFD-1554-05.00	MCFD-2554-01.03	MCFD-2572-02.00
MC-1555-1.0	MCFD-2554-02.00	MCFD-2556-03.00
MC-1555-2.0	MCFD-2554-02.01	MCFD-1556-01.00
MCFD-1556-01.01	MCFD-2554-03.00	

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Chemical and Volume Control System is an integral part of the Emergency Core Cooling System (ECCS) and provides high pressure injection and recirculation of borated water to the Reactor Coolant System cold legs following Small-Break and Large-Break Loss of Coolant Accidents (SB and LBLOCAs) and Main Steam Line Breaks. The Chemical and Volume Control System is also used to provide negative reactivity, by boron injection, to the core. The Catawba UFSAR Section 9.3.4, Chemical and Volume Control System provides additional information concerning the Catawba Chemical and Volume Control System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Chemical and Volume Control System are listed in Table 3.3-5.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Chemical and Volume Control System:

CN-1554-1.0
CN-1554-1.1
CN-1554-1.2
CN-1554-1.3
CN-1554-1.4
CN-1554-1.5
CN-1554-1.6
CN-1554-1.7
CN-1554-1.8
CN-2554-1.0
CN-2554-1.1
CN-2554-1.2
CN-2554-1.3
CN-2554-1.4
CN-2554-1.5
CN-2554-1.6
CN-2554-1.7
CN-2554-1.8
CN-1555-1.1
CN-1556-1.0
CN-1561-1.0
CN-1562-1.0
CN-1562-1.2
CN-1567-1.0
CN-1567-1.1
CN-1567-1.3
CN-1570-1.0
CN-1572-1.2
CN-2555-1.1
CN-2561-1.0
CN-2562-1.0
CN-2562-1.2
CN-2570-1.0
CN-2572-1.2

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.5 Component Cooling System

McGuire Nuclear Station – The Component Cooling System is a closed loop system that maintains cooling to the essential header components as required for plant conditions, maintains an intermediate system pressure boundary between the Reactor Coolant System and the Nuclear Service Water System to prevent potential radioactive release, provides containment isolation, and maintains containment closure for shutdown. The McGuire UFSAR Section 9.2.4, Component Cooling System, provides additional information concerning the McGuire Component Cooling System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Component Cooling System are listed in Table 3.3-6.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Component Cooling System:

MCFD-1573-01.00	MCFD-1573-03.00	MCFD-2573-02.00
MCFD-1573-01.01	MCFD-1573-03.01	MCFD-2573-02.01
MCFD-1573-02.00	MCFD-1573-04.00	MCFD-2573-03.00
MCFD-1573-02.01	MCFD-2573-01.00	MCFD-2573-03.01
MCFD-1573-02.02	MCFD-2573-01.01	MCFD-2573-04.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Component Cooling System is a closed loop system relied upon to maintain cooling to the essential header components as required for plant conditions, maintains an intermediate pressure boundary between the Reactor Coolant System and the Nuclear Service Water System to prevent potential radioactive release, provides Containment isolation, and maintains Containment closure for shutdown. The Catawba UFSAR Section 9.2.2, Component Cooling System, provides additional information concerning the Catawba Component Cooling System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Component Cooling System are listed in Table 3.3-7.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Component Cooling System:

CN-1573-1.0	CN-1573-1.9	CN-2573-1.3
CN-1573-1.1	CN-1573-2.0	CN-2573-1.4
CN-1573-1.2	CN-1573-2.1	CN-2573-1.5
CN-1573-1.3	CN-1573-2.2	CN-2573-1.7
CN-1573-1.4	CN-1573-2.3	CN-2573-2.0
CN-1573-1.5	CN-2573-1.0	CN-2573-2.1
CN-1573-1.6	CN-2573-1.1	CN-2573-2.2
CN-1573-1.7	CN-2573-1.2	CN-2573-2.3

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.6 Condenser Circulating Water System

McGuire Nuclear Station – The Condenser Circulating Water System provides a suction source of water to the turbine-driven auxiliary feedwater pump for events requiring the activation of the Standby Shutdown Facility. The McGuire UFSAR Section 10.4.5, Condenser Circulating Water System, provides additional information concerning the McGuire Condenser Circulating Water System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Condenser Circulating Water System are listed in Table 3.3-8.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Condenser Circulating Water System:

MCFD-1604-01.02	MCFD-2604-01.00	MCFD-2604-01.02

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Condenser Circulating Water System provides a suction source of water to the turbine-driven auxiliary feedwater pump for events requiring the activation of the Standby Shutdown Facility. The Catawba UFSAR Section 10.4.5, Condenser Circulating Water System, provides additional information concerning the Catawba Condenser Circulating Water System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Condenser Circulating Water System are listed in Table 3.3-8.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Condenser Circulating Water System:

CN-1604-1.0	CN-1592-1.0	CN-2604-1.2
CN-1604-1.1	CN-2604-1.0	CN-2604-1.3
CN-1604-1.2	CN-2604-1.1	CN-2592-1.0
CN-1604-1.3		

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.7 Containment Ventilation Systems

McGuire Nuclear Station – The purpose of the Upper and Lower Containment Ventilation Systems is to provide cooling to the upper and lower compartments of Containment during normal operation and shutdown. The Upper and Lower Containment Ventilation Systems contain RTDs that are required for post-accident monitoring in accordance with the EQ Rule. No mechanical components have an intended function; therefore, no aging management review is required. Since instruments are not highlighted on the flow diagrams, no flow diagrams have been marked for these systems.

Catawba Nuclear Station – The corresponding Catawba system does not meet the license renewal scoping criteria.

2.3.3.8 Control Area Ventilation System and Chilled Water Systems

McGuire Nuclear Station – The Control Area Ventilation System and the Control Area Chilled Water Systems combine to form one system to provide the normal and emergency ventilation requirements to the Control Room and Control Room Area. The McGuire UFSAR Section 6.4, Control Area (Habitability) Ventilation System, provides additional information concerning the McGuire Control Area Ventilation System and the Control Area Chilled Water Systems. The mechanical components, component functions, and materials of construction for the McGuire Control Area Ventilation System and the Control Area Chilled Water Systems are listed in Table 3.3-11 and Table 3.3-9, respectively.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Control Area Ventilation System and the Control Area Chilled Water Systems:

MCFD-1577-1.0	MCFD-1578-02.00	MCFD-1618-01.00
MCFD-1578-1.0	MCFD-1578-03.00	MCFD-1618-02.00
MCFD-1578-1.1	MCFD-1578-04.00	MCFD-1618-04.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Control Room Area Ventilation System and Control Area Chilled Water System combine to form one system whose purpose is to provide the normal and emergency ventilation requirements to the Control Room and Control Room Area. The Catawba UFSAR, Section 9.4.1, Control Room Area Ventilation, provides additional information concerning the Catawba Control Room Area Ventilation and Control Area Chilled Water Systems. The mechanical components, component functions, and materials of construction for the Catawba Control Room Area Ventilation and Control Area Chilled Water Systems are listed in Table 3.3-11 and Table 3.3-10, respectively.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Control Room Area Ventilation and Control Area Chilled Water Systems:

CN-1578-1.0	CN-1578-2.0	CN-1578-2.3
CN-1578-1.1	CN-1578-2.1	CN-1578-2.4
CN-1578-1.2	CN-1578-2.2	CN-1578-2.5
CN-1578-1.3		

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.9 Conventional Waste Water Treatment System

McGuire Nuclear Station – The Conventional Waste Water Treatment System maintains low water level in the Standby Shutdown Facility (SSF) sump to prevent flooding of SSF equipment. The McGuire UFSAR Section 9.2.8, Conventional Waste Water Treatment System, provides additional information concerning the McGuire Conventional Waste Water Treatment System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Conventional Waste Water Treatment System are listed in Table 3.3-12.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the McGuire Conventional Wastewater Treatment System:

MCFD-1583-01.00

This flow diagram is contained in Reference [2.3.3-1].

Catawba Nuclear Station – No system corresponding to the McGuire Conventional Waste Water Treatment System is within the scope of license renewal at Catawba. The SSF sump pump, specifically credited at McGuire, does not meet the license renewal scoping criteria at Catawba.

2.3.3.10 Diesel Building Ventilation System

McGuire Nuclear Station – The Diesel Building Ventilation System maintains temperature control for each Diesel Building when its associated diesel generator is running. The McGuire UFSAR Section 9.4.6, Diesel Building, provides additional information concerning the McGuire Diesel Building Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Building Ventilation System are listed in Table 3.3-13.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Building Ventilation System:

MC-1579-1.0 MC-2579-1.0

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Building Ventilation System maintains temperature control for each Diesel Building when its associated diesel generator is running. The Catawba UFSAR Section 9.4.4, Diesel Building Ventilation System, provides additional information concerning the Catawba Diesel Building Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Building Ventilation System are listed in Table 3.3-13.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Building Ventilation System:

CN-1579-1.0 CN-2579-1.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.11 Diesel Generator Air Intake and Exhaust System

McGuire Nuclear Station – The Diesel Generator Air Intake and Exhaust System supplies sufficient air to the diesel generator engines for fuel consumption and removes exhaust from the diesel generator engines to the atmosphere outside the building. The McGuire UFSAR Section 9.5.11, Diesel Generator Air Intake and Exhaust System, provides additional information concerning the McGuire Diesel Generator Air Intake and Exhaust System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Generator Air Intake and Exhaust System are listed in Table 3.3-14.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Generator Air Intake and Exhaust System:

MCFD-1609-05.00 MCFD-2609-05.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Generator Engine Air Intake and Exhaust System supplies sufficient air to the diesel generator engines for fuel consumption and removes exhaust from the diesel generator engines to the atmosphere outside the building. The Catawba UFSAR Section 9.5.8, Diesel Generator Air Intake and Exhaust System, provides additional information concerning the Catawba Diesel Generator Engine Air Intake and Exhaust System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Generator Engine Air Intake and Exhaust System are listed in Table 3.3-14.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Generator Engine Air Intake and Exhaust System:

CN-1609-5.0 CN-2609-5.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.12 Diesel Generator Cooling Water System

McGuire Nuclear Station – The Diesel Generator Cooling Water System maintains the temperature of each emergency diesel generator engine and support systems within a required operating range. The McGuire UFSAR Section 9.5.5, Diesel Generator Cooling Water System, provides additional information concerning the McGuire Diesel Generator Cooling Water System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Generator Cooling Water System are listed in Table 3.3-15.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Generator Cooling Water System:

MCFD-1609-01.00	MCFD-1609-07.00	MCFD-2609-01.01
MCFD-1609-01.01	MCFD-2609-01.00	MCFD-2609-07.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Generator Engine Cooling Water System maintains the temperature of each emergency diesel generator engine and support systems within a required operating range. The Catawba UFSAR Section 9.5.5, Diesel Generator Engine Cooling Water, provides additional information concerning the Catawba Diesel Generator Engine Cooling Water System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Generator Engine Cooling Water System are listed in Table 3.3-16.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Generator Engine Cooling Water System:

CN-1609-1.0	CN-2609-1.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.13 Diesel Generator Crankcase Vacuum System

McGuire Nuclear Station – The Diesel Generator Crankcase Vacuum System purges the diesel engine crankcase to reduce the concentration of combustible gases. The McGuire UFSAR Section 9.5.9, Diesel Generator Crankcase Vacuum System, provides additional information concerning the McGuire Diesel Generator Crankcase Vacuum System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Generator Crankcase Vacuum System are listed in Table 3.3-17.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Generator Crankcase Vacuum System:

MCFD-1609-06.00 MCFD-2609-06.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Generator Engine Crankcase Vacuum System purges the diesel engine crankcase to reduce the concentration of combustible gases. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Generator Engine Crankcase Vacuum System are listed in Table 3.3-17.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Generator Engine Crankcase Vacuum System:

CN-1609-6.0 CN-2609-6.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.14 Diesel Generator Fuel Oil System

McGuire Nuclear Station – The Diesel Generator Fuel Oil System is relied upon to maintain two trains of fuel oil storage and supply for the Emergency Diesel Generators for a period of operation of no less than five days. The McGuire UFSAR Section 9.5.4, Diesel Generator Fuel Oil System, provides additional information concerning the McGuire Diesel Generator Fuel Oil System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Generator Fuel Oil System are listed in Table 3.3-18.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Generator Fuel Oil System:

MCFD-1609-03.00	MCFD-2609-03.00
MCFD-1609-03.01	MCFD-2609-03.01

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Generator Engine Fuel Oil System is relied upon to maintain two trains of fuel oil storage and supply for the Emergency Diesel Generators for a period of operation of no less than seven days. The Catawba UFSAR Section 9.5.4, Diesel Generator Engine Fuel Oil System, provides additional information concerning the Catawba Diesel Generator Engine Fuel Oil System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Generator Engine Fuel Oil System are listed in Table 3.3-19.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Generator Engine Fuel Oil System:

CN-1609-3.0	CN-2609-3.0
CN-1609-3.1	CN-2609-3.1

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.15 Diesel Generator Lube Oil System

McGuire Nuclear Station – The Diesel Generator Lube Oil System supplies lubricating oil to the diesel engine and its bearings, crankshaft, thrust faces, and other friction surfaces during both standby mode and operation mode of the diesel generator. The McGuire UFSAR Section 9.5.7, Diesel Generator Lubricating Oil System, provides additional information concerning the McGuire Diesel Generator Lube Oil System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Generator Lube Oil System are listed in Table 3.3-20.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Generator Engine Lube Oil System:

MCFD-1609-02.00	MCFD-2609-02.00
MCFD-1609-02.01	MCFD-2609-02.01

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Generator Engine Lube Oil System supplies lubricating oil to the diesel engine and its bearings, crankshaft, thrust faces, and other friction surfaces during both standby mode and operation mode of the diesel generator. The Catawba UFSAR Section 9.5.7, Diesel Generator Engine Lube Oil System, provides additional information concerning the Catawba Diesel Generator Engine Lube Oil System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Generator Engine Lube Oil System are listed in Table 3.3-21.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Generator Engine Lube Oil System:

CN-1609-2.0	CN-2609-2.0
CN-1609-2.2	CN-2609-2.2

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.16 Diesel Generator Room Sump Pump System

McGuire Nuclear Station – The Diesel Generator Room Sump Pump System removes leakage from equipment drains in the Diesel Building and protects the diesel generators from flooding due to a Nuclear Service Water System pipe rupture in one of the diesel rooms acting simultaneously with a turbine building flood. The McGuire UFSAR Section 9.5.10, Diesel Generator Room Sump Pump System, provides additional information concerning the McGuire Diesel Generator Room Sump Pump System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Generator Room Sump Pump System are listed in Table 3.3-22.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Generator Room Sump Pump System:

MCFD-1609-01.00	MCFD-2609-01.00	MCFD-2609-07.00
MCFD-1609-07.00	MCFD-2609-01.01	

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Generator Room Sump Pump System removes normal leakage and drainage from various equipment in the diesel generator rooms. The Catawba UFSAR Section 9.5.9, Diesel Generator Room Sump Pump System, provides additional information concerning the Catawba Diesel Generator Room Sump Pump System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Generator Room Sump Pump System are listed in Table 3.3-22.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Generator Room Sump Pump System:

CN-1609-7.0	CN-2609-7.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.17 Diesel Generator Starting Air System

McGuire Nuclear Station – The Diesel Generator Starting Air System provides fast start capability for the emergency diesel engine by using compressed air to roll the engine until it starts. The Diesel Generator Starting Air System also supplies air to the diesel controls to operate and shutdown the engine. The McGuire UFSAR Section 9.5.6, Diesel Generator Starting Air System, provides additional information concerning the McGuire Diesel Generator Starting Air System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Diesel Generator Starting Air System are listed in Table 3.3-23.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Diesel Generator Starting Air System:

MCFD-1609-04.00	MCID-1499-VG.03
MCFD-2609-04.00	MCID-2499-VG.03

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Diesel Generator Engine Starting Air System provides fast start capability for the emergency diesel engine by using compressed air to roll the engine until it starts. The Diesel Generator Engine Starting Air System also supplies air to the diesel controls to operate and shutdown the engine. The Catawba UFSAR Section 9.5.6, Diesel Generator Engine Starting Air System, provides additional information concerning the Catawba Diesel Generator Engine Starting Air System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Diesel Generator Engine Starting Air System are listed in Table 3.3-24.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Diesel Generator Engine Starting Air System:

CN-1609-4.0	CN-2609-4.0
CN-1609-4.1	CN-2609-4.1

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.18 Drinking Water System

McGuire Nuclear Station – No portion of the McGuire Drinking Water System is within the scope of license renewal. Only the Duke Class F portions of the Drinking Water System are in scope at Catawba. McGuire has no Class F components in the Drinking Water System.

Catawba Nuclear Station – The Drinking Water System is a municipal water system consisting of a water tower, pumps, and chemical treatment equipment providing chlorinated drinking water to the plant. The Drinking Water System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Drinking Water System are listed in Table 3.3-25.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Drinking Water System:

CN-1601-2.3 CN-1601-2.4

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.19 Fire Protection System

McGuire Nuclear Station – The Interior/Exterior Fire Protection System provides fire suppression to protect the capability to shut down the reactor and maintain it in a safe shutdown condition and to minimize radioactive releases to the environment in the event of a fire. In addition, the system provides water to the condenser circulating water pump and low level intake pump bearings. The McGuire UFSAR Section 9.5.1, Fire Protection System, provides additional information concerning the McGuire Interior/Exterior Fire Protection System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Interior/Exterior Fire Protection System are listed in Table 3.3-26.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Interior/Exterior Fire Protection System:

MCFD-1599-01.00	MCFD-1599-02.03	MCFD-2599-04.00
MCFD-1599-02.01	MCFD-1599-03.00	MCFD-1604-01.00
MCFD-1599-02.02	MCFD-1599-04.00	

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Interior/Exterior Fire Protection System provides fire suppression to protect the capability to shut down the reactor and maintain it in a safe shutdown condition and to minimize radioactive releases to the environment in the event of a fire. The Catawba UFSAR Section 9.5.1, Fire Protection System provides additional information concerning the Catawba Interior/Exterior Fire Protection System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Interior/Exterior Fire Protection System are listed in Table 3.3-27.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Interior/Exterior Fire Protection System:

CN-1599-1.0	CN-1599-2.4	CN-1574-2.1
CN-1599-1.2	CN-1599-4.0	CN-1574-2.5
CN-1599-2.1	CN-1599-4.2	CN-2574-2.1
CN-1599-2.2	CN-2599-4.0	CN-2574-2.5
CN-1599-2.3	CN-2599-4.2	

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.20 Fuel Handling Building Ventilation System

McGuire Nuclear Station – The Fuel Handling Building Ventilation System maintains ventilation in the spent fuel pool of Units 1 and 2 to permit personnel access. The exhaust portion of the Fuel Handling Building Ventilation System controls airborne radioactivity in the fuel pool area during normal operation, anticipated operational transients, and following postulated fuel handling accidents. The McGuire UFSAR Section 9.4.2, Auxiliary Building, provides additional information concerning the McGuire Fuel Handling Building Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Fuel Handling Building Ventilation System are listed in Table 3.3-28.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Fuel Handling Building Ventilation System:

MC-1577-1.0	MC-2577-1.0
MC-1577-3.0	MC-2577-3.0

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Fuel Handling Area Ventilation System maintains ventilation in the spent fuel pool of Units 1 and 2 to permit personnel access. The exhaust portion of the Fuel Handling Area Ventilation System controls airborne radioactivity in the fuel pool area during normal operation, anticipated operational transients, and following postulated fuel handling accidents. The Catawba UFSAR Section 9.4.2, Fuel Building Ventilation System, provides additional information concerning the Catawba Fuel Handling Area Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Fuel Handling Area Ventilation System are listed in Table 3.3-28.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Fuel Handling Area Ventilation System:

CN-1577-2.0	CN-1577-3.0	CN-1577-2.1
CN-2577-2.0	CN-2577-3.0	CN-2577-2.1

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.21 Groundwater Drainage System

McGuire Nuclear Station – The Groundwater Drainage System prevents hydrostatic loads on the Reactor and Auxiliary Building substructures. The Groundwater Drainage System maintains an acceptable groundwater level for the Auxiliary Building by transferring water out of the Auxiliary Building and mitigates the consequences of certain postulated flooding events. The McGuire UFSAR Section 9.5.8, Groundwater Drainage System, provides additional information concerning the McGuire Groundwater Drainage System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Groundwater Drainage System are listed in Table 3.3-29. The structural portions of the Groundwater Drainage System are addressed in Section 2.4.2.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the McGuire Groundwater Drainage System:

MCFD-1581-01.00

This flow diagram is contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Groundwater Drainage System prevents hydrostatic loads on the Reactor and Auxiliary Building substructures. The Groundwater Drainage System maintains an acceptable groundwater level for the Auxiliary Building by transferring water out of the Auxiliary Building and mitigates the consequences of certain postulated flooding events. The Catawba UFSAR Section 9.5.11, Groundwater Drainage System, provides additional information concerning the Catawba Groundwater Drainage System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Groundwater Drainage System are listed in Table 3.3-29. The structural portions of the Groundwater Drainage System are addressed in Section 2.4.2.

The following flow diagram has have been marked to indicate the license renewal evaluation boundary for the Catawba Groundwater Drainage System:

CN-1581-1.0

This flow diagram is contained in Reference [2.3.3-1].

2.3.3.22 Hydrogen Bulk Storage System

McGuire Nuclear Station – The Hydrogen Bulk Storage System supplies hydrogen to the Volume Control Tank (Chemical and Volume Control System). The Hydrogen Bulk Storage System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Hydrogen Bulk Storage System are listed in Table 3.3-30.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Hydrogen Bulk Storage System:

MCFD-1554-02.00	MCFD-1603-01.00	MCFD-2554-02.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Hydrogen Bulk Storage System supplies hydrogen to the Volume Control Tank (Chemical and Volume Control System). The Hydrogen Bulk Storage System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Hydrogen Bulk Storage System are listed in Table 3.3-30.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Hydrogen Bulk Storage System:

CN-1603-1.0	CN-2554-1.1
CN-1554-1.1	CN-1567-1.4

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.23 Instrument Air System

McGuire Nuclear Station – The Instrument Air System provides dry oil-free air for instrumentation, testing, and control air requirements. The McGuire UFSAR Section 9.3.1, Compressed Air Systems, provides additional information concerning the McGuire Instrument Air System. The mechanical components, component functions, and materials of construction for the McGuire Instrument Air System are listed in Table 3.3-31.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Instrument Air System:

MCFD-1605-01.02	MCFD-1593-01.00	MCFD-2605-01.04
MCFD-1605-01.03	MCFD-1593-01.03	MCFD-2605-01.13
MCFD-1605-01.04	MCFD-1602-01.02	MCFD-2605-01.16
MCFD-1605-01.13	MCFD-1609-04.00	MCFD-2553-02.00
MCFD-1605-01.14	MCFD-1605-03.01	MCFD-2562-02.00
MCFD-1605-01.16	MCFD-1605-02.02	MCFD-2609-04.00
MCFD-1605-01.17	MCFD-2605-01.02	MCFD-2605-03.01
MCFD-1553-02.00	MCFD-2605-01.03	MCFD-2605-02.02
MCFD-1562-02.00		

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Instrument Air System supplies clean, oil-free, dried compressed air to all air-operated instrumentation and valves for both units. The Catawba UFSAR Section 9.3.1, Compressed Air System, provides additional information concerning the Catawba Instrument Air System. The mechanical components, component functions, and materials of construction for the Catawba Instrument Air System are listed in Table 3.3-31.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Instrument Air System:

CN-1605-1.4	CN-1562-1.1	CN-2562-1.1
CN-1605-1.14	CN-2605-1.5	

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.24 Liquid Waste System

McGuire Nuclear Station – The Liquid Waste Recycle and Liquid Waste Monitor and Disposal Systems collect, segregate, and process the reactor-grade and non-reactor grade liquid wastes produced during station operation, refueling, or maintenance. Portions of the Liquid Waste Recycle System function as part of the Reactor Coolant System leakage detection systems. The McGuire UFSAR Section 11.2, Liquid Waste System, provides additional information concerning the McGuire Liquid Waste Recycle and Liquid Waste Monitor and Disposal Systems. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Liquid Waste Recycle and Liquid Waste Monitor and Disposal Systems are listed in Table 3.3-32.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Liquid Waste Recycle and Liquid Waste Monitor and Disposal Systems:

MCFD-1565-01.00	MCFD-1565-05.00	MCFD-2565-01.00
MCFD-1565-01.01	MCFD-1565-06.00	MCFD-2565-01.01
MCFD-1565-02.00	MCFD-1565-07.00	MCFD-2565-03.00
MCFD-1565-03.00	MCFD-1595-01.02	MCFD-2565-07.00
MCFD-1565-04.00		

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Liquid Radwaste System collects, segregates, and processes all radioactive and potentially radioactive liquids generated in the plant. In general, all reactor grade liquids are recycled and all non-reactor grade liquids are processed and disposed of in accordance with applicable NRC regulations. The system is designed to control and minimize releases of radioactivity to the environment. The Catawba UFSAR Section 11.2, Liquid Radwaste System, provides additional information concerning the Catawba Liquid Radwaste System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Liquid Radwaste System are listed in Table 3.3-32.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Liquid Radwaste System:

CN-1556-1.0	CN-1565-2.0	CN-2565-2.1
CN-1565-1.0	CN-1565-2.1	CN-2565-2.2
CN-1565-1.1	CN-1565-2.2	CN-2565-2.4
CN-1565-1.4	CN-1565-2.4	CN-2565-2.5
CN-1565-1.5	CN-1565-2.5	CN-2565-2.6
CN-1565-1.7	CN-1565-2.6	CN-1604-2.0
CN-1565-1.8	CN-1567-1.0	CN-2604-2.0
CN-1565-1.9	CN-2565-2.0	CN-1566-1.2

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.25 Miscellaneous Structures Ventilation System

McGuire Nuclear Station – The Turbine Building Ventilation System at McGuire performs the corresponding functions as the Miscellaneous Structures Ventilation System at Catawba. See Section 2.3.3.37 for the McGuire Turbine Building Ventilation System.

Catawba Nuclear Station – The Catawba Miscellaneous Structures Ventilation System includes the Standby Shutdown Facility (SSF) HVAC. The SSF HVAC portion of the Miscellaneous Structures Ventilation System provides the environmental controls necessary to ensure that SSF equipment is maintained operable during postulated fires and station blackout. The mechanical components subject to aging management review, their intended functions, and materials of construction for the SSF HVAC portion of the Catawba Miscellaneous Structures Ventilation System are listed in Table 3.3-33.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the SSF HVAC portion of the Catawba Miscellaneous Structures Ventilation System:

CN-1579-4.3

This flow diagram is contained in Reference [2.3.3-1].

2.3.3.26 Nitrogen System

McGuire Nuclear Station – The Nitrogen System provides a safety-related supply of nitrogen to the pneumatic actuators on the feedwater isolation valves. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Nitrogen System are listed in Table 3.3-34.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Nitrogen System:

MCFD-1602-01.02	MCFD-2602-01.00
MCFD-1554-02.00	MCFD-2554-02.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Nitrogen System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this systems are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Nitrogen System are listed in Table 3.3-34.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the Catawba Nitrogen System:

CN-1602-1.0

This flow diagram is contained in Reference [2.3.3-1].

2.3.3.27 Nuclear Sampling System

McGuire Nuclear Station – The Nuclear Sampling System provides a means of obtaining the more frequently taken samples during normal plant operation from the station's nuclear-safety related systems in a convenient, shielded, and safe environment. The system also provides a means of sampling the Reactor Coolant and Containment atmosphere following a loss-of-coolant-accident (LOCA) to monitor the reactor and determine the degree of core damage. The McGuire UFSAR Section 9.3.2, Nuclear Sampling System, provides additional information concerning the McGuire Nuclear Sampling System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Nuclear Sampling System are listed in Table 3.3-35.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Nuclear Sampling System:

MCFD-1572-01.00	MCFD-1572-02.00	MCFD-2554-02.00
MCFD-1553-02.00	MCFD-1572-03.00	MCFD-2562-02.00
MCFD-1553-01.00	MCFD-1580-01.00	MCFD-2562-02.01
MCFD-1572-01.01	MCFD-2572-01.00	MCFD-2572-02.00
MCFD-1554-02.00	MCFD-2553-02.00	MCFD-2572-03.00
MCFD-1562-02.00	MCFD-2553-01.00	MCFD-2580-01.00
MCFD-1562-02.01	MCFD-2572-01.01	

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Nuclear Sampling System provides a means of obtaining the more frequently taken samples during normal plant operation from the station's nuclear-safety related systems in a convenient, shielded, and safe environment. The system also provides a means of sampling the Reactor Coolant and Containment atmosphere following a loss-of-coolant-accident (LOCA) to monitor the reactor and determine the degree of core damage. The Catawba UFSAR Section 9.3.2, Process Sampling and Post-Accident Sampling Systems, provides additional information concerning the Catawba Nuclear Sampling System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Nuclear Sampling System are listed in Table 3.3-35.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Nuclear Sampling System:

CN-1572-01.00	CN-1572-01.02	CN-2572-01.01
CN-1553-01.00	CN-1572-01.04	CN-2554-01.01
CN-1553-01.01	CN-1580-01.00	CN-2562-01.01
CN-1572-01.01	CN-2572-01.00	CN-2572-01.02
CN-1554-01.01	CN-2553-01.00	CN-2572-01.04
CN-1562-01.01	CN-2553-01.01	CN-2580-01.00

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.28 Nuclear Service Water System

McGuire Nuclear Station – The Nuclear Service Water System provides cooling water from Lake Norman or the Standby Nuclear Service Water Pond to various safety-related and non-safety related heat exchangers. In addition, the system acts as an assured source of makeup water for various requirements and the normal supply of water for the Containment Ventilation Cooling Water System. The McGuire UFSAR Section 9.2.2, Nuclear Service Water System and Ultimate Heat Sink, provides additional information concerning the McGuire Nuclear Service Water System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Nuclear Service Water System are listed in Table 3.3-36.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Nuclear Service Water System:

MCFD-1570-01.01	MCFD-1592-01.01	MCFD-2574-03.00
MCFD-1573-01.01	MCFD-1604-03.00	MCFD-2574-03.01
MCFD-1574-01.00	MCFD-1609-01.00	MCFD-2574-04.00
MCFD-1574-01.01	MCFD-1609-01.01	MCFD-2592-01.01
MCFD-1574-02.00	MCFD-2570-01.01	MCFD-2604-03.00
MCFD-1574-02.01	MCFD-2573-01.01	MCFD-2604-03.01
MCFD-1574-03.00	MCFD-2574-01.01	MCFD-2609-01.00
MCFD-1574-03.01	MCFD-2574-02.00	MCFD-2609-01.01
MCFD-1574-04.00	MCFD-2574-02.01	MCFD-1604-01.02
MCFD-1581-01.00		

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Nuclear Service Water System, along with Lake Wylie and the Standby Nuclear Service Water Pond, provides the ultimate heat sink for various safety-related heat loads during normal operation and design basis events. The Nuclear Service Water System also supplies emergency makeup water to various safety-related systems during postulated design basis events, water for fire protection hose stations in the diesel buildings and Nuclear Service Water Pumphouse, and cooling flow and flush water for non-QA heat loads and functions during normal operation. The Catawba UFSAR Section 9.2.1, Nuclear Service Water System, provides additional information concerning the Catawba Nuclear Service Water System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Nuclear Service Water System are listed in Table 3.3-37.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Nuclear Service Water System:

CN-1565-1.3	CN-1574-1.4	CN-2573-1.1
CN-1565-1.10	CN-1574-1.5	CN-2573-2.2
CN-1570-1.0	CN-1574-2.0	CN-2573-2.3
CN-1573-1.1	CN-1574-2.1	CN-2574-2.0
CN-1573-2.2	CN-1574-2.2	CN-2574-2.1
CN-1573-2.3	CN-1574-2.4	CN-2574-2.2
CN-1574-1.0	CN-1574-2.5	CN-2574-2.4
CN-1574-1.1	CN-1574-2.8	CN-2574-2.5
CN-1574-1.2	CN-2570-1.0	CN-2574-2.7

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.29 Nuclear Service Water Pump Structure Ventilation System

McGuire Nuclear Station – No system corresponding to the Catawba Nuclear Service Water Pump Structure Ventilation System exists at McGuire. McGuire has no Nuclear Service Water Pump Structure.

Catawba Nuclear Station – The Nuclear Service Water Pump Structure Ventilation System creates and maintains a suitable environmental temperature for the operation of equipment located in the Nuclear Service Water Pump Structure. The Catawba UFSAR Section 9.4.8, Nuclear Service Water Pump Structure Ventilation System, provides additional information concerning the Catawba Nuclear Service Water Pump Structure Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Nuclear Service Water Pump Structure Ventilation System are listed in Table 3.3-38.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the Catawba Nuclear Service Water Pump Structure Ventilation System:

CN-1557-2.0

This flow diagram is contained in Reference [2.3.3-1].

2.3.3.30 Nuclear Solid Waste Disposal System

McGuire Nuclear Station – The Nuclear Solid Waste Disposal System is relied upon to contain solid radioactive waste materials as they are produced in the station. The McGuire UFSAR Section 11.5, Nuclear Solid Waste Disposal System, provides additional information concerning the McGuire Nuclear Solid Waste Disposal System. The mechanical components, component functions, and materials of construction for the McGuire Nuclear Solid Waste Disposal System are listed in Table 3.3-39.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the McGuire Nuclear Solid Waste Disposal System:

MCFD-1566-01.00

This flow diagram is contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Solid Radwaste System provides capacity to contain and store radioactive waste materials as they are produced in the station and prepares the waste for eventual shipment to a licensed offsite disposal facility. The Solid Radwaste System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components, component functions, and materials of construction for the Catawba Solid Radwaste System are listed in Table 3.3-39.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the Catawba Solid Radwaste System:

CN-1566-1.2

This flow diagram is contained in Reference [2.3.3-1].

2.3.3.31 Reactor Coolant Pump Motor Oil Collection Sub-System

McGuire Nuclear Station – Each reactor coolant pump motor at McGuire Nuclear Station is equipped with an oil collection system that contains any oil leakage that meets the requirements of 10 CFR 50, Appendix R, Section III.O. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Reactor Coolant Pump Motor Oil Collection Sub-System are listed in Table 3.3-40.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Reactor Coolant Pump Motor Oil Collection Sub-System:

MCFD-1553-04.00 MCFD-2553-04.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – Each reactor coolant pump motor at Catawba Nuclear Station is equipped with an oil collection system that contains any oil leakage that meets the requirements of 10 CFR 50, Appendix R, Section III.O. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Reactor Coolant Pump Motor Oil Collection Sub-System are listed in Table 3.3-40.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Reactor Coolant Pump Motor Oil Collection Sub-System:

CN-1553-1.3 CN-2553-1.3

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.32 Reactor Coolant System (Non-Class 1 Components)

McGuire Nuclear Station – The non-Class 1 portions of the Reactor Coolant System (excluding the RCP motor oil collection sub-system) are relied upon to provide and maintain Containment isolation and closure and maintain system pressure boundary integrity. The reactor vessel leak-off line is included within this set of components and is relied upon only in the event the reactor vessel flange inner seal leaks. The mechanical components subject to aging management review, their intended functions, and materials of construction for the non-Class 1 portions of the McGuire Reactor Coolant System are listed in Table 3.3-41.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the non-Class 1 portions of the McGuire Reactor Coolant System:

MCFD-1553-01.00	MCFD-1562-03.01	MCFD-2554-03.01
MCFD-1553-02.00	MCFD-1563-01.00	MCFD-2561-01.00
MCFD-1553-02.01	MCFD-1567-02.01	MCFD-2562-03.00
MCFD-1554-03.01	MCFD-2553-01.00	MCFD-2562-03.01
MCFD-1561-01.00	MCFD-2553-02.00	MCFD-2563-01.00
MCFD-1562-03.00	MCFD-2553-02.01	

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The non-Class 1 portions of the Reactor Coolant System (excluding the RCP motor oil collection sub-system) are relied upon to provide and maintain Containment isolation and closure and maintain system pressure boundary integrity. The reactor vessel leak-off line is included within this set of components and is relied upon only in the event the reactor vessel flange inner seal leaks. The mechanical components subject to aging management review, their intended functions, and materials of construction for the non-Class 1 portions of the Catawba Reactor Coolant System are listed in Table 3.3-41.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the non-Class 1 portions of the Catawba Reactor Coolant System:

CN-1553-1.0	CN-1567-1.1	CN-2553-1.1
CN-1553-1.1	CN-2553-1.0	CN-2553-1.2
CN-1553-1.2		

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.33 Recirculated Cooling Water System

McGuire Nuclear Station – No portion of the McGuire Recirculated Cooling Water System is within the scope of license renewal. Only the Duke Class F portions of the Recirculated Cooling Water System are in scope at Catawba. McGuire has no Class F components in the Recirculated Cooling Water System.

Catawba Nuclear Station – The Recirculated Cooling Water System is a closed cooling system that delivers clean, rust-inhibited cooling water of a regulated temperature to various equipment in the Turbine Buildings, Auxiliary Building, and Service Building. The Recirculated Cooling Water System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Recirculated Cooling Water System are listed in Table 3.3-42.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the Catawba Recirculated Cooling Water System:

CN-1600-1.1

This flow diagram is contained in Reference [2.3.3-1].

2.3.3.34 Spent Fuel Cooling System

McGuire Nuclear Station – The Spent Fuel Cooling System removes heat from the spent fuel pool and maintains the purity and optical clarity of the pool water for fuel handling operations. The purification loop provides an alternate means for removing impurities from either the refueling canal/transfer canal water during refueling or the refueling water storage tank water following refueling. The fuel pool water also serves as a source of makeup water to the Reactor Coolant System during a Standby Shutdown System event. The McGuire UFSAR Section 9.1.3, Spent Fuel Cooling and Purification, provides additional information concerning the McGuire Spent Fuel Cooling System. The mechanical components, component functions, and materials of construction for the McGuire Spent Fuel Cooling System are listed in Table 3.3-43.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Spent Fuel Cooling System:

MCFD-1570-01.00	MCFD-1571-01.00	MCFD-2570-01.01
MCFD-1570-01.01	MCFD-2570-01.00	MCFD-2571-01.00

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Spent Fuel Cooling System, in conjunction with the Component Cooling Water system and Nuclear Service Water system, is designed to remove heat from the Spent Fuel Pool and maintain purity and optical clarity of the pool water during fuel handling operations. The purification loop provides an alternate means for removing impurities from either the refueling cavity/transfer canal water during refueling or the Refueling Water Storage Tank water following refueling. The Catawba UFSAR Section 9.1.3, Spent Fuel Cooling and Purification, provides additional information concerning the Catawba Spent Fuel Cooling System. The mechanical components, component functions, and materials of construction for the Catawba Spent Fuel Cooling System are listed in Table 3.3-43.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Spent Fuel Cooling System:

CN-1554-1.8	CN-1570-1.1	CN-2570-1.0
CN-1570-1.0	CN-2554-1.8	CN-2570-1.1

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.35 Standby Shutdown Diesel

McGuire Nuclear Station – The Standby Shutdown Diesel System provides an alternate and independent means of achieving and maintaining a Hot Standby condition for one or both units following a postulated fire event. The diesel provides power to the standby shutdown facility required components, instrumentation, and controls for a period of up to 72 hours. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Standby Shutdown Diesel System are listed in Table 3.3-44.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Standby Shutdown Diesel System:

MCFD-1560-01.00	MCFD-1560-02.00	MC-1614-4

These flow diagrams are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Standby Shutdown Diesel System provides an alternate and independent means of achieving and maintaining a Hot Standby condition for one or both units following a postulated fire event. The diesel provides power to the standby shutdown facility required components, instrumentation, and controls for a period of up to 72 hours. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Standby Shutdown Diesel System are listed in Table 3.3-44.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Standby Shutdown Diesel System:

CN-1560-1.0	CN-1560-2.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.36 Turbine Building Sump Pump System

McGuire Nuclear Station – No portion of the McGuire Turbine Building Sump Pump System is within the scope of license renewal. Only the Duke Class F portions of the Turbine Building Sump Pump System are in scope at Catawba. McGuire has no Class F components in the Turbine Building Sump Pump System.

Catawba Nuclear Station – The Turbine Building Sump Pump System serves as a collection point for the contents of Liquid Radwaste System sumps when the contents of the sumps contain less than predetermined levels of radiation, as sensed by radiation monitors in the discharge lines. The Turbine Building Sump Pump System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this systems are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Turbine Building Sump Pump System are listed in Table 3.3-45.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Turbine Building Sump Pump System:

CN-1604-2.0 CN-2604-2.0

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.37 Turbine Building Ventilation System

McGuire Nuclear Station – The Turbine Building Ventilation System includes the HVAC system in the Standby Shutdown Facility (SSF), of which a portion is entitled Standby Shutdown Facility HVAC System. The SSF HVAC portion of the Turbine Building Ventilation System provides the heating, ventilation and air conditioning requirements for the SSF and consists of air conditioning and ventilation subsystems. The McGuire UFSAR Section 9.4.4, Turbine Building, provides additional information concerning the SSF HVAC portion of the McGuire Turbine Building Ventilation System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the SSF HVAC portion of the McGuire Turbine Building Ventilation System are listed in Table 3.3-46.

The following flow diagram has been marked to indicate the license renewal evaluation boundary for the SSF HVAC portion of the McGuire Turbine Building Ventilation System:

MC-1614-4

This flow diagram is contained in Reference [2.3.3-1].

Catawba Nuclear Station – No portion of the Catawba Turbine Building Ventilation System is within the scope of license renewal. The Catawba Miscellaneous Structures Ventilation System that provides SSF HVAC is addressed in Section 2.3.3.25.

2.3.3.38 Waste Gas System

McGuire Nuclear Station – The Waste Gas System removes fission gases from radioactive contaminated fluids and contains these gases in holdup tanks indefinitely. Storage and subsequent decay of these gases eliminates the need for regularly scheduled discharge of these radioactive gases from the system into the atmosphere during normal plant operation. The McGuire UFSAR Section 11.3, Waste Gas System, provides additional information concerning the McGuire Waste Gas System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Waste Gas System are listed in Table 3.3-47.

The following is a list of the flow diagrams and instrument details that have been marked to indicate the license renewal evaluation boundary for the McGuire Waste Gas System:

MCFD-1567-01.00	MCFD-1567-04.00	MCFD-2554-03.00
MCFD-1567-02.00	MCFD-1554-02.00	MCFD-1556-02.01
MCFD-1567-02.01	MCFD-1554-03.00	MCFD-1565-06.00
MCFD-1567-03.00	MCFD-2554-02.00	MCID-1499-WG.28
MCFD-1567-03.01		

These flow diagrams and instrument details are contained in Reference [2.3.3-1].

Catawba Nuclear Station – The Waste Gas System removes fission product gases from radioactive fluids and contains these gases for a time sufficient to allow ample decay of the nuclides prior to release in accordance with applicable NRC regulations. The system is designed to control and minimize releases of radioactive effluent to the environment by reducing the fission product gas concentration in the reactor coolant which may escape during maintenance operations or from equipment leaks. The Catawba UFSAR Section 11.3, Waste Gas System, provides additional information concerning the Catawba Waste Gas System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Waste Gas System are listed in Table 3.3-47.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Waste Gas System:

CN-1567-1.0	CN-1567-1.4	CN-1556-1.2
CN-1567-1.1	CN-1554-1.1	CN-1556-1.3
CN-1567-1.2	CN-2554-1.1	CN-1565-1.5
CN-1567-1.3		

These flow diagrams are contained in Reference [2.3.3-1].

2.3.3.39 References for Section 2.3.3

2.3.3-1. M. S. Tuckman (Duke) letter dated June 13, 2001 to Document Control Desk (NRC), *License Renewal Evaluation Boundary Drawings.* McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2, Docket Nos. 50-369, 50-370, 50-413, and 50-414.

2.3.4 STEAM AND POWER CONVERSION SYSTEMS

The methodology to identify the mechanical systems within the scope of license renewal is described in Section 2.1.1 of this Application. Section 2.1.2.1 contains a description of the methodology to identify the mechanical components that are subject to aging management review.

Systems, structures and components are located in this section consistent with their locations in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (July 1981) except where the internal operating environment or function suggests a more appropriate location within the Application. In these situations, the system, structure or component has been located in the more appropriate section. For example, the Condenser Circulating Water System which is normally described within Chapter 10 of NUREG-0800, described in Section 2.3.3, Auxiliary Systems, of this Application because its internal operating environment is raw water.

The following mechanical systems are described in Section 2.3.4, Steam and Power Conversion Systems, in the sections as indicated:

- Auxiliary Feedwater System (Section 2.3.4.1)
- Auxiliary Steam System (Section 2.3.4.2)
- Condensate System (Section 2.3.4.3)
- Condensate Storage System (Section 2.3.4.4)
- Feedwater System (Section 2.3.4.5)
- Feedwater Pump Turbine Exhaust System (Section 2.3.4.6)
- Feedwater Pump Turbine Hydraulic Oil System (Section 2.3.4.7)
- Main Steam Auxiliary Equipment System (Section 2.3.4.9)
- Main Steam System (Section 2.3.4.8)
- Main Steam Supply to Auxiliary Equipment System (Section 2.3.4.9)
- Main Steam Vent to Atmosphere System (Section 2.3.4.10)
- Main Turbine Hydraulic Oil System (Section 2.3.4.11)
- Main Turbine Lube Oil and Purification System (Section 2.3.4.12)
- Turbine Exhaust System (Section 2.3.4.6)

2.3.4.1 Auxiliary Feedwater System

McGuire Nuclear Station – The Auxiliary Feedwater System is a nuclear safety-related system which serves as a backup to the Feedwater System to ensure the safety of the plant and protection of equipment. The Auxiliary Feedwater System is essential to prevent an unacceptable decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and to establish stable hot standby conditions. The Auxiliary Feedwater System can be used during an emergency as well as during normal startup and shutdown operations. The McGuire UFSAR Section 10.4.10, Auxiliary Feedwater System, provides additional information concerning the McGuire Auxiliary Feedwater System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Auxiliary Feedwater System are listed in Table 3.4-1.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Auxiliary Feedwater System:

MCFD-1574-02.00	MCFD-1592-01.01	MCFD-2591-01.01
MCFD-1574-03.00	MCFD-1617-01.00	MCFD-2592-01.00
MCFD-1584-01.00	MCFD-2574-02.00	MCFD-2592-01.01
MCFD-1591-01.01	MCFD-2574-03.00	MCFD-2617-01.00
MCFD-1592-01.00	MCFD-2584-01.00	

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Auxiliary Feedwater System is a nuclear safety related system which serves as a backup to the Feedwater System to ensure the safety of the plant and protection of equipment. The Auxiliary Feedwater System is essential to prevent an unacceptable decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and to establish stable hot standby conditions. The Auxiliary Feedwater System can be used during an emergency as well as during normal startup and shutdown operations. The Catawba UFSAR Section 10.4.9, Auxiliary Feedwater System, provides additional information concerning the Catawba Auxiliary Feedwater System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Auxiliary Feedwater System are listed in Table 3.4-1.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Auxiliary Feedwater System:

CN-1592-1.0	CN-1584-1.0	CN-2573-2.3
CN-1592-1.1	CN-1591-1.1	CN-2574-2.1
CN-1565-2.2	CN-2592-1.0	CN-2574-2.5
CN-1573-2.2	CN-2592-1.1	CN-2584-1.0
CN-1573-2.3	CN-2565-2.2	CN-2591-1.1
CN-1574-2.1	CN-2573-2.2	CN-1593-1.2
CN-1574-2.5		

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.2 Auxiliary Steam System

McGuire Nuclear Station – The Auxiliary Steam System provides steam to various plant equipment as required during all modes of plant operation, including condensate cleanup, startup, normal operation, and shutdown. The Auxiliary Steam System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Auxiliary Steam System are listed in Table 3.4-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Auxiliary Steam System:

MCFD-1595-1.0	MCFD-2595-1.0	MCFD-2593-1.1
MCFD-1595-1.2	MCFD-1593-1.2	MCFD-1554-5.0

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Auxiliary Steam System provides steam to various plant equipment as required during all modes of plant operation, including condensate cleanup, startup, normal operation, and shutdown. The Auxiliary Steam System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Auxiliary Steam System are listed in Table 3.4-2.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Auxiliary Steam System:

CN-1595-1.0	CN-2595-1.0	CN-2593-1.1
CN-1595-1.2	CN-1593-1.1	

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.3 Condensate System

McGuire Nuclear Station – No portion of the McGuire Condensate System is within the scope of license renewal. Only the Duke Class F portions of the Condensate System are in scope at Catawba. McGuire has no Class F components in the Condensate System.

Catawba Nuclear Station – The Condensate System provides water to various plant equipment as required during all modes of plant operation, including condensate cleanup, startup, normal operation, and shutdown. The Condensate System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Condensate System are listed in Table 3.4-3.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Condensate System:

CN-1590-1.8 CN-2590-1.8

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.4 Condensate Storage System

McGuire Nuclear Station – No portion of the McGuire Condensate Storage System is within the scope of license renewal. Only the Duke Class F portions of the Condensate Storage System are in scope at Catawba. McGuire has no Class F components in the Condensate Storage System.

Catawba Nuclear Station – The Condensate Storage System provides a source of water for various plant equipment as required during all modes of plant operation, including condensate cleanup, startup, normal operation, and shutdown. The Condensate Storage System is a non-safety system whose postulated failure could prevent satisfactory accomplishment of certain safety-related functions. To preclude these postulated failures, portions of this system are seismically designed (i.e., Duke Class F). All components within the seismically designed piping boundaries of this system are within the scope of license renewal per §54.4(a)(2). The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Condensate Storage System are listed in Table 3.4-4.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Condensate Storage System:

CN-1590-2.1 CN-2590-2.1

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.5 Feedwater System

McGuire Nuclear Station – The Feedwater System takes treated Condensate System water, heats it further to improve the plant's thermal cycle efficiency, and delivers it at the required flow rate, pressure and temperature to the steam generators. The Feedwater System is designed to maintain proper vessel water levels with respect to reactor power output and turbine steam requirements. The McGuire UFSAR Section 10.4.7, Condensate and Feedwater System, provides additional information concerning the McGuire Feedwater System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Feedwater System are listed in Table 3.4-5.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Feedwater System:

MCFD-1591-01.01	MCFD-2591-01.01

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Feedwater System takes treated Condensate System water, heats it further to improve the plant's thermal cycle efficiency, and delivers it at the required flow rate, pressure and temperature to the steam generators. The Feedwater System is designed to maintain proper vessel water levels with respect to reactor power output and turbine steam requirements. The Catawba UFSAR Section 10.4.7, Condensate and Feedwater System, provides additional information concerning the Catawba Feedwater System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Feedwater System are listed in Table 3.4-5.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Feedwater System:

CN-1591-1.1	CN-2591-1.1
CN-1592-1.1	CN-2592-1.1

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.6 Feedwater Pump Turbine Exhaust System

McGuire Nuclear Station – The Turbine Exhaust System exhausts steam from the turbine-driven auxiliary feedwater pump turbine and the feedwater pump turbines. The Turbine Exhaust System is not described in the McGuire UFSAR. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Turbine Exhaust System are listed in Table 3.4-6.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Turbine Exhaust System:

MCFD-1593-01.02	MCFD-1612-04.00	MCFD-2593-02.00
MCFD-1593-02.00	MCFD-2593-01.02	MCFD-2612-04.00

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Feedwater Pump Turbine Exhaust System provides a flowpath for the exhaust steam from the turbine-driven auxiliary feedwater pump turbine. The steam to the turbine-driven auxiliary feedwater pump turbine is provided by the Main Steam System. The Catawba UFSAR Sections 10.3, Main Steam System, provides additional information concerning the design and operation of these systems. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Feedwater Pump Turbine Exhaust System are listed in Table 3.4-6.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Feedwater Pump Turbine Exhaust System:

CN-1593-1.2	CN-2593-1.2

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.7 Feedwater Pump Turbine Hydraulic Oil System

McGuire Nuclear Station – The Feedwater Pump Turbine Hydraulic Oil System provides emergency trip to the feedwater pump turbine steam valves and overspeed exercisers for ATWS mitigation. The McGuire Feedwater Pump Turbine Hydraulic Oil System is not described in the McGuire UFSAR. The turbine trip signal causes pressure to be bled off the hydraulic system causing the stop and governor valves to close. The components required to meet these functions are either active components or are passive components whose failure will not prevent the desired action from occurring. Failure of the pressure boundary of the valve bodies or piping will create a loss of hydraulic pressure causing the stop and governor valves to close which would not prevent the safety function. Therefore, the components are in scope, but no aging management review is required.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Feedwater Pump Turbine Hydraulic Oil System:

MCFD-1616-01.00	MCFD-2616-01.00
MCFD-1616-02.00	MCFD-2616-02.00

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Feedwater Pump Turbine Hydraulic Oil System provides emergency trip to the feedwater pump turbine steam valves and overspeed exercisers for ATWS mitigation. The Catawba Feedwater Pump Turbine Hydraulic Oil System is not described in the Catawba UFSAR. The turbine trip signal causes pressure to be bled off the hydraulic system causing the stop and governor valves to close. The components required to meet these functions are either active components or are passive components whose failure will not prevent the desired action from occurring. Failure of the pressure boundary of the valve bodies or piping will create a loss of hydraulic pressure causing the stop and governor valves to close which would not prevent the safety function. Therefore, the components are in scope, but no aging management review is required.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Feedwater Pump Turbine Hydraulic Oil System:

CN-1616-1.0	CN-2616-1.0
CN1616-1.1	CN-2616-1.1
CN-1616-2.0	CN-2616-2.0
CN-1616-2.1	CN-2616-2.1

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.8 Main Steam System

McGuire Nuclear Station – The Main Steam System dissipates heat from the Reactor Coolant System, provides main steam overpressure protection, minimizes positive reactivity effects associated with a main steam line rupture, minimizes the containment temperature increase associated with a main steam line rupture within containment, and provides steam to the turbine driven auxiliary feedwater pump, as needed. The McGuire UFSAR Section 10.3, Main Steam Supply System, provides additional information concerning the McGuire Main Steam System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Main Steam System are listed in Table 3.4-7.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Main Steam System:

MCFD-1593-01.00	MCFD-1593-02.00	MCFD-2593-01.03
MCFD-1593-01.01	MCFD-2593-01.00	MCFD-2593-02.00
MCFD-1593-01.02	MCFD-2593-01.01	MCFD-2612-04.00
MCFD-1593-01.03	MCFD-2593-01.02	MCFD-1591-01.01

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Main Steam System dissipates heat from the Reactor Coolant System, provides main steam overpressure protection, minimizes positive reactivity effects associated with a main steam line rupture, minimizes the containment temperature increase associated with a main steam line rupture within containment, and provides steam to the turbine driven auxiliary feedwater pump, as needed. The Catawba UFSAR Section 10.3, Main Steam Supply System, provides additional information concerning the Catawba Main Steam System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Main Steam System are listed in Table 3.4-7.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Main Steam System:

CN-1593-1.0	CN-1591-1.1	CN-2593-1.2
CN-1593-1.1	CN-1565-2.2	CN-2593-1.3
CN-1593-1.2	CN-1592-1.0	CN-2593-1.7
CN-1593-1.3	CN-2593-1.0	CN-2591-1.1
CN-1593-1.7	CN-2593-1.1	CN-2565-2.2

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.9 Main Steam Supply to Auxiliary Equipment

McGuire Nuclear Station – The Main Steam Supply to Auxiliary Equipment transfers steam to the turbine driven auxiliary feedwater pump turbine, so that the design bases of the Auxiliary Feedwater System can be met. The McGuire UFSAR Section 10.3, Main Steam Supply System, provides additional information concerning the McGuire Main Steam Supply to Auxiliary Equipment. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Main Steam Supply to Auxiliary Equipment System are listed in Table 3.4-8.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Main Steam Supply to Auxiliary Equipment System:

MCFD-1593-01.02 MCFD-2593-01.02

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Main Steam Auxiliary Equipment transfers steam to the turbine driven auxiliary feedwater pump turbine, so that the design bases of the Auxiliary Feedwater System can be met. The Catawba UFSAR, Section 10.3, Main Steam Supply System, provides additional information concerning the Catawba Main Steam Auxiliary Equipment. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Main Steam Auxiliary Equipment System are listed in Table 3.4-8.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Main Steam Auxiliary Equipment System:

CN-1593-1.1 CN-2593-1.1

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.10 Main Steam Vent to Atmosphere System

McGuire Nuclear Station – The Main Steam Vent to Atmosphere System dissipates heat from the Reactor Coolant System, provides main steam overpressure protection, minimizes positive reactivity effects associated with a main steam line rupture, and minimizes the containment temperature increase associated with a main steam line rupture within containment. The McGuire UFSAR Section 10.3, Main Steam Supply System, provides additional information concerning the McGuire Main Steam Vent to Atmosphere System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the McGuire Main Steam Vent to Atmosphere System are listed in Table 3.4-9.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Main Steam Vent to Atmosphere System:

MCFD-1593-01.00	MCFD-1593-01.03	MCFD-2593-01.01
MCFD-1593-01.01	MCFD-2593-01.00	MCFD-2593-01.03

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Main Steam Vent to Atmosphere System dissipates heat from the Reactor Coolant System, provides main steam overpressure protection, minimizes positive reactivity effects associated with a main steam line rupture, and minimizes the containment temperature increase associated with a main steam line rupture within containment. The Catawba UFSAR Section 10.3, Main Steam Supply System, provides additional information concerning the Catawba Main Steam Vent to Atmosphere System. The mechanical components subject to aging management review, their intended functions, and materials of construction for the Catawba Main Steam Vent to Atmosphere System are listed in Table 3.4-9.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Main Steam Vent to Atmosphere System:

CN-1593-1.0	CN-2593-1.0

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.11 Main Turbine Hydraulic Oil System

McGuire Nuclear Station – The Main Turbine Hydraulic Oil System provides a means to trip the main turbine to mitigate the plant response to an ATWS event. The Main Turbine Hydraulic Oil System is not described in the McGuire UFSAR. The components in the Main Turbine Hydraulic Oil System are required to maintain pressure boundary integrity for normal system operation. However, an operational loss of pressure in the hydraulic oil system, or a failure of the pressure boundary of the components highlighted on the attached mechanical system flow diagrams, will produce a turbine trip signal which is a failure in the safe direction. Because a turbine trip signal is the system intended function, there are no component intended functions applicable to the components highlighted on the mechanical system flow diagrams. Therefore, no aging management review is required.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Main Turbine Hydraulic Oil System:

MCFD-1615-01.00
MCFD-1615-02.00
MCFD-1615-02.01
MCFD-1607-01.00
MCFD-1607-01.01

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Main Turbine Hydraulic Oil System provides a means to trip the main turbine to mitigate the plant response to an ATWS event. The Main Turbine Hydraulic Oil System is not described in the Catawba UFSAR. The components in the Main Turbine Hydraulic Oil System are required to maintain pressure boundary integrity for normal system operation. However, an operational loss of pressure in the hydraulic oil system, or a failure of the pressure boundary of the components highlighted on the attached mechanical system flow diagrams, will produce a turbine trip signal which is a failure in the safe direction. Because a turbine trip signal is the system intended function, there are no component intended functions applicable to the components highlighted on the mechanical system flow diagrams. Therefore, no aging management review is required.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Main Turbine Hydraulic Oil System:

CN-1615-1.1	CN-1607-1.0	CN-2615-1.3
CN-1615-1.2	CN-2615-1.1	CN-2607-1.0
CN-1615-1.3	CN-2615-1.2	

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.12 Main Turbine Lube Oil and Purification System

McGuire Nuclear Station – The Main Turbine Lube Oil and Purification System provides a means to trip the main turbine to mitigate the plant response to an ATWS event. The Main Turbine Lube Oil and Purification System is not described in the McGuire UFSAR. The components in the Main Turbine Lube Oil and Purification System are required to maintain pressure boundary integrity for normal system operation. However, an operational loss of pressure in the hydraulic oil system, or a failure of the pressure boundary of the components highlighted on the attached mechanical system flow diagrams, will produce a turbine trip signal which is a failure in the safe direction. Because a turbine trip signal is the system intended function, there are no component intended functions applicable to the components highlighted on the mechanical system flow diagrams. Therefore, no aging management review is required.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the McGuire Main Turbine Lube Oil and Purification System:

MCFD-1607-01.00	MCFD-2607-01.00
MCFD-1607-01.01	MCFD-2607-01.01

These flow diagrams are contained in Reference [2.3.4-1].

Catawba Nuclear Station – The Main Turbine Lube Oil and Purification System provides a means to trip the main turbine to mitigate the plant response to an ATWS event. The Main Turbine Lube Oil and Purification System is not described in the Catawba UFSAR. The components in the Main Turbine Lube Oil and Purification System are required to maintain pressure boundary integrity for normal system operation. However, an operational loss of pressure in the hydraulic oil system, or a failure of the pressure boundary of the components highlighted on the attached mechanical system flow diagrams, will produce a turbine trip signal which is a failure in the safe direction. Because a turbine trip signal is the system intended function, there are no component intended functions applicable to the components highlighted on the mechanical system flow diagrams. Therefore, no aging management review is required.

The following is a list of the flow diagrams that have been marked to indicate the license renewal evaluation boundary for the Catawba Main Turbine Lube Oil and Purification System:

CN-1607-1.0	CN-2607-1.0

These flow diagrams are contained in Reference [2.3.4-1].

2.3.4.13 References for Section 2.3.4

2.3.4-1. M. S. Tuckman (Duke) letter dated June 13, 2001 to Document Control Desk (NRC), *License Renewal Evaluation Boundary Drawings.* McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2, Docket Nos. 50-369, 50-370, 50-413, and 50-414.

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2.4 SCOPING AND SCREENING RESULTS: STRUCTURES

Note: The structure and structural component descriptions are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

The methodology to identify structures within the scope of license renewal is described in Section 2.1.1 of this Application. Section 2.1.2.2 contains a description of the methodology to identify the structural components that are subject to aging management review.

Structures and structural components are located in this section consistent with Section 2.4 of the draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (August 2000). Structures that are attached to or contained within a larger structure are reviewed with the larger structure as noted below. The following structures are described in Section 2.4, Structures and Structural Components in the section as indicated:

- Auxiliary Building (including Control Building, Diesel Generator Buildings, Fuel Buildings, Groundwater Drainage System, Main Steam Doghouses, and UHI Tank Building (Catawba only)) (Section 2.4.2.1)
- Condenser Cooling Water Intake Structure – McGuire Nuclear Station fire pump rooms only (Section 2.4.2.2)
- Nuclear Service Water Structures (Section 2.4.2.3)
- Reactor Building Units 1 and 2 (including Concrete Shield Building, Steel Containment, Reactor Building Internal Structures) (Section 2.4.1)
- Standby Nuclear Service Water Pond Dam (Section 2.4.2.4)
- Standby Shutdown Facility (Section 2.4.2.5)
- Turbine Buildings (including Service Building) (Section 2.4.2.6)
- Unit Vent Stack (Section 2.4.2.7)
- Yard Structures (including Low Pressure Service Water Intake Structure (Catawba only), Refueling Water Storage Tank foundation and missile wall, Reactor Makeup Water Storage Tank foundations (McGuire only), trenches, and Yard Drainage System (Catawba only)) (Section 2.4.2.8)

Equipment and component supports that are located in all of the above structures and that are within the scope of license renewal are described in Section 2.4.3.

2.4.1 REACTOR BUILDINGS

Note: The Reactor Building description is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

2.4.1.1 Concrete Shield Building

The Concrete Shield Building (or Reactor Building) structure is part of the containment system that is designed to ensure that an acceptable upper limit of leakage of radioactive material is not exceeded under design basis events. The Reactor Building is a seismic Category I structure at McGuire and Catawba. Each Reactor Building is a reinforced concrete structure composed of a right cylinder with a shallow dome and flat circular foundation. The Reactor Building houses the Steel Containment Vessel and is designed to provide biological shielding as well as missile protection for the Steel Containment Vessel. An annulus space is provided between the Reactor Building shell and the Steel Containment Vessel for control of containment external temperatures and pressures. The annulus space also provides a controlled air volume for filtering and access to penetrations for testing and inspection.

Additional information concerning the Concrete Shield Building at McGuire is contained in McGuire UFSAR Section 3.8.1, Concrete Containment.

Additional information concerning the Concrete Shield Building at Catawba is contained in Catawba UFSAR Section 3.8.1, Concrete Containment.

The structural components, component functions, and materials of construction for the Concrete Shield Buildings are listed in Table 3.5-1.

2.4.1.2 Steel Containment

The Steel Containment (or Containment) surrounds the Reactor Coolant System and functions as the primary containment. The Steel Containment is a freestanding welded seismic Category I steel structure with a vertical cylinder, hemispherical dome, and a flat base. The Steel Containment shell is anchored to the Concrete Shield Building foundation by means of anchor bolts around the circumference of the cylinder base. The base of the Containment is a liner plate encased in concrete and anchored to the Concrete Shield Building foundation.

The steel components included within the Steel Containment are as follows:

- Steel Containment Vessel
- Equipment Hatch
- Personnel Air Locks
- Fuel Transfer Penetration
- Mechanical Penetrations

- Electrical Penetrations

Additional information concerning the Steel Containment at McGuire is contained in McGuire UFSAR Section 3.8.2, Steel Containment System.

Additional information concerning the Steel Containment at Catawba is contained in Catawba UFSAR Section 3.8.2, Steel Containment.

The structural components, component functions, and materials of construction for the Steel Containments are listed in Table 3.5-1.

2.4.1.3 Reactor Building Internal Structures

The Reactor Building Internal Structures consist of a variety of reinforced concrete and structural steel structures. The internal structures enclose the Reactor Coolant System and provide biological shielding and pressure boundaries for the lower, intermediate, and upper volumes of the containment interior. These structures also provide support and restraint for all major equipment, components, and systems located within the Reactor Building. The internal structures are supported on the concrete Reactor Building foundation. The Reactor Building Internal Structures include:

- Accumulator floor
- Base slab
- Containment recirculation sump screen assembly
- Crane wall
- CRDM missile shield
- Operating floor
- Pressure seals and gaskets
- Pressurizer enclosures
- Reactor Vessel cavity wall
- Refueling canal
- Steam Generator enclosures
- Ice condenser structural components

Note that Nuclear Steam Supply System (NSSS) component supports are described in Section 2.4.3, Component Supports.

Additional information concerning the Reactor Building Internal Structures at McGuire is contained in McGuire UFSAR Section 3.8.3, Concrete and Structural Steel Internal Structures of the Steel Containment. Additional information concerning the Ice Condenser at McGuire is contained in McGuire UFSAR Section 6.2.2, Ice Condenser System.

Additional information concerning the Reactor Building Internal Structures at Catawba is contained in Catawba UFSAR Section 3.8.3, Concrete and Structural Steel Internal Structures of the Steel Containment. Additional information concerning the Ice Condenser at Catawba is contained in Catawba UFSAR Section 6.7, Ice Condenser System.

The structural components, component functions, and materials of construction for the Reactor Building Internal structures are listed in Table 3.5-1.

2.4.2 OTHER STRUCTURES

Note: The structure descriptions are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

2.4.2.1 Auxiliary Buildings

The Auxiliary Buildings are a collection of structures that house equipment necessary for the safe operation of the plant. Each nuclear station has one Auxiliary Building that is shared by both reactor units. The Auxiliary Building is a seismic Category I reinforced concrete structure. The Auxiliary Building is integrally connected with the Spent Fuel Building and the Inside Main Steam Doghouse. The Auxiliary Building is connected with the Diesel Generator Building by cable tunnels. Additional information concerning the Auxiliary Buildings at McGuire is contained in McGuire UFSAR Section 3.8.4.1, Auxiliary Building. Additional information concerning the Auxiliary Buildings at Catawba is contained in Catawba UFSAR Section 3.8.4, Other Seismic Category I Structures.

The Control Building is a part of the reinforced concrete Auxiliary Building. The Control Building is a seismic Category I structure which houses the control room, battery room, and the cable room.

The Diesel Generator Buildings are free-standing reinforced concrete structures. The Diesel Generator Buildings are seismic Category I structures which house the emergency diesel generators.

The Fuel Buildings are reinforced concrete structures. The Fuel Buildings are seismic Category I structures which provide storage for the new and spent fuel.

The Groundwater Drainage System is provided for the Auxiliary Building, Diesel Generator Buildings, and the Reactor Buildings and is designed to maintain the normal groundwater level near the base of these structures. The Groundwater Drainage System is composed of a network of horizontal foundation underdrains and continuous exterior wall drains. Three groundwater sumps are located along the perimeter of the Auxiliary Building. Additional information concerning the McGuire Groundwater Drainage System is contained in McGuire UFSAR Section 2.4.13, Groundwater. Additional information concerning the Catawba Groundwater Drainage System is contained in Catawba UFSAR Section 2.4.13, Groundwater.

The Main Steam Doghouses are reinforced concrete structures . The Main Steam Doghouses are seismic Category I structures which house the high pressure steam and feedwater piping.

The Upper Head Injection Tank Building at Catawba is a reinforced concrete structure. The Upper Head Injection Tank Building houses equipment that is within the scope of license renewal.

The structural components, component functions, and materials of construction for the Auxiliary Building Structures are listed in Table 3.5-2.

2.4.2.2 Condenser Cooling Water Intake Structure

McGuire Nuclear Station – The Condenser Cooling Water Intake Structure houses the three main fire pumps for the station. The Fire Pump rooms are the only parts of the structure that are within the scope of license renewal. The Condenser Cooling Water Intake Structure is constructed of reinforced concrete and carbon steel and provides structural support and/or shelter to components relied on during certain postulated fire events. The Condenser Cooling Water Intake Structure is a Category III structure and is not designed to withstand design basis seismic loadings.

The structural components, component functions, and materials of construction for the Condenser Cooling Water Intake Structure are listed in Table 3.5-2.

Catawba Nuclear Station – Only the portion of the McGuire Condenser Cooling Water Intake Structure that supports the fire pumps is within the scope of license renewal. The fire pumps at Catawba are supported by the Low Pressure Service Water Intake Structure. The Low Pressure Service Water Intake Structure is included with the Yard Structures.

2.4.2.3 Nuclear Service Water Structures

McGuire Nuclear Station – The Nuclear Service Water Structures include both the Standby Nuclear Service Water Pond Intake Structure and the Standby Nuclear Service Water Pond Discharge Structure. The Standby Nuclear Service Water Pond Intake Structure is a completely submerged reinforced concrete structure located at the bottom of the Standby Nuclear Service Water Pond east of the Standby Nuclear Service Water Pond Dam. The Intake Structure is designed to act as the headwall for the nuclear service water intake pipes and to provide missile protection for these pipes. The Standby Nuclear Service Water Pond Discharge Structure is located on the northern portion of the Standby Nuclear Service Water Pond near the pond surface. The Discharge Structure provides a concrete headwall to prevent erosion around the discharge pipes. Missile protection of the discharge pipes is provided by the soil backfill over a stepped concrete slab.

The structural components, component functions, and materials of construction for the Nuclear Service Water Structures are listed in Table 3.5-2.

Catawba Nuclear Station – The Nuclear Service Water Structures include several structures. The Nuclear Service Water and Standby Nuclear Service Water Pump Structure is a reinforced concrete structure founded on solid rock. The reinforced concrete roof, exterior walls, and interior wall provide missile protection. Reinforced concrete roof hatches provide a fire barrier and a missile barrier. Pressure doors are provided in the pump structure to withstand tornado induced negative pressure. The interior wall and some exterior walls provide fire barriers.

The Nuclear Service Water Conduit Manholes and the Nuclear Service Water Intake Structure are seismic Category I structures constructed of reinforced concrete. The Nuclear Service Water Intake Structure is submerged in the plant intake channel and is designed to house the Nuclear Service Water Intake Pipe, provide an intake chamber, secure fish impingement screens, provide missile protection for the intake pipe and act as an earth/silt retaining wall.

The Standby Nuclear Service Water Discharge Structures are reinforced concrete headwall structures designed as seismic Category I structures. Two structures are located within the pond. Each discharge structure is designed to house two Standby Nuclear Service Water discharge pipes, provide missile protection for the discharge piping, and act as an earth retaining wall.

The Standby Nuclear Service Water Intake Structure is a reinforced concrete box-shaped structure designed as seismic Category I. The Standby Nuclear Service Water Intake Structure houses the Nuclear Service Water Intake Pipe, provides an intake chamber, secures the fish impingement screens, provides missile protection for the intake pipe and acts as an earth/silt retaining wall.

The Standby Nuclear Service Water Pond Outlet is a seismic Category I structure consisting of a steel pipe located in the south abutment of the Standby Nuclear Service Water Pond with a reinforced concrete headwall on the Standby Nuclear Service Water Pond side and a reinforced concrete endwall on the Lake Wylie side. The headwall contains and protects the pipe, supports the missile shield, supports the weir and its missile shield, and contains the trash rack.

The structural components, component functions, and materials of construction for the Nuclear Service Water Structures are listed in Table 3.5-2.

2.4.2.4 Standby Nuclear Service Water Pond Dam

The Standby Nuclear Service Water Pond Dam is an earthen embankment that has been designed as a seismic Category I structure. The Standby Nuclear Service Water Pond Dam impounds the water within the Standby Nuclear Service Water Pond at each station to provide an alternate source of water for the Standby Nuclear Service Water System. The dam provides structural and/or functional support to safety-related equipment and ultimate heat sink following a postulated loss of coolant accident or loss of lake.

The structural components, component functions, and materials of construction for the Standby Nuclear Service Water Pond Dam are listed in Table 3.5-2.

2.4.2.5 Standby Shutdown Facility

The Standby Shutdown Facility structure houses a dedicated diesel generator and its supporting equipment, electrical equipment room batteries, and control rooms for both units of each station. The Standby Shutdown Facility structure is a steel-frame and masonry structure that is designed as a Category III structure. It is not designed to withstand design basis seismic loadings. The Standby Shutdown Facility structure provides structural support and/or shelter to components relied on during certain postulated events.

The structural components, component functions, and materials of construction for the Standby Shutdown Facility structure are listed in Table 3.5-2.

2.4.2.6 Turbine Buildings (including Service Building)

The Turbine Buildings and Service Building are Category III structures and are not designed to withstand design basis seismic loadings. The Turbine Buildings (including Service Building) at each station are constructed of a steel frame superstructure supported on a reinforced concrete substructure and provide structural support and/or shelter to components relied on during certain postulated events. The Turbine Buildings (including Service Building) are supported on foundations bearing on dense soil, partially weathered rock, and rock. At McGuire, the southern portion of the Service Building and the southwest portion of the Unit 1 Turbine Building are underlain by compacted soil and are supported on end bearing caissons.

The structural components, component functions, and materials of construction for the Turbine Buildings (including Service Building) are listed in Table 3.5-2.

2.4.2.7 Unit Vent Stack

The Unit Vent Stack is a seismic Category I structure that is a stiffened cylindrical shell. The vent stack is attached to the outside of the Reactor Building shell wall and is supported vertically by the Auxiliary Building roof. The Unit Vent Stack is provided for each unit and provides the primary release point for gaseous effluents from the unit.

The structural components, component functions, and materials of construction for the Unit Vent Stack are listed in Table 3.5-2.

2.4.2.8 Yard Structures

McGuire Nuclear Station – The Refueling Water Storage Tank is founded on residual soils and its foundation is a poured-in-place reinforced concrete composite structure. The Refueling Water Storage Tank missile wall encloses the foundation mat. The Refueling Water Storage Tank missile wall is a free-standing reinforced concrete structure that is capable of containing an assured source of water. The Refueling Water Storage Tank missile wall and foundation are seismic Category I structures. The Reactor Makeup Water Storage Tank foundation is designed as a seismic Category I structure and is constructed of reinforced concrete.

Trenches are provided throughout the McGuire yard to allow underground routing of cables and piping. Trenches that are within the scope of license renewal are the Refueling Water Storage Tank pipe trenches, the Standby Shutdown facility cable trenches, and the Condenser Cooling Water Intake structure cable trenches. Trenches are constructed of reinforced concrete and covers for the trenches are either made of concrete or consist of checkered steel plates.

The structural components, component functions, and materials of construction for the Yard Structures are listed in Table 3.5-2.

Catawba Nuclear Station – The Yard Structures of Catawba include the Yard Drainage System that provides a drainage system capable of protecting all safety-related structures from flooding during a local probable maximum precipitation event. The system consists of catch basin inlets that are connected by corrugated metal pipes to form several networks. The catch basis inlets are constructed of angle iron and grating.

The Refueling Water Storage Tank is founded on weathered rock or fill concrete to weathered rock and its foundation is a poured-in-place reinforced concrete composite structure. The Refueling Water Storage Tank missile wall encloses the foundation mat. The Refueling Water Storage Tank missile wall is a free-standing reinforced concrete structure that is capable of containing an assured source of water. The Refueling Water Storage Tank missile wall and foundation are seismic Category I structures.

Trenches are provided throughout the Catawba yard to allow underground routing of cables and piping. Trenches that are within the scope of license renewal are the Refueling Water Storage Tank pipe trenches and the Standby Shutdown facility cable trenches. Trenches are constructed of reinforced concrete and covers for the trenches are either made of concrete or consist of checkered steel plates.

The Low Pressure Service Water Intake Structure is a reinforced concrete structure that supports the fire pumps. The fire pumps are within the scope of license renewal and thus, the support structure is also within scope.

The structural components, component functions, and materials of construction for the Yard Structures are listed in Table 3.5-2.

2.4.3 COMPONENT SUPPORTS

Component supports are those components that provide support or enclosure for mechanical and electrical equipment. Component supports includes battery racks, cable tray and conduit, cable tray and conduit supports, control boards, crane rails, enclosures, equipment component supports, HVAC duct supports, instrument line supports, instrument racks & frames, lead shielding supports, new fuel storage racks, pipe supports, stair, platform and grating supports, and spent fuel storage racks. These supports are constructed of steel or stainless steel and are located in all of the structures within the scope of license renewal for McGuire Nuclear Station and Catawba Nuclear Station.

Also included within this section covering component supports are the Class 1 (Nuclear Steam Supply System) Supports. These Class 1 component supports include Reactor Coolant System piping supports, pressurizer upper and lower lateral supports, reactor vessel support, control rod drive seismic structure supports, steam generator vertical, lower lateral, and upper supports, and reactor coolant pump lateral and vertical support assemblies. Additional information concerning the Class 1 component supports at McGuire is contained in McGuire UFSAR Section 5.5.14, Component Supports. Additional information concerning the Class 1 component supports at Catawba is contained in Catawba UFSAR Section 5.4.14, Component Supports.

The structural components, component functions, and materials of construction for the Component Supports are listed in Table 3.5-3.

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2.5 SCOPING AND SCREENING RESULTS: ELECTRICAL AND INSTRUMENTATION AND CONTROLS

Note: The electrical and instrumentation and controls descriptions are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

The methodology to identify the electrical and instrumentation and controls (I&C) components that are subject to an aging management review is described in Section 2.1 (scoping) and Section 2.1.2.3 (passive screening and long-lived screening). To be subject to an aging management review, a component must meet all three criteria: scoping—§54.4(a), passive screening—§54.21(a)(1)(i), and long-lived screening §54.21(a)(1)(ii). Conversely, if a component or commodity does not meet any one of these criteria, then it is not subject to an aging management review.

The passive electrical and I&C commodities are identified in Table 2.1-2. Other electrical and I&C components are active and not subject to an aging management review.

Switchyard Systems were found not to meet any of the scoping criteria of §54.4(a). Consequently, the passive electrical commodities of switchyard bus, transmission conductors and high-voltage insulators are not within the scope of license renewal.

The Unit Main Power System and Nonsegregated-Phase Bus in the 6.9kV Normal Auxiliary Power System were found not to meet any of the scoping criteria of §54.4(a). Consequently, the passive electrical commodity of phase bus is not within the scope of license renewal.

The passive electrical commodity of uninsulated ground conductors was found not to meet any of the scoping criteria of §54.4(a). Consequently, uninsulated ground conductors are not within the scope of license renewal.

Electrical components included in the McGuire and Catawba *Environmental Qualification Program* do not meet the long-lived screening criterion of §54.21(a)(1)(ii). Consequently, some insulated cables and connections are not subject to an aging management review and none of the electrical portions of all electrical and I&C penetration assemblies are subject to an aging management review.

The result of this review is the determination that the electrical components that are subject to an aging management review are:

- Non-EQ insulated cables and connections
  (power, instrumentation and control applications; connections include plug-in connectors, splices and terminal blocks)

The intended function of insulated cables and connections is to provide an electrical connection to specified sections of an electrical circuit to deliver voltage, current or signals.

The pressure boundary function that may be associated with some electrical and I&C components (elements, RTDs, sensors, thermocouples, transducers, and heaters) is included in the process of identifying the mechanical pressure boundaries and is included in the applicable mechanical reviews within this Application (e.g., Sections 2.3, 3.1, 3.2, 3.3, and 3.4). Electrical components are supported by structural components (e.g., cable trays, conduit and cable trenches) that are included in the structural review provided in Sections 2.4 and 3.5.

The aging management review for non-EQ insulated cables and connections is provided in Section 3.6 of this Application and is summarized in Table 3.6-5.

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3.0 AGING MANAGEMENT REVIEW RESULTS

3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Note: The aging management reviews for all Reactor Coolant System components are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

The following Reactor Coolant System components are evaluated in the Table 3.1-1:

- Class 1 Piping, Valves, and Pump Casings
- Pressurizer
- Reactor Vessel and Control Rod Drive Mechanism Pressure Boundary
- Reactor Vessel Internals
- Steam Generator

3.1.1 AGING MANAGEMENT REVIEW RESULTS TABLES

The results of the aging management review for each component is provided in a table, as indicated above. Information contained in each table was obtained in the following manner:

Column 1 – The component types listed in Column 1 were identified through the screening methodology described in Section 2.1.2 of this application and are on the marked plant drawings identified in Section 2.3.2 of this application.

Column 2 – The component functions listed in Column 2 were obtained from plant specific engineering documents using the screening methodology described in Section 2.1.2.

Column 3 – The materials listed in Column 3 were obtained from the drawings identified in Section 2.3.2 of this application and other plant specific engineering documents.

Column 4 – The internal and external environments listed in Column 4 were obtained from plant specific engineering documents. External environments are also noted on the drawings identified in Section 2.3.2 of this Application. These environments are as follows:

- **Borated Water –** Borated water is demineralized water treated with boric acid.

- **Treated water –** Treated water is demineralized water that may be deareated, treated with a biocide or corrosion inhibitors, or a combination of these treatments. Treated water does not include borated water, which is separately evaluated.

- **Reactor Building –** The Reactor Building environment is moist air. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry.

Column 5 – The aging effects listed in Column 5 were obtained using the following aging effects identification process. The aging effects that require management during the period of extended operation have been determined by reviewing the plant-specific materials of construction (Column 3) and operating environments (Column 4) for each structure and component (Column 1) that is subject to an aging management review. The aging effects identification process assumes that licensed activities will continue to be conducted in accordance with the facilities' current licensing basis (e.g., use of low enriched uranium dioxide fuel only).

To provide reasonable assurance that the aging effects that require management for a specific material-environment combination are the only aging effects of concern for McGuire and Catawba, Duke also has performed a review of industry experience and NRC generic communications relative to these structures and components. Finally, relevant McGuire and Catawba operating experience have been reviewed to provide further confidence that the set of aging effects for the specific material-environment combinations have been identified. Taken together, the steps of this methodology provide reasonable assurance that the aging effects that require management during the period of extended operation for McGuire and Catawba structures and components have been identified.

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 [Footnote 3.1 - 1]. The Class 1 components fabricated from cast austenitic stainless steel (CASS) that are evaluated using the criteria in the May 19, 2000 letter are piping components, valve bodies, reactor coolant pump casings and main flanges,

3.1 - 1. 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.

and parts of the CRDM latch housing. The CASS components were evaluated using the criteria set forth in this letter with the following results:

- A delta ferrite calculation was performed for CASS piping and pump casings and only the McGuire Unit 1, Primary Loop 2, 27 ½ inch ID Loop B cold leg elbow exceeds the NRC established threshold for susceptibility to thermal embrittlement and requires aging management review for license renewal.
- Thermal embrittlement of the CASS valve bodies will be managed by the *Inservice Inspection Plan*.
- The CRDM latch housings are centrifigally cast and are not susceptible to thermal embrittlement.
- The reactor vessel internals CASS items will be managed by the *Reactor Vessel Internals Inspection.*

Reduction in fracture toughness due to irradiation embrittlement of the alloy steel reactor vessel is an aging effect only in the beltline region. Beltline is defined by 10 CFR 50.61 (a)(3) as the region of the reactor vessel that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage. Reduction of fracture toughness is also considered for stainless steel components that are adjacent to active fuel assemblies.

This aging effects identification process is consistent with that process used in Section 3.5 of the Oconee Nuclear Station license renewal application. Furthermore, in NUREG-1723, the staff concluded that based on its review of the information provided in Sections 3.5.1 and 3.5.2 of the Oconee application, “the applicant has identified the aging effects that are associated with mechanical systems components reviewed in [Section 3.5].” This aging effects identification process provides reasonable assurance that the aging effects that require management during the period of extended operation have been identified.

Column 6 – The aging management programs and activities listed in Column 6 are credited to manage the effects of aging for the period of extended operation.

For bolting, in addition to the aging management programs listed, information from operating experience indicates that there are additional elements of bolting maintenance procedures that should be considered, such as personnel training, installation and maintenance procedures, plant-specific bolting degradation history, and corrective measures. The NRC captured the lessons from this experience in IE Bulletin 82-02 [Footnote 3.1 - 2] and directed each licensee to assure that these lessons were being incorporated at their plant. In response to IE Bulletin 82-02, Duke provided the results of the in-house investigation and provided assurance that bolting maintenance practices did indeed consider these lessons learned. In summary, routine maintenance practices have included use of properly trained personnel and procedural guidance to construct bolted closures. The continuation of routine maintenance practices reviewed under IE Bulletin 82-02 will assure aging management of mechanical closure integrity for bolted closures in the Reactor Coolant System.

3.1.2 AGING MANAGEMENT PROGRAMS

The following aging management programs and activities are credited to manage the effects of aging for the components of the Reactor Coolant System listed in Section 3.1:

- Alloy 600 Aging Management Review
- Bottom-Mounted Instrumentation Thimble Tube Inspection Program
- Chemistry Control Program *
- Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program*
- Flow Accelerated Corrosion Program*
- Fluid Leak Management Program*
- Inservice Inspection Plan*
- Reactor Coolant System Operational Leakage Monitoring Program*
- Reactor Vessel Integrity Program*
- Reactor Vessel Internals Inspection *
- Steam Generator Surveillance Program*

** This aging management program/activity is equivalent or similar to the corresponding program/activity that has been previously reviewed and found acceptable by the NRC staff during the Oconee License Renewal review, as documented in NUREG-1723.*

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in Table 3.1-1 will be maintained consistent with the current licensing basis for the period of extended operation.

3.1 - 2. IE Bulletin 82-02 dated June 2, 1982, *Degradation of Threaded Fasteners in Reactor Coolant Pressure Boundary of PWR Plants*.

Table 3.1-1 Aging Management Review Results – Reactor Coolant System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Exterior Surfaces and Bolted Closures					
Exterior surfaces of pressure boundary components	PB	Alloy Steel, Carbon Steel	Reactor Building	Loss of Material	Fluid Leak Management Program
Valve Bolting Material (all bolting is less than 2 inches)	PB	Stainless Steel	Reactor Building	Cracking Loss of Preload	Inservice Inspection Plan
Reactor Coolant Pump Main Flange Bolts	PB	Alloy Steel, Stainless Steel	Reactor Building	Cracking Loss of material (ferritic fasteners) Loss of Preload	Inservice Inspection Plan Fluid Leak Management Program
Pressurizer Manway Cover Bolts/Studs	PB	Alloy Steel	Reactor Building	Cracking Loss of Material Loss of Preload	Inservice Inspection Plan Fluid Leak Management Program
Reactor Vessel Closure Studs, nuts, and washers	PB	Alloy Steel	Reactor Building	Cracking, Loss of Material Loss of Preload	Inservice Inspection Plan Reactor Coolant System Operational Leakage Monitoring Program
Steam Generator Bolting	PB	Alloy Steel	Reactor Building	Cracking Loss of Material Loss of Preload	Fluid Leak Management Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Exterior Surfaces and Bolted Closures (continued)					
Exterior Surfaces of Pressure Boundary Components	PB	CASS, Stainless Steel	Reactor Building	None Identified	None Required
Reactor Vessel and Pressurizer Integral Attachments	Support	Alloy Steel, Carbon Steel	Reactor Building	Cracking Loss of Material	Inservice Inspection Plan Fluid Leak Management Program
Class 1 Piping, Valve Bodies, and Pump Casings					
Hot and Cold Leg Pipe – 31” ID, 29” ID, 27.5” ID	PB	CASS	Borated water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
MNS Unit 1 27.5” ID Loop B elbow	PB	CASS (SA-351 CF8A)	Borated Water	Loss of Material Cracking Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan
Pipe and Fittings NPS≤ 1”	PB	Stainless steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Class 1 Piping, Valve Bodies, and Pump Casings (continued)					
Pipe fittings and branch connections 1” > NPS < 4” (includes sample scoops and thermowells)	PB	Stainless steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Pipe, fittings, and branch connections NPS ≥ 4” (includes spray scoops)	PB	Stainless steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Orifices	PB, TH	Stainless steel	Borated Water	Loss of Material Cracking	Chemistry Control Program
Forged Stainless Steel Valve bodies and/or Bonnets	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program
Cast Stainless Steel Valve bodies and/or Bonnets	PB	CASS	Borated Water	Loss of Material Cracking Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Class 1 Piping, Valve Bodies, and Pump Casings (continued)					
Reactor Coolant Pump Casings	PB	CASS	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Main Pump Closure Flange	PB	CASS	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Thermal Barrier Heat Exchanger piping (tubing) and flanges	PB	Stainless steel	Treated Water	Loss of Material Cracking	Chemistry Control Program
			Borated Water	Loss of Material Cracking	Chemistry Control Program

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pressurizer					
Lower Head Shell Upper Head Manway	PB	Alloy Steel Clad with Stainless Steel	Borated Water	Loss of Material (cladding) Cracking	Chemistry Control Program Inservice Inspection Plan
Surge Nozzle Spray Nozzle Relief Nozzle Safety Nozzle	PB	Alloy Steel Clad with Stainless Steel (Nickel Based Alloy Weld Build Up)	Borated Water	Loss of Material (cladding) Cracking	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Immersion Heaters Sheath	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program
Surge and Spray Nozzle Thermal Sleeves	PB	Stainless Steel (Nickel-Based Alloy Weld)	Borated Water	Loss of Material Cracking	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pressurizer (continued)					
Support Skirt and Flange	Component Support	Carbon Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inservice Inspection Plan
Manway Insert	PB	Stainless Steel or Nickel-Based Alloy	Borated Water	Loss of material Cracking	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Heater Well Nozzle	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Instrument Nozzles	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Surge Nozzle Safe End Spray Nozzle Safe End Relief Nozzle Safe End Safety Nozzle Safe End	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel and CRDM Pressure Boundary Components					
Closure head dome, flange, ring and vessel flange	PB	Alloy Steel Stainless Steel Clad	Borated Water	Loss of Material (cladding) Cracking	Chemistry Control Program Inservice Inspection Plan
Upper (nozzle) shell	PB	Alloy Steel Stainless Steel Clad	Borated Water	Loss of Material (cladding) Cracking	Chemistry Control Program Inservice Inspection Plan
Primary Inlet and Outlet Nozzles	PB	Alloy Steel Forging Stainless Steel Clad (Nickel Based Alloy Buttering and weld)	Borated Water	Loss of Material (cladding) Cracking	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Inlet and Outlet Nozzle Safe Ends	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Intermediate shell	PB	Alloy Steel Stainless Steel Clad	Borated Water	Loss of Material (cladding) Cracking Reduction of Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Integrity Program

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel and CRDM Pressure Boundary Components (continued)					
Lower Shell	PB	Alloy Steel Stainless Steel Clad	Borated Water	Loss of Material (cladding) Cracking Reduction of Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Integrity Program
Bottom head spherical ring, dome	PB	Alloy Steel Stainless Steel Clad	Borated Water	Loss of Material (cladding) Cracking	Chemistry Control Program Inservice Inspection Plan
CRDM Housings	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
CRDM Housings	PB	CASS	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel and CRDM Pressure Boundary Components (continued)					
CRDM Housings	PB	Nickel Based Alloy	Borated Water	Loss of Material Cracking	Alloy 600 Aging Management Review Chemistry Control Program CRDM and Other Vessel Closure Penetration Inspection Program Inservice Inspection Plan Reactor Coolant System Operational Leakage Monitoring Program
UHI Auxiliary Head Adapter Flange	PB	Nickel Based Alloy and Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan CRDM and Other Vessel Closure Penetration Inspection Program
Head Vent Penetration	PB	Nickel Based Alloy and Stainless Steel	Borated Water	Loss of Material Cracking	Alloy 600 Aging Management Review Chemistry Control Program CRDM and Other Vessel Closure Penetration Inspection Program
Thimble Assembly	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Bottom Mounted Thimble Tube Instrumentation Inspection Program

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel and CRDM Pressure Boundary Components (continued)					
Bottom Mounted Instrumentation tubes (penetrations)	PB	Nickel Based Alloy	Borated Water	Loss of Material Cracking	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Thimble Guide Tubes	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Thimble Seal Table	PB	Stainless Steel	Borated Water	Loss of Material Cracking	Chemistry Control Program
Core Support Pads	Support	Nickel Based Alloy	Borated Water	Loss of Material Cracking	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Reactor Vessel Internals					
Upper Core Support Structure					
Upper Support Assembly (Forging, Plates, Weld))	1,2,3,4	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Upper Support Column	1,2,4	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Upper Support Column (Base, Conduit Support, Thermocouple stop (U1))	1,2,4	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel Internals (continued)					
Upper Core Support Structure (continued)					
Upper Support Column Bolts	1,2,4	Stainless Steel	Borated Water	Cracking Loss of Material Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Upper Core Plate	1,2,3,4	Stainless Steel	Borated Water	Cracking Loss of Material Dimensional Changes Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Upper Core Plate Alignment Pins	1	Stainless Steel	Borated Water	Cracking Loss of Material Dimensional Changes	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Fuel Alignment Pins	1	Stainless Steel	Borated Water	Cracking Loss of Material Dimensional Changes Loss of Preload	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel Internals (continued)					
Upper Core Support Structure (continued)					
Hold Down Spring	1	Stainless Steel	Borated Water	Cracking Loss of Material Loss of Preload	Chemistry Control Program Inservice Inspection Plan
Thermocouple Column and Crossrun Assemblies	4	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
17X17 and 15X15 Guide Tube Assembly	2 (17X17 only), 3	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel Internals (continued)					
Upper Core Support Structure (continued)					
15X15 and 17X17 Guide Tube Assembly	3	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
UHI Flow Columns	3	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
UHI Flow Columns (Base)	3	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel Internals (continued)					
Lower Core Support Structure					
Core Barrel Flange Core Barrel Outlet Nozzles Neutron Panels Irradiation Specimen Holder Fasteners	1,3,4,5,6	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Irradiation Specimen Holder (spring)	5	Nickel Based Alloy	Borated Water	Loss of Material Cracking	Alloy 600 Aging Management Review Inservice Inspection Plan Chemistry Control Program
Baffle and Former Plates	1,3,6	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel Internals (continued)					
Lower Core Support Structure (continued)					
Baffle Bolts (baffle to baffle, baffle to former)	1,3	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Lower Core Plate Fuel Alignment Pins Lower Support Column Bolts	1,3,4,5	Stainless Steel	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness Dimensional Changes Loss of Preload (bolting)	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection
Lower Support Plate (forging) Lower Core Support Columns	1,3,4,5,6	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Radial Keys and fasteners	1	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Vessel Internals (continued)					
Lower Core Support Structure (continued)					
Clevis Inserts and fasteners	1	Nickel Based Alloy	Borated Water	Cracking Loss of Material	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Bottom Mounted Instrumentation					
Bottom Mounted Instrumentation (Plates, forgings, welds, energy absorber, fasteners)	3,4,5	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Bottom Mounted Instrumentation (upper end, cruciform)	4	CASS	Borated Water	Cracking Loss of Material Reduction in Fracture Toughness	Chemistry Control Program Inservice Inspection Plan Reactor Vessel Internals Inspection

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
			Steam Generators		
Primary Head/Cladding (CNS – 2 only)	PB	Alloy steel Stainless Steel Clad	Borated Water	Loss of Material (cladding) Cracking	Chemistry Control Program Inservice Inspection Plan
Primary Nozzle Closure Rings (CNS – 2 only)	PB	Nickel Based Alloy	Borated Water	Cracking Loss of Material	Chemistry Control Program Alloy 600 Aging Management Review
Secondary Manway (CNS – 2 only)	PB	Alloy Steel	Treated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Secondary Manway Covers Handhole Covers (CNS – 2 only)	PB	Alloy Steel	Treated Water	Loss of Material	Fluid Leak Management Program
Handhole Pad Minor Nozzle Bosses (CNS – 2 only)	PB	Alloy Steel	Treated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Steam Generators (continued)					
Tubesheet/ primary and secondary cladding	PB	Alloy Steel Nickel Based Alloy Clad	Borated Water (primary)	Cracking Loss of Material (cladding)	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
			Treated Water (secondary)	Loss of Material	Chemistry Control Program
Tubes/plugs	PB	Nickel Based Alloy (Alloy 690) (CNS-2 – Alloy 600)	Borated Water (Primary Side)	Cracking Loss of Material	Chemistry Control Program Steam Generator Surveillance Program
			Treated Water (Secondary Side)	Cracking Loss of Material	Chemistry Control Program Steam Generator Surveillance Program
Primary Nozzles	PB	Alloy Steel Stainless Steel Clad (CNS-2 carbon steel w/ Stainless Steel Clad) (Nickel Based Alloy weld buttering)	Borated Water	Cracking Loss of Material (cladding)	Alloy 600 Aging Management Review (Note 3) Chemistry Control Program Inservice Inspection Plan

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Steam Generators (continued)					
Primary Nozzle Safe Ends (Note 3)	PB	Stainless Steel	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Primary Manway Cover Plate/Diaphragm (Note 3)	PB	Alloy Steel/ Nickel Based Alloy	Borated Water	Cracking Loss of Material	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Primary Divider Plate	PB	Nickel Based Alloy	Borated Water	Cracking Loss of Material	Alloy 600 Aging Management Review Chemistry Control Program Inservice Inspection Plan
Steam drum boiler shells, Steam dome conical shells handhole	PB	Alloy Steel	Treated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Handhole Diaphragm (Note 3)	PB	Nickel Based Alloy	Treated Water	Cracking Loss of Material	Chemistry Control Program

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
		Steam Generators (continued)			
Small Nozzles (Note 3)	PB	Nickel Based Alloy Alloy Steel Weld Buildup	Treated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Primary Manway and Manway Insert (CNS – 2 only)	PB	Carbon Steel Stainless Steel Clad	Borated Water	Cracking Loss of Material	Chemistry Control Program Inservice Inspection Plan
Primary Chamber Drain and Coupling (CNS – 2 only)	PB	Stainless Steel, Nickel based Alloy	Borated Water	Cracking Loss of Material	Alloy 600 Aging Management Review Chemistry Control Program
Feedwater Thermal Sleeve, Feedwater Limiter Steam Outlet Nozzle (CNS – 2 only)	PB	Nickel based Alloy	Treated Water	Cracking Loss of Material	Chemistry Control Program

Table 3.1-1 Aging Management Review Results – Reactor Coolant System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Steam Generators (continued)					
Flow Restrictor	PB,TH	Stainless Steel	Treated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Steam Outlet Nozzle Safe End (Note 3)	PB	Carbon Steel	Treated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Auxiliary Feedwater Nozzle Main Feedwater Nozzle Steam Outlet Nozzle	PB	Alloy Steel	Treated Water	Loss of Material Cracking	Chemistry Control Program Flow Accelerated Corrosion Program (Main Feedwater Nozzle) Inservice Inspection Plan
Auxiliary Feedwater Nozzle Safe End (Note 3)	PB	Nickel-Based Alloy	Treated Water	Loss of Material Cracking	Chemistry Control Program Inservice Inspection Plan
Auxiliary Feedwater Distribution System (Note 3)	PB	Alloy Steel	Treated Water	Loss of Material	Chemistry Control Program

Notes for Table 3.1-1 Aging Management Review Results – Reactor Coolant System:

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.
	Reactor Vessel Internals functions
1	Provide support and orientation of the reactor core (i.e. the fuel assemblies)
2	Provide support, orientation, guidance and protection of the control rod assemblies.
3	Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
4	Provide a passageway for support, guidance, and protection for the incore instrumentation.
5	Provide secondary core support for limiting the downward displacement of the core support structure in the event of a postulated failure of the core barrel.
6	Provide neutron shielding to the reactor vessel and provide support for vessel material test specimens.
(2)	**Material**
	Alloy Steel refers to low alloy steel. Nickel Based Alloys include Inconel and Alloys 600 and 690.
CASS	Cast Austenitic Stainless Steel
(3)	The topic applies to McGuire Units 1 and 2 and to Catawba Unit 1 steam generators only.

This is the last page of Section 3.1

3.2 AGING MANAGEMENT OF ENGINEERED SAFETY FEATURES

Note: The aging management reviews for all engineered safety features systems are considered to be generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

The following mechanical systems are evaluated in the indicated tables in Section 3.2, Engineered Safety Features:

- Annulus Ventilation System (Table 3.2-1)
- Containment Isolation System (Table 3.2-2)
- Containment Air Return & Hydrogen Skimmer System (Table 3.2-3)
- Containment Spray System (Table 3.2-4)
- Containment Valve Injection Water System (Table 3.2-5)
- Refueling Water System (Table 3.2-6)
- Residual Heat Removal System (Table 3.2-7)
- Safety Injection System (Table 3.2-8)

3.2.1 AGING MANAGEMENT REVIEW RESULTS TABLES

The results of the aging management review for each system of this section are provided in a table, as indicated above. Information contained in each table was obtained in the following manner:

Column 1 – The component types listed in Column 1 were identified through the screening methodology described in Section 2.1.2 of this application and are on the marked plant drawings identified in Section 2.3.2 of this application.

Column 2 – The component functions listed in Column 2 were obtained from plant specific engineering documents using the screening methodology described in Section 2.1.2.

Column 3 – The materials listed in Column 3 were obtained from the drawings identified in Section 2.3.2 of this application and other plant specific engineering documents.

Column 4 – The internal and external environments listed in Column 4 were obtained from plant specific engineering documents. External environments are also noted on the drawings identified in Section 2.3.2 of this application. These environments are as follows:

- **Air-Gas –** Compressed air is ambient air that has been filtered and compressed for use in plant equipment. Compressed air may be either dry or oiled. Compressed gases include carbon dioxide, hydrogen, nitrogen, Freon, or refrigeration gases used to replace Freon due to environmental concerns.

- **Borated Water –** Borated water is demineralized water treated with boric acid.

- **Raw Water –** Raw water is water from a lake, pond, or river that has been rough-filtered and possibly treated with a biocide.

- **Treated water –** Treated water is demineralized water that may be deareated, treated with a biocide or corrosion inhibitors, or a combination of these treatments. Treated water does not include borated water, which is evaluated separately.

- **Sheltered environment –** The ambient conditions within the sheltered environment may or may not be controlled. The sheltered environment atmosphere is a moist air environment. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry.

- **Reactor Building –** The Reactor Building environment is moist air. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry.

- **Ventilation** – Ambient air that is conditioned to maintain a suitable environment for equipment operation and personnel occupancy.

- **Yard –** Yard environment is a moist air environment in which equipment is exposed to heat, cold, and precipitation.

Column 5 – The aging effects listed in Column 5 were obtained using the following aging effects identification process. The aging effects that require management during the period of extended operation have been determined by reviewing the plant-specific materials of construction (Column 3) and operating environments (Column 4) for each structure and component (Column 1) that is subject to an aging management review.

To provide reasonable assurance that the aging effects that require management for a specific material-environment combination are the only aging effects of concern for McGuire and Catawba, Duke also performed a review of industry experience and NRC generic communications relative to these structures and components. Finally, relevant McGuire and Catawba operating experience have been reviewed to provide further confidence that the set of aging effects for the specific material-environment combinations have been identified. Taken together, the steps of this methodology provide reasonable assurance that the aging effects that require management during the period of extended operation for McGuire and Catawba structures and components have been identified.

This aging effects identification process is consistent with that process used in Section 3.5 of the Oconee Nuclear Station license renewal application. Furthermore, in NUREG-1723, the staff concluded that based on its review of the information provided in Sections 3.5.1 and 3.5.2 of the Oconee application, "the applicant has identified the aging effects that are associated with mechanical systems components reviewed in [Section 3.5]." This aging effects identification process provides reasonable assurance that the aging effects that require management during the period of extended operation have been identified.

Column 6 – The aging management programs and activities listed in Column 6 are credited to manage the effects of aging for the period of extended operation.

3.2.2 AGING MANAGEMENT PROGRAMS

The following aging management programs and activities are credited to manage the effects of aging for the engineered safety features systems listed in Section 3.2:

- Borated Water Systems Stainless Steel Inspection *
- Chemistry Control Program *
- Flow Accelerated Corrosion Program *
- Fluid Leak Management Program *
- Galvanic Susceptibility Inspection *
- Heat Exchanger Performance Testing Activities – Containment Spray Heat Exchangers
- Heat Exchanger Preventive Maintenance Activities – Containment Spray
- Inspection Program for Civil Engineering Structures and Components *
- Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection
- Service Water Piping Corrosion Program *
- Treated Water Systems Stainless Steel Inspection *

** This aging management program/activity is equivalent or similar to the corresponding program/activity that has been previously reviewed and found acceptable by the NRC staff during the Oconee License Renewal review, as documented in NUREG-1723.*

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in Tables 3.2-1 through 3.2-8 will be maintained consistent with the current licensing basis for the period of extended operation.

Table 3.2-1 Aging Management Review Results – Annulus Ventilation System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Air Flow Monitors	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Ductwork	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Ductwork	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Filters	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe (MNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Tubing	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	CU	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.2-1 Aging Management Review Results – Annulus Ventilation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies (MNS Only)	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required

Notes for Table 3.2-1 Aging Management Review Results – Annulus Ventilation System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CU Copper
CS Carbon Steel
SS Stainless Steel

Table 3.2-2 Aging Management Review Results – Containment Isolation System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Breathing Air System					
Pipe	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	SS	Air-Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Air-Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Containment Air Release and Addition System					
Pipe (CNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (CNS Only)	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Containment Air Release and Addition System (continued)					
Tubing (CNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Tubing (CNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies (CNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Containment Hydrogen Sample and Purge System (Catawba Nuclear Station)					
Orifice	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Containment Hydrogen Sample and Purge System (Catawba Nuclear Station) (continued)					
Pipe	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Tubing	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Tubing	PB	CU	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Containment Purge Ventilation System					
Pipe	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Containment Purge Ventilation System (continued)					
Tubing (CNS Only)	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Tubing (CNS Only)	PB	CU	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Tubing	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Valve Bodies (CNS Only)	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies (CNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Containment Ventilation Cooling Water (McGuire Nuclear Station only)					
Annubars	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required
Annubars	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required
Pipe	PB	CS	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Containment Ventilation Cooling Water (McGuire Nuclear Station only)(continued)					
Tubing	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Containment Ventilation Cooling Water (McGuire Nuclear Station only)(continued)					
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Conventional Chemical Addition System (McGuire Nuclear Station only)					
Pipe	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Equipment Decontamination (Catawba Nuclear Station only)					
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Embedded	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Equipment Decontamination (Catawba Nuclear Station only)(continued)					
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Ice Condenser Refrigeration System					
Pipe	PB	CS	Treated Water (Note 3)	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Ice Condenser Refrigeration System (continued)					
Pipe	PB	CS	Treated Water (Note 3)	Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (MNS only)	PB	CS	Ventilation (Note 5)	None Identified	None Required
			Embedded (Note 6)	None Identified	None Required
Pipe	PB	CS	Ventilation (Note 5)	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe (CNS only)	PB	SS	Ventilation	None Identified	None Required
			Embedded (Note 6)	None Identified	None Required
Pipe (MNS only)	PB	SS	Ventilation (Note 4)	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	SS	Ventilation (Note 4)	None Identified	None Required
			Reactor Building	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Ice Condenser Refrigeration System (continued)					
Pipe (MNS only)	PB	SS	Ventilation (Note 5)	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe	PB	Transite	Ventilation (Note 5)	None Identified	None Required
			Embedded (Note 6)	None Identified	None Required
Valve Bodies	PB	CS	Treated Water (Note 3)	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water (Note 3)	Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Ice Condenser Refrigeration System (continued)					
Valve Bodies	PB	SS	Treated Water (Note 3)	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water (Note 3)	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies (MNS only)	PB	SS	Ventilation (Note 4)	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Ventilation (Notes 4, 5)	None Identified	None Required
			Reactor Building	None Identified	None Required
Make-up Demineralized Water System					
Pipe	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	
			Sheltered	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Make-up Demineralized Water System (continued)					
Valve Bodies	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	
			Sheltered	None Identified	None Required
Station Air System					
Pipe	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	SS	Air-Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Air-Gas	None Identified	None Required
			Reactor Building	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (1)**	**Material (2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Steam Generator Blowdown Recycle System					
Pipe (CNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (CNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe (MNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steam Generator Blowdown Recycle System (continued)					
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (Catawba only)
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies (CNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies (MNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steam Generator Wet Lay-Up Recirculation System					
Pipe (MNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-2 Aging Management Review Results – Containment Isolation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steam Generator Wet Lay-Up Recirculation System (continued)					
Valve Bodies (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.2-2 Aging Management Review Results – Containment Isolation System:

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
(2)	**Material**
BR	Brass
CU	Copper
CS	Carbon Steel
SS	Stainless Steel
Transite	Non-metallic cement-asbestos
(3)	50% weight/volume aqueous solution containing corrosion inhibitors is maintained between –5°F and 0°F.
(4)	Infrequent (temporary) exposure to either borated ice/air (~20°F) mixture or ~20°F air for ice loading components. Normal environment is ambient Reactor Building air. (McGuire only)
(5)	Also exposed infrequently to small amounts of spillage/defrost water, as well as ethylene glycol solution. Surface coated to minimize effects of sodium tetraborate. Environment is predominantly cold air from the ice condenser. (McGuire only)
(6)	Although external temperature is above freezing, a gradient consistent with the wear slab/structural subfloor temperature gradient is experienced (i.e. upper portions of embedded pipe are at a lower temperature than the lower portion which exits into the Reactor Building.)

Table 3.2-3 Aging Management Review Results – Containment Air Return & Hydrogen Skimmer System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Ductwork (CNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Expansion Joints (CNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Tubing (MNS Only)	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Tubing (MNS Only)	PB	CU	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Tubing (CNS Only)	PB	SS	Air-Gas	None Identified	None Required
			Reactor Building	None Identified	None Required

Table 3.2-3 Aging Management Review Results – Containment Air Return & Hydrogen Skimmer System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.2-3 Aging Management Review Results – Containment Air Return & Hydrogen Skimmer System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CU Copper
CS Carbon Steel
SS Stainless Steel

Table 3.2-4 Aging Management Review Results – Containment Spray System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Flow Orifices	TH, PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
McGuire Heat Exchangers 1NSHX0003, 1NSHX0004, and 2 NSHX0003					
Heat Exchangers (1NSHX0003, 1NSHX0004, 2NSHX0003) [Channel Head]	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	
		CS	Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchangers (NSHX0003, 1NSHX0004, 2NSHX0003) [Shell]	PB	CS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
		SS (Note 3)	Raw Water	Loss of Material	Service Water Piping Corrosion Program
		CS	Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-4 Aging Management Review Results – Containment Spray System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Heat Exchangers (NSHX0003, 1NSHX0004, 2NSHX0003) [Tubes]	PB, HT	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	
			Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Containment Spray
				Fouling	Heat Exchanger Performance Testing Activities – Containment Spray Heat Exchangers
Heat Exchangers (NSHX0003, 1NSHX0004, 2NSHX0003) [Tube Sheet]	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Raw Water	Loss of Material	Service Water Piping Corrosion Program
McGuire Heat Exchanger 2NSHX0004					
Heat Exchanger (2NSHX0004) [Channel Head]	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
		SS (Note 4)	Sheltered	None Identified	None Required
		CS (Note 4)	Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-4 Aging Management Review Results – Containment Spray System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
McGuire Heat Exchanger 2NSHX0004 (continued)					
Heat Exchanger (2NSHX0004) [Shell]	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	
		SS (Note 4)	Sheltered	None Identified	None Required
		CS (Note 4)	Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger (2NSHX0004) [Tubes]	PB, HT	TI	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Containment Spray
				Fouling	Heat Exchanger Performance Testing Activities – Containment Spray Heat Exchangers
			Borated Water	None Identified	None Required
Heat Exchanger (2NSHX0004) [Tubesheets]	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program

Table 3.2-4 Aging Management Review Results – Containment Spray System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Catawba Heat Exchangers					
Heat Exchangers [Channel Head]	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
		SS	Sheltered	None Identified	None Required
		CS (Note 5)	Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchangers [Shell]	PB	CS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchangers [Tubes]	PB, HT	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Containment Spray
				Fouling	Heat Exchanger Performance Testing Activities – Containment Spray Heat Exchangers

Table 3.2-4 Aging Management Review Results – Containment Spray System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Heat Exchangers [Tubesheet]	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
		CS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
Pipe	PB	SS	Borated Water (Note 6)	Loss of Material	Borated Water Systems Stainless Steel Inspection
				Cracking	
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Ventilation (Note 7)	None Identified	None Required
			Reactor Building	None Identified	None Required
Pump Casings	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Spray Nozzles	PB, SP	SS	Ventilation (Note 7)	None Identified	None Required
			Reactor Building	None Identified	None Required

Table 3.2-4 Aging Management Review Results – Containment Spray System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Tubing	PB	SS	Borated Water	Loss of Material Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Ventilation (Note 7)	None Identified	None Required
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Ventilation (Note 7)	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Loss of Material Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Ventilation (Note 7)	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Ventilation (Note 7)	None Identified	None Required
			Reactor Building	None Identified	None Required

Notes for Table 3.2-4 Aging Management Review Results – Containment Spray System:

(1) Component Function

HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
SP	Provide spray flow so that sufficient spray flow and/or flow pattern are maintained
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS	Carbon Steel
SS	Stainless Steel
TI	Titanium

(3) The shells of 1NSHX0003, 1NSHX0004 and 2NSHX0003 are constructed of carbon steel. However, certain nozzles on the shell cover are stainless steel, with an overlay on surrounding carbon steel shell material. Therefore, both a Carbon Steel/Raw Water and a Stainless Steel/Raw Water material/environment combination exist for these shells.

(4) Certain portions of heat exchanger sub-components have both Stainless Steel and Carbon Steel portions exposed to a Sheltered Environment.

(5) While the majority of the external surface of the heat exchanger sub-component is stainless steel, certain portions are carbon steel.

(6) Portions of subject pipe vertical risers inside the Reactor Building, are subject to an alternate wet/dry borated water environment as system fill level changes during discharge valve testing and due to downstream components being open to Reactor Building ambient air.

(7) Portions of subject pipe, valves and spray nozzles, although designed for borated water, are normally open to the Reactor Building ambient air (considered a Ventilation environment for license renewal considerations) since they are located above the expected elevation of the FWST, to which the Containment Spray system is directly connected. Additionally, tubing and valves associated with Containment pressure measurement are open to the Reactor Building ambient air.

Table 3.2-5 Aging Management Review Results – Containment Valve Injection Water System

(Catawba Nuclear Station only)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Pipe	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Reactor Building	None Identified	None Required
Tanks	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Tubing	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required

Table 3.2-5 Aging Management Review Results – Containment Valve Injection Water System (Catawba Nuclear Station only) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Tubing	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required

Notes for Table 3.2-5 Aging Management Review Results – Containment Valve Injection Water System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

SS Stainless Steel

Table 3.2-6 Aging Management Review Results – Refueling Water System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Expansion Joint (MNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Cracking (Note 3)	Borated Water Systems Stainless Steel Inspection
				Loss of Material (Note 3)	Borated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Cracking (Note 3)	Borated Water Systems Stainless Steel Inspection
				Loss of Material (Note 3)	Borated Water Systems Stainless Steel Inspection
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required

Table 3.2-6 Aging Management Review Results – Refueling Water System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Refueling Water Storage Tank (MNS only)	PB	CS	Ventilation	Loss of Material	Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Refueling Water Storage Tank (MNS only)	PB	CS	Borated Water	Loss of Material	Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection
				Loss of Material (Note 3)	
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Refueling Water Storage Tank (CNS only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Cracking (Note 3)	Borated Water Systems Stainless Steel Inspection
				Loss of Material (Note 3)	Borated Water Systems Stainless Steel Inspection
			Yard	None Identified	None Required
Refueling Water Storage Tank (CNS only)	PB	SS	Ventilation	None Identified	None Required
			Yard	None Identified	None Required

Table 3.2-6 Aging Management Review Results – Refueling Water System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Cracking (Note 3)	Borated Water Systems Stainless Steel Inspection
				Loss of Material (Note 3)	Borated Water Systems Stainless Steel Inspection
			Reactor Building	None Identified	None Required

Table 3.2-6 Aging Management Review Results – Refueling Water System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
					Chemistry Control Program
				Cracking (Note 3)	Borated Water Systems Stainless Steel Inspection
				Loss of Material (Note 3)	Borated Water Systems Stainless Steel Inspection
					Borated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Valve Bodies (MNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required

Notes for Table 3.2-6 Aging Management Review Results – Refueling Water System:

(1) Component Function

HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS	Carbon Steel
SS	Stainless Steel

(3) Component subject to alternate wetting and drying

Table 3.2-7 Aging Management Review Results – Residual Heat Removal System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Heat Exchangers, RHR (tubes)	PB, HT	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
Heat Exchangers, RHR (tube sheet)	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
Heat Exchangers, RHR (channel head)	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Heat Exchangers, RHR (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-7 Aging Management Review Results – Residual Heat Removal System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
RHR Pump Seal Water (tubes)	PB, HT	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
RHR Pump Seal Water (shell)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Orifices	PB, TH	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Heat Exchanger RHR Pump Seal Water (cover) (CNS only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.2-7 Aging Management Review Results – Residual Heat Removal System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Orifices (All others)	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Pump Casings, RHR	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.2-7 Aging Management Review Results – Residual Heat Removal System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.2-7 Aging Management Review Results – Residual Heat Removal System:

(1) Component Function

HT Provide heat transfer so that system and/or component operating temperatures are maintained.

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS Carbon Steel
SS Stainless Steel

Table 3.2-8 Aging Management Review Results – Safety Injection System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Orifices	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Orifices	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB, TH	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Orifices	PB,TH	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	CS	Air-Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-8 Aging Management Review Results – Safety Injection System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	CS	Air-Gas	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Air-Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Safety Injection Cold Leg Accumulators (eight)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
		CS	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Safety Injection Cold Leg Accumulators (eight)	PB	SS	Air-Gas	None Identified	None Required
		CS	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Safety Injection Pump Casings (four)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.2-8 Aging Management Review Results – Safety Injection System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Air-Gas	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.2-8 Aging Management Review Results – Safety Injection System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Air-Gas	None Identified	None Required
			Reactor Building	None Identified	None Required

Notes for Table 3.2-8 Aging Management Review Results – Safety Injection System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS Carbon Steel
SS Stainless Steel

This is the last page of Section 3.2

3.3 AGING MANAGEMENT OF AUXILIARY SYSTEMS

Note: The aging management reviews for all auxiliary systems are considered to be generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

The following mechanical systems are evaluated in the indicated tables in Section 3.3, Auxiliary Systems:

- Auxiliary Building Ventilation System (Table 3.3-1)
- Boron Recycle System (Table 3.3-2)
- Building Heating Water System – Catawba Nuclear Station (Table 3.3-3)
- Chemical & Volume Control System – McGuire Nuclear Station (Table 3.3-4)
- Chemical & Volume Control System – Catawba Nuclear Station (Table 3.3-5)
- Component Cooling System – McGuire Nuclear Station (Table 3.3-6)
- Component Cooling System – Catawba Nuclear Station (Table 3.3-7)
- Condenser Circulating Water System (Table 3.3-8)
- Control Area Chilled Water Systems – McGuire Nuclear Station (Table 3.3-9)
- Control Area Chilled Water Systems – Catawba Nuclear Station (Table 3.3-10)
- Control Area Ventilation (Table 3.3-11)
- Conventional Wastewater Treatment System (Table 3.3-12)
- Diesel Building Ventilation System (Table 3.3-13)
- Diesel Generator Air Intake and Exhaust System (Table 3.3-14)
- Diesel Generator Cooling Water System – McGuire Nuclear Station (Table 3.3-15)
- Diesel Generator Cooling Water System – Catawba Nuclear Station (Table 3.3-16)
- Diesel Generator Crankcase Vacuum System (Table 3.3-17)
- Diesel Generator Fuel Oil System – McGuire Nuclear Station (Table 3.3-18)
- Diesel Generator Fuel Oil System – Catawba Nuclear Station (Table 3.3-19)
- Diesel Generator Lube Oil System – McGuire Nuclear Station (Table 3.3-20)
- Diesel Generator Lube Oil System – Catawba Nuclear Station (Table 3.3-21)
- Diesel Generator Room Sump Pump System (Table 3.3-22)
- Diesel Generator Starting Air System – McGuire Nuclear Station (Table 3.3-23)
- Diesel Generator Starting Air System – Catawba Nuclear Station (Table 3.3-24)
- Drinking Water System – Catawba Nuclear Station (Table 3.3-25)
- Fire Protection System – McGuire Nuclear Station (Table 3.3-26)
- Fire Protection System – Catawba Nuclear Station (Table 3.3-27)
- Fuel Handling Area Ventilation System – Catawba Nuclear Station (Table 3.3-28)
- Fuel Handling Building Ventilation System – McGuire Nuclear Station (Table 3.3-28)
- Groundwater Drainage System (Table 3.3-29)

- Heating Water System – McGuire Nuclear Station (Table 3.3-3)
- Hydrogen Bulk Storage System (Table 3.3-30)
- Instrument Air System (Table 3.3-31)
- Liquid Radwaste System (Table 3.3-32)
- Liquid Waste Recycle System (Table 3.3-32)
- Liquid Waste Monitor and Disposal System (Table 3.3-32)
- Miscellaneous Structures Ventilation System (Table 3.3-33)
- Nitrogen System (Table 3.3-34)
- Nuclear Sampling System (Table 3.3-35)
- Nuclear Service Water System – McGuire Nuclear Station (Table 3.3-36)
- Nuclear Service Water System – Catawba Nuclear Station (Table 3.3-37)
- Nuclear Service Water Pump Structure Ventilation System (Table 3.3-38)
- Nuclear Solid Waste Disposal System (Table 3.3-39)
- Reactor Coolant Pump Motor Oil Collection Sub-System (Table 3.3-40)
- Reactor Coolant System (Non-Class 1 Components) (Table 3.3-41)
- Recirculated Cooling Water System – Catawba Nuclear Station (Table 3.3-42)
- Solid Radwaste System (Table 3.3-39)
- Spent Fuel Cooling System (Table 3.3-43)
- Standby Shutdown Diesel (Table 3.3-44)
- Turbine Building Sump Pump System (Table 3.3-45)
- Turbine Building Ventilation System (Table 3.3-46)
- Waste Gas System (Table 3.3-47)

3.3.1 Aging Management Review Results Tables

The results of the aging management review for each system of this section is provided a table, as indicated above. Information contained in each table was obtained in the following manner:

Column 1 – The component types listed in Column 1 were identified through the screening methodology described in Section 2.1.2 of this application and are on the marked plant drawings identified in Section 2.3.4 of this application.

Column 2 – The component functions listed in Column 2 were obtained from plant specific engineering documents using the screening methodology described in Section 2.1.2.

Column 3 – The materials listed in Column 3 were obtained from the drawings identified in Section 2.3.4 of this application and other plant specific engineering documents.

Column 4 – The internal and external environments listed in Column 4 were obtained from plant specific engineering documents. External environments are also noted on the drawings identified in Section 2.3.4 of this application. These environments are as follows:

- **Air-Gas –** Compressed air is ambient air that has been filtered and compressed for use in plant equipment. Compressed air may either be either dry or oiled. Compressed gases include carbon dioxide, hydrogen, nitrogen, Freon, or refrigeration gases used to replace Freon due to environmental concerns.

- **Borated Water –** Borated water is demineralized water treated with boric acid.

- **Embedded Environment –** A component encased in concrete is in an embedded environment. The concrete forms a tight seal around the external surfaces of the component.

- **Oil and Fuel Oil –** Lubricating oil is an organic fluid used to reduce friction between moving parts. Fuel oil is the fuel used for the emergency diesel generators.

- **Raw Water –** Raw water is water from a lake, pond, or river that has been rough-filtered and possibly treated with a biocide.

- **Reactor Building –** The Reactor Building environment atmosphere is a moist air environment. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry.

- **Sheltered Environment –** The ambient conditions within the sheltered environment may or may not be controlled. The sheltered environment atmosphere is a moist air environment. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry.

- **Treated water –** Treated water is demineralized water that may be deareated, treated with a biocide or corrosion inhibitors, or a combination of these treatments. Treated water does not include borated water, which is separately evaluated.

- **Underground Environment –** Components in an underground environment are in contact with soil and possibly groundwater. Components located underground are normally coated and wrapped to prevent the soil and groundwater from contacting the surface of the component.

- **Ventilation –** Ambient air that is conditioned to maintain a suitable environment for equipment operation and personnel occupancy.

- **Yard –** Yard environment is a moist air environment in which equipment is exposed to heat, cold, and precipitation.

Column 5 – The aging effects listed in Column 5 were obtained using the following aging effects identification process. The aging effects that require management during the period of extended operation have been determined by reviewing the plant-specific materials of construction (Column 3) and operating environments (Column 4) for each structure and component (Column 1) that is subject to an aging management review.

To provide reasonable assurance that the aging effects that require management for a specific material-environment combination are the only aging effects of concern for McGuire and Catawba, Duke also has performed a review of industry experience and NRC generic communications relative to these structures and components. Finally, relevant McGuire and Catawba operating experience have been reviewed to provide further confidence that the set of aging effects for the specific material-environment combinations have been identified. Taken together, the steps of this methodology provide reasonable assurance that the aging effects that require management during the period of extended operation for McGuire and Catawba structures and components have been identified.

This aging effects identification process is consistent with that process that had been used in Section 3.5 of the Oconee Nuclear Station license renewal application. Furthermore, in NUREG-1723, the staff concluded that based on its review of the information provided in Sections 3.5.1 and 3.5.2 of the Oconee application, "the applicant has identified the aging effects that are associated with mechanical systems components reviewed in [Section 3.5]." This aging effects identification process provides reasonable assurance that the aging effects that require management during the period of extended operation have been identified.

Column 6 – The aging management programs and activities listed in Column 6 are credited to manage the effects of aging for the period of extended operation.

3.3.2 AGING MANAGEMENT PROGRAMS

The following aging management programs and activities are credited to manage the effects of aging for the auxiliary systems listed in Section 3.3:

- Chemistry Control Program *
- Fire Protection Program*
- Flow Accelerated Corrosion Program *
- Fluid Leak Management Program *
- Galvanic Susceptibility Inspection *
- Heat Exchanger Preventive Maintenance Activities – Component Cooling
- Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water
- Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air
- Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units
- Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers
- Heat Exchanger Preventive Maintenance – Control Area Chilled Water
- Inspection Program for Civil Engineering Structures and Components *
- Liquid Waste System Inspection
- Preventive Maintenance Activities – Condenser Circulating Water Systems Internal Coatings Inspection
- Performance Testing Activities – Component Cooling Heat Exchangers
- Performance Testing Activities – Diesel Engine Cooling Water Heat Exchanger
- Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchanger
- Selective Leaching Inspection*
- Service Water Piping Corrosion Program*
- Sump Pump System Inspection
- Treated Water Systems Stainless Steel Inspection*
- Waste Gas System Inspection

** This aging management program/activity is equivalent to the corresponding program/activity that has been previously reviewed and found acceptable by the NRC staff during the Oconee License Renewal review, as documented in NUREG-1723.*

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in Tables 3.3-1 through 3.3-40 will be maintained consistent with the current licensing basis for the period of extended operation.

Table 3.3-1 Aging Management Review Results – Auxiliary Building Ventilation System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Air Flow Monitors	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Containment Spray Pump Motor Air Handling Unit (Plenum Assembly) (MNS Only)	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Containment Spray Pump Motor Air Handling Unit (Tubes) (MNS Only)	HT, PB	Cu	Raw Water	Fouling	Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units
			Ventilation	None Identified	None Required
Ductwork	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Filter	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-1 Aging Management Review Results – Auxiliary Building Ventilation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Fuel Pool Cooling Pump Air Handling Unit (Plenum Assembly) (MNS Only)	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Fuel Pool Cooling Pump Air Handling Unit (Tubes) (MNS Only)	HT, PB	Cu	Raw Water	Fouling	Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units
			Ventilation	None Identified	None Required
Residual Heat Removal Pump Motor Air Handling Unit (Plenum Assembly) (MNS Only)	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Residual Heat Removal Pump Motor Air Handling Unit (Tubes) (MNS Only)	HT, PB	Cu	Raw Water	Fouling	Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Motor Air Handling Units
			Ventilation	None Identified	None Required

Table 3.3-1 Aging Management Review Results – Auxiliary Building Ventilation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pump Room Heater-Demister (CNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Shutdown Panel Area Air Conditioning Unit Condenser (tubes) (CNS Only)	HT, PB	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Gas (Freon-22)	None Identified	None Required
Shutdown Panel Area Air Conditioning Unit Condenser (tube sheets) (CNS Only)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Gas (Freon-22)	None Identified	None Required
Shutdown Panel Area Air Conditioning Unit Condenser (shells) (CNS Only)	PB	CS	Gas (Freon-22)	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-1 Aging Management Review Results – Auxiliary Building Ventilation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Shutdown Panel Area Air Conditioning Unit Condenser (tube-side bonnet) (CNS Only)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Shutdown Panel Area Air Handling Unit (CNS Only)	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Shutdown Panel Area Heaters (CNS Only)	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	Cu	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-1 Aging Management Review Results – Auxiliary Building Ventilation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program

Notes for Table 3.3-1 Aging Management Review Results – Auxiliary Building Ventilation System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CS Carbon Steel
Cu Copper
Cu-Ni Copper-Nickel
GS Galvanized Steel
SS Stainless Steel

Table 3.3-2 Aging Management Review Results – Boron Recycle System
(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Eductors (MNS)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Filters	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Flow Meters	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices (CNS)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-2 Aging Management Review Results – Boron Recycle System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe (MNS Only)	PB	CS	Air-Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-2 Aging Management Review Results – Boron Recycle System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Recycle Evaporative Feed Demineralizers	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Recycle Holdup Tanks	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Strainers (CNS)	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-2 Aging Management Review Results – Boron Recycle System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies (MNS Only)	PB	CS	Air-Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Air-Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-2 Aging Management Review Results – Boron Recycle System:

(1) Component Function

HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS	Carbon Steel
SS	Stainless Steel

(3) Component subject to alternate wetting and drying which may concentrate contaminants

Table 3.3-3 Aging Management Review Results – Building Heating Water System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies (CNS only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-3 Aging Management Review Results – Building Heating Water System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel
SS Stainless Steel

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Boric Acid Blenders	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boric Acid Filters	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boric Acid Tank	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boric Acid Tank	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Boric Acid Transfer Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boron Meters	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Cation Bed Demineralizer Resin Traps	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Cation Bed Demineralizers	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Centrifugal Charging Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Excess Letdown Heat Exchangers (channel head)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Excess Letdown Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Excess Letdown Heat Exchangers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Excess Letdown Heat Exchanger (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Flow Meters (Turbine Meters)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Letdown Heat Exchangers (channel head)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Letdown Heat Exchanger (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Letdown Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Letdown Heat Exchangers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Mixed Bed Demineralizers	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Mixed Bed Demineralizer Resin Traps	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifice	PB, TH	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Reactor Coolant Filters	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Accumulators (non-wetted)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Accumulators (wetted)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Suction Stabilizers	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Regenerative Heat Exchangers (shell)	PB	CASS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Regenerative Heat Exchangers (channel head)	PB	CASS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Regenerative Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Regenerative Heat Exchangers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Regenerative Heat Exchangers (Interconnecting Piping)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Seal Water Heat Exchangers (channel head)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Seal Water Heat Exchanger (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Seal Water Heat Exchanger (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Seal Water Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Seal Water Injection Filters	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Seal Water Return Filters	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Spray Nozzles (Volume Control Tank)	SP, PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Gas	None Identified	None Required
Standby Makeup Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Standby Makeup Pump Filters	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Standby Makeup Pump Pulsation Dampener (non-wetted)	PB	CS	Gas	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Standby Makeup Pump Pulsation Dampener (wetted)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Standby Makeup Pump Suction Dampener (non-wetted)	PB	SS	Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Standby Makeup Pump Suction Dampener (wetted)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Volume Control Tanks	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Volume Control Tanks	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-4 Aging Management Review Results – Chemical & Volume Control System (McGuire Nuclear Station):

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
SP Provide spray flow so that sufficient spray flow and/or flow pattern are maintained.
TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CASS Cast Austenitic Stainless Steel (SA351-CF8)
CS Carbon Steel
SS Stainless Steel

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Boric Acid Blenders	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boric Acid Recirculating Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boric Acid Tank	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boric Acid Tank	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Boric Acid Transfer Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Boron Meters	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Cation Bed Demineralizers	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Centrifugal Charging Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Excess Letdown Heat Exchangers (channel head)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Excess Letdown Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Excess Letdown Heat Exchangers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Excess Letdown Heat Exchanger (shell)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Filters	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Flow Meters	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Letdown Heat Exchangers (channel head)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Letdown Heat Exchanger (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Letdown Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Letdown Heat Exchangers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Mixed Bed Demineralizers	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifice	PB, TH	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB, TH	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Orifices	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Reciprocating Charging Pump Discharge Pulsation Dampeners (bellows exterior)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Discharge Pulsation Dampeners (bellows interior)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Suction Pulsation Dampeners (bellows exterior)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reciprocating Charging Pump Suction Pulsation Dampeners (bellows interior)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Regenerative Heat Exchangers (channel head)	PB	CASS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Regenerative Heat Exchangers (interconnecting piping)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Regenerative Heat Exchangers (shell)	PB	CASS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Regenerative Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Regenerative Heat Exchangers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Seal Water Heat Exchangers (channel head)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Seal Water Heat Exchanger (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Seal Water Heat Exchangers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Seal Water Heat Exchangers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Standby Makeup Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Standby Makeup Pump Discharge Strainer	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Standby Makeup Suction Pulsation Dampener (bellows exterior)	PB	SS	Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Standby Makeup Suction Pulsation Dampener (bellows interior)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Strainers (simplex)	PB, FI	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Unit 1 Standby Makeup Discharge Pulsation Dampener (bellows exterior)	PB	SS	Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Unit 1 Standby Makeup Discharge Pulsation Dampener (bellows interior)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Unit 2 Standby Makeup Discharge Pulsation Dampener (bellows exterior)	PB	CS	Gas	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Unit 2 Standby Makeup Discharge Pulsation Dampener (bellows interior)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Volume Control Tank Spray Nozzles	SP, PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Volume Control Tanks	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Volume Control Tanks	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-5 Aging Management Review Results – Chemical & Volume Control System (Catawba Nuclear Station):

(1) Component Function

FI	Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
SP	Provide spray flow so that sufficient spray flow and/or flow pattern are maintained.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CASS	Cast Austenitic Stainless Steel (SA351-CF8)
CS	Carbon Steel
SS	Stainless Steel

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Flexible Hose	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Flexible Hose	PB	Inconel 625	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Heat Exchanger, KC (tubes)	PB, HT	Admiralty Brass	Raw Water	Fouling	Performance Testing Activities – Component Cooling Heat Exchanger
				Loss of Material	Heat Exchanger Preventive Maintenance Activities - Component Cooling
			Treated Water	Cracking	Chemistry Control Program
				Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, KC (tube sheet)	PB	CS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities - Component Cooling
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, KC (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, KC (channel head)	PB	CS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities - Component Cooling
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB evaporator package evaporator condenser (tubes)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB evaporator package evaporator condenser (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB evaporator package evaporator condenser (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB evaporator package distillate cooler (tubes)	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB evaporator package distillate cooler (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB evaporator package distillate cooler (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB evaporator package vent condenser (tubes)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB evaporator package vent condenser (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB evaporator package vent condenser (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NC pump motor upper bearing oil cooler (tubes)	PB	Cu	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Heat Exchanger, NC pump motor upper bearing oil cooler (tube sheet)	PB	Tube side – Cu alloy	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
		Shell side – CS	Oil	None Identified	None Required

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NC pump motor upper bearing oil cooler (shell)	PB	CS	Oil	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NC pump motor upper bearing oil cooler (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NC pump motor lower bearing oil cooler (tubes)	PB	Cu-Ni	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Heat Exchanger, WL reactor coolant drain tank (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, WL reactor coolant drain tank (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, WL reactor coolant drain tank (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NM steam generator blowdown sample (tubes)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NM steam generator blowdown sample (manifold)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NM steam generator blowdown sample (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NM pressurizer sample (tubes)	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NM pressurizer sample (manifold)	PB	SS	Borated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
Heat Exchanger, NM pressurizer sample (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NM residual heat removal loop sample (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NM residual heat removal loop sample (manifold)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NM residual heat removal loop sample (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NM reactor coolant hot leg sample (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NM reactor coolant hot leg sample (manifold)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NM reactor coolant hot leg sample (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, WG compressor package (tubes)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, WG compressor package (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, WG compressor package (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, WL waste evaporator package vent condenser (tubes)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Liquid Waste System Inspection
				Cracking	Liquid Waste System Inspection
Heat Exchanger, WL waste evaporator package vent condenser (tube sheet)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Treated Water	Loss of Material	Liquid Waste System Inspection
				Cracking	Liquid Waste System Inspection

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, WL waste evaporator package vent condenser (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, WL waste evaporator package distillate cooler (tubes)	PB	SS	Treated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, WL waste evaporator package distillate cooler (tube sheet)	PB	SS	Treated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, WL waste evaporator package distillate cooler (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, WL waste evaporator package evaporator condenser (tubes)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection
Heat Exchanger, WL waste evaporator package evaporator condenser (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, WL waste evaporator package evaporator condenser (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Orifice	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifice	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pump Casing, KC	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Tank, KC Surge	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tank, KC Surge	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tank, KC Surge	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water (Alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-6 Aging Management Review Results – Component Cooling System (McGuire Nuclear Station):

(1) **System Abbreviations -** The following system abbreviations are used in the component type designations:
KC – Component Cooling Water System
NB – Boron Recycle System
NC – Reactor Coolant System
NM – Nuclear Sampling System
WG – Waste Gas System
WL – Liquid Waste Recycle System

(2) **Component Function**

HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(3) **Material**

CS	Carbon Steel
CU	Copper
Cu-Ni	Copper-Nickel Alloy
SS	Stainless Steel

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Annubar Tube	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Flexible Hose	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Flexible Hose	PB	Inconel 625	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Heat Exchanger, CA Pump Motor Cooler (tubes)	PB, HT	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
				Fouling	Chemistry Control Program
			Ventilation	None Identified	None Required
Heat Exchanger, CA Pump Motor Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, CA Pump Motor Cooler (shell)	PB	CS	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, CA Pump Motor Cooler (channel head)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Heat Exchanger, KC – 1A, 2B (tubes)	PB, HT	SS	Raw Water	Fouling	Performance Testing Activities – Component Cooling Heat Exchangers
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Component Cooling
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, KC – 1B, 2A (tubes)	PB, HT	Admiralty Brass	Raw Water	Fouling	Performance Testing Activities – Component Cooling Heat Exchangers
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Component Cooling
			Treated Water	Cracking	Chemistry Control Program
				Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, KC (tube sheet)	PB	CS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Component Cooling
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, KC (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, KC (channel head)	PB	CS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Component Cooling
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, KC Pump Motor Cooler (tubes)	PB, HT	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
				Fouling	Chemistry Control Program
			Ventilation	None Identified	None Required
Heat Exchanger, KC Pump Motor Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, KC Pump Motor Cooler (shell)	PB	CS	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, KC Pump Motor Cooler (channel head)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Heat Exchanger, KF Pump Motor Cooler (tubes)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
Heat Exchanger, KF Pump Motor Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	Function (Note 2)	Material (Note 3)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Heat Exchanger, KF Pump Motor Cooler (shell)	PB	CS	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, KF Pump Motor Cooler (channel head)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Heat Exchanger, NB Evaporator Concentrates (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB Evaporator Concentrates (manifold)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB Evaporator Concentrates (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB Evaporator Concentrates Pump Bearing Coolers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB Evaporator Concentrates Pump Bearing Coolers (manifold)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB Evaporator Concentrates Pump Bearing Coolers (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB Evaporator Concentrates Sample Coolers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB Evaporator Concentrates Sample Coolers (manifold)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB Evaporator Package Evaporator Condensers (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB Evaporator Concentrates Sample Coolers (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB Evaporator Package Evaporator Condensers (tubes)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB Evaporator Package Evaporator Condensers (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB Evaporator Package Distillate Coolers (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB Evaporator Package Distillate Coolers (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB Evaporator Package Distillate Coolers (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NB Evaporator Package Ventilation Condensers (tubes)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, NB Evaporator Package Ventilation Condensers (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NB Evaporator Package Ventilation Condensers (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NC Pump Motor Upper Bearing Oil Cooler (tubes)	PB	Cu	Treated Water	Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Heat Exchanger, NC Pump Motor Upper Bearing Oil Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Oil	None Identified	None Required
Heat Exchanger, NC Pump Motor Upper Bearing Oil Cooler (shell)	PB	CS	Oil	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NC Pump Motor Upper Bearing Oil Cooler (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NC Pump Motor Lower Bearing Oil Cooler (tubes)	PB	Cu	Treated Water	Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Heat Exchanger, ND Pump Motor Cooler (tubes)	PB, HT	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, ND Pump Motor Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, ND Pump Motor Cooler (shell)	PB	CS	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, ND Pump Motor Cooler (channel head)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NS Pump Motor Cooler (tubes)	PB, HT	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
Heat Exchanger, NS Pump Motor Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NS Pump Motor Cooler (shell)	PB	CS	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components Fluid Leak Management Program
Heat Exchanger, NS Pump Motor Cooler (channel head)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Heat Exchanger, NI Pump Motor Cooler (tubes)	PB, HT	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NI Pump Motor Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NI Pump Motor Cooler (shell)	PB	CS	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NI Pump Motor Cooler (channel head)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Heat Exchanger, NV Pump Motor Cooler (tubes)	PB, HT	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NV Pump Motor Cooler (tube sheet)	PB	Cu-Ni (tube side)	Treated Water	Loss of Material	Chemistry Control Program
		CS (shell side)	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NV Pump Motor Cooler (shell)	PB	CS	Ventilation	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, NV Pump Motor Cooler (channel head)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Heat Exchanger, NV Centrifugal Charging Pump Bearing Oil Coolers (tubes)	PB, HT	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NV Centrifugal Charging Pump Bearing Oil Coolers (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Heat Exchanger, NV Centrifugal Charging Pump Bearing Oil Coolers (shell)	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Heat Exchanger, NV Centrifugal Charging Pump Bearing Oil Coolers (channel head)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Heat Exchanger, NV Centrifugal Charging Pump Speed Reducer Oil Coolers (tubes)	PB, HT	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NV Centrifugal Charging Pump Speed Reducer Oil Coolers (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Heat Exchanger, NV Centrifugal Charging Pump Speed Reducer Oil Coolers (shell)	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Heat Exchanger, NV Centrifugal Charging Pump Speed Reducer Oil Coolers (channel head)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Heat Exchanger, NI Pump Bearing Oil Coolers (tubes)	PB, HT	Cu-Ni	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Heat Exchanger, NI Pump Bearing Oil Coolers (tube sheet)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, NI Pump Bearing Oil Coolers (shell)	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Heat Exchanger, NI Pump Bearing Oil Coolers (channel head)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Heat Exchanger, WL Reactor Coolant Drain Tank (tubes)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchanger, WL Reactor Coolant Drain Tank (tube sheet)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchanger, WL Reactor Coolant Drain Tank (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Orifices	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil (Note 4)	None Identified	None Required
Pump Casing, KC	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Tank, KC Surge	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tank, KC Surge	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tank, KC Surge	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type (Note 1)	**Function (Note 2)**	**Material (Note 3)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water (alternate wet/dry)	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil (Note 4)	None Identified	None Required

Notes for Table 3.3-7 Aging Management Review Results – Component Cooling System (Catawba Nuclear Station)

(1) **System Abbreviations -** The following system abbreviations are used in the component type designations:
CA – Auxiliary Feedwater System
KC – Component Cooling Water System
KF – Fuel Pool Cooling System
NB – Boron Recycle System
NC – Reactor Coolant System
ND – Residual Heat Removal System
NI – Safety Injection System
NM – Nuclear Sampling System
NS – Containment Spray System
NV – Chemical & Volume Control System
WL – Liquid Waste Recycle System

(2) **Component Function**

HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(3) **Material**

CS Carbon Steel
CU Copper
Cu-Ni Copper-Nickel Alloy
SS Stainless Steel

(4) Plug valve located inside oil enclosure for RCP motor upper bearing oil cooler.

Table 3.3-8 Aging Management Review Results – Condenser Circulating Water System

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Embedded	None Identified	None Required
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe (Catawba only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Underground	Loss of Material	Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection

Table 3.3-8 Aging Management Review Results – Condenser Circulating Water System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe (Catawba only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pump Casing (Catawba Only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Strainer (Catawba only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-8 Aging Management Review Results – Condenser Circulating Water System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Valve Bodies (Catawba only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-8 Aging Management Review Results – Condenser Circulating Water System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Chemical Feeders	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Chemical Feeders	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Chemical Feeders	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Chemical Feeders	PB	CI	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Area Chilled Water Pump Casings	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Area Chilled Water Pump Casings	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Condenser Tubes)	HT, PB	Cu-Ni	Raw Water	Fouling	Heat Exchanger Preventive Maintenance Activities-Control Area Chilled Water
				Loss of Material	Heat Exchanger Preventive Maintenance Activities-Control Area Chilled Water
			Gas	None Identified	None Required

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Condenser Tube Sheets)	PB	Cu-Ni	Raw Water	Loss of Material	Service Water Piping Corrosion Program
		CS	Gas	None Identified	None Required
Control Room Area Chiller (Condenser Shells)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Condenser Channel Heads)	PB	CS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities-Control Area Chilled Water
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Economizers)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Evaporator Tubes)	HT, PB	Cu	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Gas	None Identified	None Required
Control Room Area Chiller (Evaporator Tube Sheets)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Gas	None Identified	None Required
Control Room Area Chiller (Evaporator Channel Heads)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Evaporator Shells)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Oil Cooler Tubes)	PB	AB	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Control Room Area Chiller (Oil Cooler Tube Sheets)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Control Room Area Chiller (Oil Cooler Channel Heads)	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Oil Cooler Shells)	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Oil Filters)	PB	CI	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Oil Filters)	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Oil Separators)	FI, PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Pump Out Condenser Tubes)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Gas	None Identified	None Required
Control Room Area Chiller (Pump Out Condenser Tube Sheets)	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
		CS	Gas	None Identified	None Required
Control Room Area Chiller (Pump Out Condenser Channel Heads)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Pump Out Condenser Shells)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Storage Tanks)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Compression Tanks	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Compression Tanks	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Flow Indicators	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Orifices	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Strainers	FI, PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-9 Aging Management Review Results – Control Area Chilled Water System (McGuire Nuclear Station):

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

AB Admiralty Brass
CI Cast Iron
CS Carbon Steel
Cu Copper
Cu-Ni Copper-Nickel Alloy
SS Stainless Steel

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chilled Water Pump Casings	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chilled Water Pump Casings	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Compression Tanks	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Compression Tanks	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Inspection For Civil Engineering Structures and Components
Control Room Area Chiller (Condenser tubes)	HT, PB	Cu-Ni	Raw Water	Fouling	Heat Exchanger Preventive Maintenance – Control Area Chilled Water
				Loss of Material	Heat Exchanger Preventive Maintenance – Control Area Chilled Water
			Gas	None Identified	None Required
Control Room Area Chiller (Condenser tube sheets)	PB	Cu-Ni	Raw Water	Loss of Material	Service Water Piping Corrosion Program
		CS	Gas	None Identified	None Required
Control Room Area Chiller (Condenser shells)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Condenser channel heads)	PB	CS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance – Control Area Chilled Water
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Economizers)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Evaporator tubes)	HT, PB	Cu	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Gas	None Identified	None Required
Control Room Area Chiller (Evaporator tube sheets)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Gas	None Identified	None Required

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Evaporator Channel Heads)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Evaporator shells)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Oil Cooler tubes)	PB	AB	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Oil Cooler tube sheets)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Control Room Area Chiller (Oil Cooler Channel Heads)	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Oil Cooler Shells)	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Oil Filters)	FI, PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Oil Separators)	FI, PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Pump Out Condenser tubes)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
			Gas	None Identified	None Required
Control Room Area Chiller (Pump Out Condenser tube sheets)	PB	Cu-Ni	Treated Water	Loss of Material	Chemistry Control Program
		CS	Gas	None Identified	None Required
Control Room Area Chiller (Pump Out Condenser Channel Heads)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Control Room Area Chiller (Pump Out Condenser Shells)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Control Room Area Chiller (Refrigerant Filters)	FI, PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Control Room Area Chiller (Storage Tanks)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Orifices	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Orifices	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Y-Strainers	FI, PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-10 Aging Management Review Results – Control Area Chilled Water System (Catawba Nuclear Station)

:

(1)	**Component Function**
FI	Provide filtration of process fluid so that downstream equipment and/or environments are protected.
HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2)	**Material**
AB	Admiralty Brass
CI	Cast Iron
CS	Carbon Steel
Cu-Ni	Copper-Nickel Alloy
SS	Stainless Steel

Table 3.3-11 Aging Management Review Results – Control Area Ventilation System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Air Handling Units (Heat Exchangers) (shells)	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Air Handling Units (Heat Exchangers) (tube sheets)	PB	SS – MNS CS – CNS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
Air Handling Units (Heat Exchangers) (tubes) (CNS Only)	HT, PB	Cu	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
Air Handling Units (Heat Exchangers) (tubes) (MNS Only)	HT, PB	SS	Treated Water	Cracking	Chemistry Control Program
				Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
Control Room Area Pressurizing Filter Trains (CNS Only)	FI, PB	SS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Ductwork	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program

Table 3.3-11 Aging Management Review Results – Control Area Ventilation System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note1)	Material (Note2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Ductwork (CNS Only)	PB	GS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Filter Trains (MNS Only)	FI, PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Orifices (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pre-Filters (MNS Only)	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	Cu	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program

Table 3.3-11 Aging Management Review Results – Control Area Ventilation System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-11 Aging Management Review Results – Control Area Ventilation System:

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CS Carbon Steel
Cu Copper
GS Galvanized Steel
SS Stainless Steel

Table 3.3-12 Aging Management Review Results – Conventional Wastewater Treatment System (McGuire Nuclear Station Only)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump Systems Inspection
			Embedded	None Identified	None Required
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump Systems Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Standby Shutdown Facility Sump Pump Casing	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Selective Leaching Inspection Sump Pump Systems Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-12 Aging Management Review Results – Conventional Wastewater Treatment System (McGuire Nuclear Station Only) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump Systems Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-12 Aging Management Review Results – Conventional Wastewater Treatment System (McGuire Nuclear Station Only):

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel
CI Cast Iron

Table 3.3-13 Aging Management Review Results – Diesel Building Ventilation System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Ductwork	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	BR	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	Cu	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-13 Aging Management Review Results – Diesel Building Ventilation System:

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2)	**Material**
BR	Brass
CS	Carbon Steel
Cu	Copper
GS	Galvanized Steel
SS	Stainless Steel

Table 3.3-14 Aging Management Review Results – Diesel Generator Air Intake and Exhaust System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Exhaust Silencers	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Intake Air Filters (CNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Intake Air Silencers	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Intake Flexible Connector (MNS Only)	PB	Rubber	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Expansion Joints	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Flexible Hoses (CNS Only)	PB	CR	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-14 Aging Management Review Results – Diesel Generator Air Intake and Exhaust System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies (CNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-14 Aging Management Review Results – Diesel Generator Air Intake and Exhaust System:

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2)	**Material**
CR	Composite Rubber (ethylene propylene)
CS	Carbon Steel
Rubber	Rubber 45 Neoprene and Cloth
SS	Stainless Steel

Table 3.3-15 Aging Management Review Results – Diesel Generator Cooling Water System (McGuire Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Annubars	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
D/G Cooling Water Surge Tanks	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
				Loss of Material (Note 3)	
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Cooling Water Surge Tanks	PB	CS	Ventilation	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Cooling Water Heat Exchangers (channel heads)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Cooling Water Heat Exchangers (shells)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-15 Aging Management Review Results – Diesel Generator Cooling Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Cooling Water Heat Exchangers (tubes)	HT, PB	Cu	Raw Water	Fouling	Performance Test Activity – Diesel Engine Cooling Water Heat Exchanger
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water
			Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
D/G Engine Cooling Water Heat Exchangers (tube sheets)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
D/G Engine Cooling Water Turbocharger Intercoolers (channel heads)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Cooling Water Turbocharger Intercoolers (shells)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-15 Aging Management Review Results – Diesel Generator Cooling Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Cooling Water Turbocharger Intercoolers (tubes)	HT, PB	Cu	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
D/G Engine Cooling Water Turbocharger Intercoolers (tube sheets)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
D/G Intercooler Pumps	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Jacket Water Circulating Pumps	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Jacket Water Heaters	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-15 Aging Management Review Results – Diesel Generator Cooling Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Jacket Water Pumps	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Flow Orifices	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Piping	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Loss of Material (Note 3)	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Piping	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-15 Aging Management Review Results – Diesel Generator Cooling Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-15 Aging Management Review Results – Diesel Generator Cooling Water System (McGuire Nuclear Station):

(1) Component Function

HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CI Cast Iron
Cu Copper
SS Stainless Steel

(3) Component subject to alternate wetting and drying which may concentrate contaminants

Table 3.3-16 Aging Management Review Results – Diesel Generator Cooling Water System (Catawba Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Driven Jacket Water Circulation Pumps	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Jacket Water Coolers (channel heads)	PB	CS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Jacket Water Coolers (shells)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-16 Aging Management Review Results – Diesel Generator Cooling Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment** **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Jacket Water Coolers (tubes)	HT, PB	BR	Raw Water	Fouling	Performance Testing Activities – Diesel Generator Engine Cooling Water Heat Exchangers
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Cooling Water
			Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
D/G Engine Jacket Water Coolers (tube sheets)	PB	CS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
D/G Engine Jacket Water Keep Warm Pumps	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
D/G Engine Jacket Water Standpipes	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Loss of Material (Note 3)	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-16 Aging Management Review Results – Diesel Generator Cooling Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Jacket Water Standpipes	PB	CS	Ventilation	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Governor Lube Oil Coolers (end covers)	PB	AL	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
D/G Governor Lube Oil Coolers (shells)	PB	AL	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
D/G Governor Lube Oil Coolers (tubes)	HT, PB	BR	Treated Water	Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
Piping	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-16 Aging Management Review Results – Diesel Generator Cooling Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-16 Aging Management Review Results – Diesel Generator Cooling Water System (Catawba Nuclear Station):

(1) Component Function

HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

AL	Aluminum
BR	Brass
CI	Cast Iron
CS	Carbon Steel
SS	Stainless Steel

(3) Component subject to alternate wetting and drying which may concentrate contaminants

Table 3.3-17 Aging Management Review Results – Diesel Generator Crankcase Vacuum System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Diesel Generator Crankcase Vacuum Blowers (MNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Diesel Generator Crankcase Vacuum Oil Separators (MNS Only)	FI	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Diesel Generator Crankcase Vacuum Oil Separators (MNS Only)	GR	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Diesel Generator Crankcase Vacuum Oil Separators (MNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-17 Aging Management Review Results – Diesel Generator Crankcase Vacuum System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Orifices (MNS Only)	PB, TH	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing (MNS Only)	PB	BR	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing (MNS Only)	PB	Cu	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-17 Aging Management Review Results – Diesel Generator Crankcase Vacuum System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies (MNS Only)	PB	BR	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies (CNS Only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-17 Aging Management Review Results – Diesel Generator Crankcase Vacuum System:

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
GR Provide gas removal so that sufficient vacuum is maintained, or to ensure sufficient fluid system level, priming, or inventory.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

BR Brass
Cu Copper
CS Carbon Steel
SS Stainless Steel

Table 3.3-18 Aging Management Review Results – Diesel Generator Fuel Oil System (McGuire Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Driven Fuel Oil Pump Casings	PB	CI	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Booster Pump Casings	PB	CI	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Day Tanks	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Day Tanks	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Duplex Filters	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Storage Tanks	PB	CS	Ventilation	None Identified	None Required
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection

Table 3.3-18 Aging Management Review Results – Diesel Generator Fuel Oil System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Fuel Oil Storage Tanks	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
D/G Fuel Oil Transfer Filters	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
D/G Fuel Oil Transfer Pump Casings	PB	CI	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Flame Arrestors	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Flow Meters	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-18 Aging Management Review Results – Diesel Generator Fuel Oil System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Pipe	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Underground	Cracking	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
Strainers	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Underground	Cracking	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection

Notes for Table 3.3-18 Aging Management Review Results – Diesel Generator Fuel Oil System (McGuire Nuclear Station):

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CI Cast Iron
CS Carbon Steel
SS Stainless Steel

Table 3.3-19 Aging Management Review Results – Diesel Generator Fuel Oil System (Catawba Nuclear Station)

(Note are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Driven Fuel Oil Pump Casings	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Driven Fuel Oil Pump Strainer Baskets	FI	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
D/G Engine Driven Fuel Oil Pump Strainer Bodies	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Fuel Oil Filters	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Motor Driven Fuel Oil Booster Pump Casings	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Motor Driven Fuel Oil Booster Pump Strainer Baskets	FI	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-19 Aging Management Review Results – Diesel Generator Fuel Oil System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
D/G Engine Motor Driven Fuel Oil Booster Pump Strainer Bodies	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Day Tanks	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Day Tanks	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Fuel Oil Storage Tanks	PB	CS	Ventilation	None Identified	None Required
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
D/G Fuel Oil Storage Tanks	PB	CS	Oil	Loss of Material	Chemistry Control Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
Flexible Hoses	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-19 Aging Management Review Results – Diesel Generator Fuel Oil System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Underground	Cracking	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
Pipe	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required
Tubing	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-19 Aging Management Review Results – Diesel Generator Fuel Oil System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Underground	Cracking	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection
Valve Bodies	PB	SS	Oil	Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required

Notes for Table 3.3-19 Aging Management Review Results – Diesel Generator Fuel Oil System (Catawba Nuclear Station):

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel
SS Stainless Steel

Table 3.3-20 Aging Management Review Results – Diesel Generator Lube Oil System (McGuire Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Before & After Lube Oil Pump Casings	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Driven Lube Oil Pump Casings	PB	Cast Iron	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Lube Oil Coolers (tube sheets)	PB	Cu-Alloy	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
D/G Lube Oil Coolers (tubes)	PB, HT	BR	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
D/G Lube Oil Coolers (tubes)	PB, HT	Cu-Ni	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
D/G Lube Oil Coolers (shells)	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-20 Aging Management Review Results – Diesel Generator Lube Oil System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Lube Oil Coolers (channel heads)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Lube Oil Engine Intake Strainers	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Lube Oil Filters	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
D/G Lube Oil Heaters	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Strainers	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-20 Aging Management Review Results – Diesel Generator Lube Oil System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Tubing	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-20 Aging Management Review Results – Diesel Generator Lube Oil System (McGuire Nuclear Station):

(1) Component Function

HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel
Cu-Alloy Muntz metal
Cu-Ni Copper Nickel Alloy
SS Stainless Steel

Table 3.3-21 Aging Management Review Results - Diesel Generator Lube Oil System (Catawba Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Driven Lube Oil Pump Casings	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Lube Oil Coolers (shell)	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Lube Oil Coolers (tube sheet)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
D/G Engine Lube Oil Coolers (tube)	HT, PB	Copper Alloy	Treated Water	Loss of Material	Chemistry Control Program
			Oil	None Identified	None Required
D/G Engine Lube Oil Coolers (channel head)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Lube Oil Filters	FI, PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-21 Aging Management Review Results - Diesel Generator Lube Oil System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Lube Oil Strainers	FI, PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Prelube Oil Filters	FI, PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Prelube Oil Pump Casings	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Prelube Oil Strainers	FI, PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Lube Oil Sump Tank Filters	FI, PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Lube Oil Sump Tanks	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Flexible Hoses	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-21 Aging Management Review Results - Diesel Generator Lube Oil System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Lube Oil Pressure Regulating Valve Strainers	FI, PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Oil	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-21 Aging Management Review Results - Diesel Generator Lube Oil System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required

Notes form Table 3.3-21 Aging Management Review Results - Diesel Generator Lube Oil System (Catawba Nuclear Station):

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected
HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel
SS Stainless Steel

Table 3.3-22 Aging Management Review Results – Diesel Generator Room Sump Pump

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Sump Room Sump Pump Casings	PB	CI – MNS CS– CNS	Raw Water	Loss of Material	Selective Leaching Inspection (MNS Only) Galvanic Susceptibility Inspection Sump Pump System Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Orifices (MNS Only)	PB	SS	Raw Water	Loss of Material	Sump Pump Systems Inspection
			Sheltered	None Identified	None Required
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump Systems Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe (CNS Only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump Systems Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-22 Aging Management Review Results – Diesel Generator Room Sump Pump (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump Systems Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Raw Water	Loss of Material	Sump Pump Systems Inspection
			Sheltered	None Identified	None Required

Notes for Table 3.3-22 Aging Management Review Results – Diesel Generator Room Sump Pump:

(1) Component Function

HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

BR Brass
CI Cast Iron
CS Carbon Steel
Cu Copper
SS Stainless Steel

(3) Component subject to alternate wetting and drying which may concentrate contaminants.

Table 3.3-23 Aging Management Review Results – Diesel Generator Starting Air System (McGuire Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Control Air Filter	PB	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required
D/G Starting Air Line Filter	PB, FI	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required
D/G Starting Air Tank	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Expansion Joints	PB	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing	PB	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-23 Aging Management Review Results – Diesel Generator Starting Air System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Valve Bodies	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-23 Aging Management Review Results – Diesel Generator Starting Air System (McGuire Nuclear Station):

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel
SS Stainless Steel

Table 3.3-24 Aging Management Review Results – Diesel Generator Starting Air System (Catawba Nuclear Station)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Afterfilters	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Starting Air Aftercoolers (tube sheets)	PB	Monel 400	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Air (moist)	None Identified	None Required
D/G Engine Starting Air Aftercoolers (tubes)	PB	SS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air
			Air (moist)	None Identified	None Required
D/G Engine Starting Air Aftercoolers (channel heads)	PB	CS	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Starting Air Aftercoolers (shells)	PB	CS	Air (moist)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-24 Aging Management Review Results – Diesel Generator Starting Air System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
D/G Engine Starting Air Compressor Inlet Filters	PB	CS	Air (moist)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
D/G Engine Starting Air Tank	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Flow Meters	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Moisture Separators	PB	CS	Air (moist)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Orifices	PB	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-24 Aging Management Review Results – Diesel Generator Starting Air System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Air (moist)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Prefilters	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Silencers	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Starting Air Distributor Filters	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-24 Aging Management Review Results – Diesel Generator Starting Air System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Tubing	PB	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Air (moist)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-24 Aging Management Review Results – Diesel Generator Starting Air System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Air (dry)	None Identified	None Required
			Sheltered	None Identified	None Required
Y-Strainers	FI, PB	CS	Air (dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-24 Aging Management Review Results – Diesel Generator Starting Air System (Catawba Nuclear Station):

(1)	**Component Function**
FI	Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2)	**Material**
CS	Carbon Steel
Monel 400	A copper nickel alloy
SS	Stainless Steel

Table 3.3-25 Aging Management Review Results – Drinking Water System

(Catawba Nuclear Station only)
(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Program s and Activities**
Pipe	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required

Notes for Table 3.3-25 Aging Management Review Results – Drinking Water System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

SS Stainless Steel

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System					
Cylinders (Halon)	PB	AS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Fire Hose Rack	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Fire Hose Rack	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Flexible Hose	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	GS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Interior Fire Protection System (continued)					
Pipe	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CU	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Pipe	PB	MI	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Pipe	PB	DI	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Pressure Switches	PB	BZ	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Rupture Discs	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Spray Nozzles	PB, SP	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Spray Nozzles	PB, SP	CS	Ventilation	None Identified	None Required
			Ventilation	None Identified	None Required
Spray Nozzles	PB, SP	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Spray Nozzles	PB, SP	SS	Ventilation	None Identified	None Required
			Ventilation	None Identified	None Required
Spray Nozzles	PB, SP	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Spray Nozzles	PB, SP	BZ	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Sprinklers	PB, SP	BR	Raw Water	Fouling	Fire Protection Program
				Loss of Material	Fire Protection Program Galvanic Susceptibility Inspection Selective Leaching Inspection
			Reactor Building	Loss of Material	Fluid Leak Management Program
Sprinklers	PB, SP	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Sprinklers	PB, SP	BZ	Raw Water	Fouling	Fire Protection Program
				Loss of Material	Fire Protection Program Galvanic Susceptibility Inspection
			Reactor Building	Loss of Material	Fluid Leak Management Program
Sprinklers	PB, SP	BZ	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Sprinklers	PB, SP	BR	Raw Water	Fouling (Note 4)	Fire Protection Program
				Loss of Material	Fire Protection Program Galvanic Susceptibility Inspection Selective Leaching Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Sprinklers	PB	BZ	Raw Water	Fouling	Fire Protection Program
				Loss of Material	Fire Protection Program Galvanic Susceptibility Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BZ	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Program
				Fouling (Note 4)	Fire Protection Program
			Reactor Building	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	BZ	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Valve Bodies	PB	BZ	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Program
				Fouling (Note 4)	Fire Protection Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BZ	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Interior Fire Protection System (continued)					
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	BR	Raw Water	Loss of Material	Selective Leaching Inspection Service Water Piping Corrosion Program Galvanic Susceptibility Inspection
				Fouling (Note 4)	Fire Protection Program
			Reactor Building	Loss Of Material	Fluid Leak Management Program
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Valve Bodies	PB	BR	Raw Water	Loss of Material	Service Water Piping Corrosion Program Galvanic Susceptibility Inspection Selective Leaching Inspection
				Fouling (Note 4)	Fire Protection Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BR	Gas	None Identified	None Required
			Sheltered (Turbine Building)	None Identified	None Required

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System					
Orifices	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Pipe	PB	GS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Pipe	PB	GS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Embedded	None Identified	None Required
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Pipe	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
Pipe	PB	DI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Pipe	PB	DI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Embedded	None Identified	None Required

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Pipe	PB	DI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Pipe	PB	DI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pulsation Dampeners	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Pump Casings (Bowls)	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
Standpipes	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Standpipes	PB	CI	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	BR	Raw Water	Loss of Material	Service Water Piping Corrosion Inspection Galvanic Susceptibility Inspection Selective Leaching Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BR	Raw Water	Loss of Material	Service Water Piping Corrosion Inspection Galvanic Susceptibility Inspection Selective Leaching Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Exterior Fire Protection System (continued)					
Valve Bodies	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Valve Bodies	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
				Fouling (Note 4)	Fire Protection Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CI	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Valve Bodies	PB	BZ	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BZ	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BZ	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	BZ	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-26 Aging Management Review Results – Fire Protection System (McGuire Nuclear Station):

(1)	**Component Function**
FI	Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
SP	Provide spray flow so that sufficient spray flow and/or flow patterns are maintained.
(2)	**Material**
AS	Alloy Steel
BR	Brass
BZ	Bronze
CI	Cast Iron
CS	Carbon Steel
DI	Ductile Iron
GS	Galvanized Steel
MI	Malleable Iron
SS	Stainless Steel
(3)	The Fluid Leak Management Program is applicable for components only within the Reactor Building or Auxiliary Building.
(4)	Fire Hose Rack Valves Only

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System					
Cylinders (CO_2)	PB	AS (Cr-Mo)	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Flexible Hose	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Fire Hose Rack	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Fire Hose Rack	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Orifices	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Pipe	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Interior Fire Protection System (continued)					
Pipe	PB	MI	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Pipe	PB	CS	Ventilation	None Identified	None Required
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Air-Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Pipe	PB	SS	Ventilation	None Identified	None Required
			Yard	None Identified	None Required
Pipe	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Spray Nozzle	PB, SP	SS	Ventilation	None Identified	None Required
			Ventilation	None Identified	None Required
Spray Nozzle	PB, SP	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Spray Nozzle	PB, SP	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Sprinkler	PB, SP	BR	Raw Water	Fouling	Fire Protection Program
				Loss of Material	Fire Protection Program Galvanic Susceptibility Inspection Selective Leaching Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Sprinkler	PB, SP	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Tanks (CO_2)	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	BR	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Valve Bodies	PB	BR	Raw Water	Fouling (Note 4)	Fire Protection Program
				Loss of Material	Galvanic Susceptibility Program Service Water Piping Corrosion Program Selective Leaching Inspection
			Reactor Building	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	BR	Raw Water	Fouling (Note 4)	Fire Protection Program
				Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3)
Valve Bodies	PB	BZ	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	BZ	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	BZ	Raw Water	Fouling (Note 4)	Fire Protection Program
				Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Valve Bodies	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Interior Fire Protection System (continued)					
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System					
Orifices	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Underground	Cracking	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Orifices	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Yard	None Identified	None Required
Pipe	PB	DI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Pipe	PB	DI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Pipe	PB	DI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Yard	None Identified	None Required
Pipe	PB	GS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pump Casing (Main Fire Pumps)	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Yard	None Identified	None Required

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Valve Bodies	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CI	Raw Water	Fouling (Note 4)	Fire Protection Program
				Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program Selective Leaching Inspection
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection

Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exterior Fire Protection System (continued)					
Valve Bodies	PB	CI	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	PVDF	Raw Water	None Identified	None Required
			Yard	None Identified	None Required

Notes for Table 3.3-27 Aging Management Review Results – Fire Protection System (Catawba Nuclear Station):

(1)	**Component Function**
FI	Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
SP	Provide spray flow so that sufficient spray flow and/or flow patterns are maintained.
(2)	**Material**
Cr-Mo	Chromium Molybdenum
AS	Alloy Steel
BR	Brass
BZ	Bronze
CI	Cast Iron
CS	Carbon Steel
DI	Ductile Iron
GS	Galvanized Steel
MI	Malleable Iron
PVDF	Polyvinyldifluoride
SS	Stainless Steel
(3)	The Fluid Leak Management Program is applicable for components only within the Reactor Building or Auxiliary Building.

Table 3.3-28 Aging Management Review Results – Fuel Handling Building Ventilation System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Air Flow Monitors	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Ductwork	PB	GS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Filter	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	Cu	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-28 Aging Management Review Results – Fuel Handling Building Ventilation System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CS Carbon Steel
Cu Copper
GS Galvanized Steel
SS Stainless Steel

Table 3.3-29 Aging Management Review Results – Groundwater Drainage System
(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pump Casings	PB	CI-MNS CS-CNS	Raw Water	Loss of Material	Selective Leaching Inspection (McGuire) Galvanic Susceptibility Inspection Sump Pump System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pump Casings	PB	CI-MNS CS-CNS	Raw Water	Loss of Material	Selective Leaching Inspection (McGuire) Galvanic Susceptibility Inspection Sump Pump System Inspection
			Raw Water	Loss of Material	Selective Leaching Inspection (McGuire) Galvanic Susceptibility Inspection Sump Pump System Inspection
Orifices (CNS Only)	PB	SS	Raw Water	Loss of Material	Sump Pump System Inspection
			Sheltered	None Identified	None Required
Pipe (CNS Only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump System Inspection
			Embedded in Concrete	None Identified	None Required

Table 3.3-29 Aging Management Review Results – Groundwater Drainage System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (CNS Only)	PB	SS	Raw Water	Loss of Material	Sump Pump System Inspection
			Sheltered	None Identified	None Required
Pipe (MNS Only)	PB	SS	Raw Water	Loss of Material	Sump Pump System Inspection
			Yard	None Identified	None Required
Tubing	PB	SS	Raw Water	Loss of Material	Sump Pump System Inspection
			Sheltered	None Identified	None Required
Valve Bodies (MNS Only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Sump Pump System Inspection
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Raw Water	Loss of Material	Sump Pump System Inspection
			Sheltered	None Identified	None Required

Notes for Table 3.3-29 Aging Management Review Results – Groundwater Drainage System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CI Cast Iron
CS Carbon Steel
SS Stainless Steel

Table 3.3-30 Aging Management Review Results – Hydrogen Bulk Storage System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe (MNS Only)	PB	SS	Gas	None Identified	None Required
			Yard	None Identified	None Required
Tubing (CNS Only)	PB	Cu	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies (CNS Only)	PB	BR	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-30 Aging Management Review Results – Hydrogen Bulk Storage System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CS Carbon Steel
Cu Copper
SS Stainless Steel

Table 3.3-31 Aging Management Review Results – Instrument Air System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Filters (Housings only) (MNS only)	PB	SS	Air	None Identified	None Required
			Sheltered	None Identified	None Required
FIV Assured VI Supply Accumulators (MNS only)	PB	SS	Air	None Identified	None Required
			Sheltered	None Identified	None Required
Instrument Air Tanks (Blackout) (MNS only)	PB	CS	Air	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Air	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	SS	Air	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe (MNS only)	PB	GS	Air	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program

Table 3.3-31 Aging Management Review Results – Instrument Air System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Tubing	PB	SS	Air	None Identified	None Required
			Reactor Building	None Identified	None Required
Tubing (MNS only)	PB	SS	Air	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing (MNS only)	PB	BR	Air	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing (MNS only)	PB	Cu	Air	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	SS	Air	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Air	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies (MNS only)	PB	CS	Air	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies (MNS only)	PB	BR	Air	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program

Notes for Table 3.3-31 Aging Management Review Results – Instrument Air System:

(1)	Component Function
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2)	Material
BR	Brass
CS	Carbon Steel
Cu	Copper
GS	Galvanized Steel
SS	Stainless Steel

Table 3.3-32 Aging Management Review Results – Liquid Waste System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Motor Driven Auxiliary Feedwater Pump Sump Pumps (CNS only)	PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Residual Heat Removal Pump and Containment Spray Pump Room Sump Pumps (CNS only)	PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Orifices (CNS only)	PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Pipe	PB	CS	Gas	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-32 Aging Management Review Results – Liquid Waste System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe (CNS only)	PB	CS	Raw Water	Loss of Material	Liquid Waste System Inspection Galvanic Susceptibility Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (Note 3)
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Borated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Pipe (CNS only)	PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required

Table 3.3-32 Aging Management Review Results – Liquid Waste System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe (CNS only)	PB	SS	Treated Water	Cracking	Chemistry Control Program (Auxiliary Steam Supply to Feed Preheater Only)
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe (CNS only)	PB	SS	Treated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Pipe (MNS only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Loss of Material (Wet/Dry)	Chemistry Control Program
				Cracking (Wet/Dry)	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe (MNS only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe (MNS only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required

Table 3.3-32 Aging Management Review Results – Liquid Waste System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe (MNS only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Separators (CNS only)	FI, PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Strainers (CNS only)	FI, PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Turbine Driven Auxiliary Feedwater Pump Sump Pumps (CNS only)	PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Tubing (CNS only)	PB	SS	Borated Water	Loss of Material	Liquid Waste System Inspection
				Cracking	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Gas	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-32 Aging Management Review Results – Liquid Waste System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies (CNS only)	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Liquid Waste System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (Note 3)
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Borated Water	Loss of Material	Liquid Waste System Inspection
				Cracking	Liquid Waste System Inspection
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Loss of Material	Liquid Waste System Inspection
				Cracking	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies (CNS only)	PB	SS	Raw Water	Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required

Table 3.3-32 Aging Management Review Results – Liquid Waste System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies (CNS only)	PB	SS	Treated Water	Cracking	Liquid Waste System Inspection
				Loss of Material	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Valve Bodies (Ventilation Unit Drains)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies (Ventilation Unit Drains)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies (Loop Seals – MNS only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Cracking (wet/dry)	Chemistry Control Program
				Loss of Material (wet/dry)	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-32 Aging Management Review Results – Liquid Waste System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Waste Drain Tank (CNS only)	PB	SS	Borated Water	Loss of Material	Liquid Waste System Inspection
				Cracking	Liquid Waste System Inspection
			Sheltered	None Identified	None Required
Waste Drain Tank (CNS only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-32 Aging Management Review Results – Liquid Waste System:

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CS Carbon Steel
Cu Copper
SS Stainless Steel

(3) The Flow Accelerated Corrosion Program applies only to the components in the steam supply line from the Auxiliary Steam System to the Evaporator Package

Table 3.3-33 Aging Management Review Results – Miscellaneous Structures Ventilation System

(Catawba Nuclear Station Only)
(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Air Handling Unit	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Ductwork	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Flexible Connectors	PB	Neoprene (Note 3)	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Plenum Section	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-33 Aging Management Review Results – Miscellaneous Structures Ventilation System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

GS Galvanized Steel

(3) Woven glass fabric double-coated neoprene.

Table 3.3-34 Aging Management Review Results – Nitrogen System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Nitrogen Supply Tanks (MNS Only)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing (MNS Only)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valves Bodies	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-34 Aging Management Review Results – Nitrogen System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

SS Stainless Steel

Table 3.3-35 Aging Management Review Results – Nuclear Sampling System

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Orifice	PB, TH	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required

Table 3.3-35 Aging Management Review Results – Nuclear Sampling System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-35 Aging Management Review Results – Nuclear Sampling System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Valve Bodies	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-35 Aging Management Review Results – Nuclear Sampling System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

SS Stainless Steel

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Centrifugal Charging Pump Bearing Oil Coolers (tubes)	HT, PB	Cu-Ni	Raw Water	Fouling	Heat Exchanger Preventive Maintenance Activities-Pump Oil Coolers
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers
			Oil	None Identified	None Required
Centrifugal Charging Pump Bearing Oil Coolers (tube sheets)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Oil	None Identified	None Required
Centrifugal Charging Pump Bearing Oil Coolers (shells)	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Centrifugal Charging Pump Bearing Oil Coolers (channel covers)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Centrifugal Charging Pump Speed Reducer Oil Coolers (tubes)	HT, PB	Cu-Ni	Raw Water	Fouling	Heat Exchanger Preventive Maintenance Activities-Pump Oil Coolers
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers
			Oil	None Identified	None Required
Centrifugal Charging Pump Speed Reducer Oil Coolers (tube sheets)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Oil	None Identified	None Required
Centrifugal Charging Pump Speed Reducer Oil Coolers (shells)	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Centrifugal Charging Pump Speed Reducer Oil Coolers (channel covers)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Expansion Joints	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Nuclear Service Water Pump Casings	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Nuclear Service Water Strainers	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Orifices	PB, TH	SS	Raw Water	Loss of Material (Note 3)	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Bearing Oil Coolers (tubes)	PB	Cu-Ni	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers
			Oil	None Identified	None Required
Reciprocating Charging Pump Bearing Oil Coolers (tube sheets)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Oil	None Identified	None Required
Reciprocating Charging Pump Bearing Oil Coolers (channel covers)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Reciprocating Charging Pump Fluid Drive Oil Coolers (tubes)	PB	Cu-Ni	Raw Water	Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers
			Oil	None Identified	None Required
Reciprocating Charging Pump Fluid Drive Oil Coolers (tube sheets)	PB	BR	Raw Water	Loss of Material	Selective Leaching Inspection Service Water Piping Corrosion Program
			Oil	None Identified	None Required

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reciprocating Charging Pump Fluid Drive Oil Coolers (channel covers)	PB	CI	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Safety Injection Pump Bearing Oil Coolers (tubes)	HT, PB	Cu-Ni	Raw Water	Fouling	Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers
				Loss of Material	Heat Exchanger Preventive Maintenance Activities – Pump Oil Coolers
			Oil	None Identified	None Required
Safety Injection Pump Bearing Oil Coolers (tube sheets)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Oil	None Identified	None Required
Safety Injection Pump Bearing Oil Coolers (shells)	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Safety Injection Pump Bearing Oil Coolers (channel covers)	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Tubing	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection

Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required

Notes for Table 3.3-36 Aging Management Review Results – Nuclear Service Water System (McGuire Nuclear Station):

(1) Component Function

HT Provide heat transfer so that system and/or component operating temperatures are maintained.

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

BR Brass
CI Cast Iron
CS Carbon Steel
Cu-Ni Copper-Nickel Alloy
SS Stainless Steel

(3) Loss of material due to corrosion will not affect the throttling function of orifices.

Table 3.3-37 Aging Management Review Results – Nuclear Service Water System (Catawba Nuclear Station)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Annubars	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Annubars	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Underground	Cracking	Preventive Maintenance Activities-Condenser Circulating Water System Internal Coating Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Flexible Hoses	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered (Note 3)	None Identified	None Required
Manways	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-37 Aging Management Review Results – Nuclear Service Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Nuclear Service Water Pump Casings	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered (Note 3)	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Orifices	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Orifices	PB, TH	SS	Raw Water	Loss of Material (Note 4)	Service Water Piping Corrosion Program
			Sheltered (Note 3)	None Identified	None Required

Table 3.3-37 Aging Management Review Results – Nuclear Service Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered (Auxiliary Building)	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered (Note 3)	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-37 Aging Management Review Results – Nuclear Service Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulation Water System Internal Coating Inspection
Pipe	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Strainers	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered (Note 3)	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Strainers	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered (Auxiliary Building)	None Identified	None Required

Table 3.3-37 Aging Management Review Results – Nuclear Service Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Tubing	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Yard	None Identified	None Required
Tubing	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered (Note 3)	None Identified	None Required
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered (Auxiliary Building)	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-37 Aging Management Review Results – Nuclear Service Water System (Catawba Nuclear Station) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Sheltered (Note 3)	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Raw Water	Loss of Material	Galvanic Susceptibility Inspection Service Water Piping Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Raw Water	Loss of Material	Service Water Piping Corrosion Program
			Yard	None Identified	None Required

Notes for Table 3.3-37 Aging Management Review Results – Nuclear Service Water System (Catawba Nuclear Station):

(1) **Component Function**

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) **Material**

CS Carbon Steel

SS Stainless Steel

(3) Sheltered environment to include the Diesel Building and/or the Pumphouse, but not the Auxiliary Building. Only in the sheltered environment of the Auxiliary Building is leakage from borated water systems possible.

(4) Loss of material due to corrosion will not affect the throttling function of orifices.

Table 3.3-38 Aging Management Review Results – Nuclear Service Water Pump Structure Ventilation System (Catawba Nuclear Station only)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Ductwork	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	BR	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	Cu	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	BR	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-38 Aging Management Review Results – Nuclear Service Water Pump Structure Ventilation System (Catawba Nuclear Station only) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-38 Aging Management Review Results – Nuclear Service Water Pump Structure Ventilation System (Catawba Nuclear Station only):

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CS Carbon Steel
Cu Copper
GS Galvanized Steel
SS Stainless Steel

Table 3.3-39 Aging Management Review Results – Nuclear Solid Waste Disposal System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment (Note 3)** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Pipe (MNS Only)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe (MNS Only) (Note 5)	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection

Table 3.3-39 Aging Management Review Results – Nuclear Solid Waste Disposal System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment (Note 3)** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe (MNS Only) (Note 5)	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Gas	None Identified	None Required
Screens (MNS Only)	FI	SS	(Note 4)	None Identified	None Required
			Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
Spent Resin Storage Tanks (MNS Only)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Spent Resin Storage Tanks (MNS Only)	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Tubing (MNS Only)	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required

Table 3.3-39 Aging Management Review Results – Nuclear Solid Waste Disposal System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment (Note 3) / External Environment	Aging Effect	Aging Management Programs and Activities
Valve Bodies	PB	SS	Treated Water	Cracking	Treated Water Systems Stainless Steel Inspection
				Loss of Material	Treated Water Systems Stainless Steel Inspection
			Sheltered	None Identified	None Required
Valve Bodies (MNS Only)	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-39 Aging Management Review Results – Nuclear Solid Waste Disposal System:

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

SS Stainless Steel

(3) The Treated Water environment of the Nuclear Solid Waste Disposal System contains spent fuel resin in solution.

(4) The Spent Fuel Resin Tank screens have no internal environment

(5) Pipe extending into the Spent Resin Storage Tank

Table 3.3-40 Aging Management Review Results – Reactor Coolant Pump Motor Oil Collection Sub-System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Flexible Hoses	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Level Gauges	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Level Gauges	PB	Glass	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Reactor Coolant Pump Motor Drain Tanks	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Reactor Coolant Pump Motor Drain Tank Pump Casings	PB	CI – MNS CS – CNS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-40 Aging Management Review Results – Reactor Coolant Pump Motor Oil Collection Sub-System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Coolant Pump Motor Lower Oil Catcher	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Reactor Coolant Pump Motor Lower Oil Pot (MNS Only)	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Reactor Coolant Pump Motor Oil Lift Enclosure	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Reactor Coolant Pump Motor Upper Oil Cooler Enclosures	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-40 Aging Management Review Results – Reactor Coolant Pump Motor Oil Collection Sub-System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Reactor Coolant Pump Motor Upper Oil Cooler Enclosures (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-40 Aging Management Review Results – Reactor Coolant Pump Motor Oil Collection Sub-System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies (MNS Only)	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-40 Aging Management Review Results – Reactor Coolant Pump Motor Oil Collection Sub-System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CI Cast Iron
CS Carbon Steel
SS Stainless Steel

Table 3.3-41 Aging Management Review Results – Reactor Coolant System (Non-Class 1 Components)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Orifices	PB, TH	SS	Gas (Note 3)	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing (CNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-41 Aging Management Review Results – Reactor Coolant System (Non-Class 1 Components) (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Gas	None Identified	None Required
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Gas	None Identified	None Required
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-41 Aging Management Review Results – Reactor Coolant System (Non-Class 1 Components):

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.
(2)	**Material**
CS	Carbon Steel
SS	Stainless Steel
(3)	The orifice will normally pass non-condensable gas, but may also pass borated water or steam, or any mixture of the three.

Table 3.3-42 Aging Management Review Results – Recirculated Cooling Water System

(Catawba Nuclear Station Only)
(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-42 Aging Management Review Results – Recirculated Cooling Water System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel

Table 3.3-43 Aging Management Review Results – Spent Fuel Cooling System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchangers (channel head)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
		CS	Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchangers (shell)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Heat Exchangers (tube sheet) (CNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
		CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Heat Exchangers (tube sheet) (MNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Table 3.3-43 Aging Management Review Results – Spent Fuel Cooling System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Heat Exchangers (tube)	PB, HT	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Chemistry Control Program
				Fouling	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Orifices	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe (CNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
Pump Casings	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-43 Aging Management Review Results – Spent Fuel Cooling System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Spacers (MNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies (CNS Only)	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Borated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program

Notes for Table 3.3-43 Aging Management Review Results – Spent Fuel Cooling System:

(1) Component Function

HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS	Carbon Steel
SS	Stainless Steel

Table 3.3-44 Aging Management Review Results – Standstandby Shutdown Diesel

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Cooling Water and Jacket Water Heating Sub-system					
Filter, Cooling Water (mounting head)	PB	CS	Treated Water	Cracking (Note 3)	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, Engine Radiator (tubes)	PB, HT	Cu	Treated Water	Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
Heat Exchanger, Engine Radiator (channel head)	PB	CS	Treated Water	Cracking (Note 3)	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Heat Exchanger, Engine Radiator (leakoff connector)	PB	Cu	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Cooling Water and Jacket Water Heating Sub-system (continued)					
Heat Exchanger, Engine Radiator (cap flange)	PB	CS	Treated Water	Cracking (Note 3)	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing	PB	CS	Treated Water	Cracking (Note 3)	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies, Jacket Water Heating	PB	BR	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Water Heater, Jacket (shell)	PB	CI	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Exhaust Sub-system					
Bellows, Exhaust	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe, Exhaust	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe, Exhaust	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Silencer, Exhaust	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Fuel Oil Sub-system					
Filter, Duplex (mounting head)	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Flame Arrestor (MNS Only)	PB	AL	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Fuel Oil Sub-system (continued)					
Level Glass	PB	GL-CNS	Fuel Oil	None Identified	None Required
		AC-MNS	Sheltered	None Identified	None Required
Level Glass	PB	GL-CNS	Ventilation	None Identified	None Required
		AC-MNS	Sheltered	None Identified	None Required
Pipe, Fuel Oil	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe, Fuel Oil	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Underground	Cracking	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection
Pipe, Fuel Oil	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required
Pipe, Fuel Oil Day Tank Vent	PB	SS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe, Fuel Oil Day Tank Vent	PB	SS	Ventilation	None Identified	None Required
			Yard	None Identified	None Required

Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Fuel Oil Sub-system (continued)					
Pipe, Fuel Oil Day Tank Drain (MNS Only)	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe, Fuel Oil Storage Tank Vent	PB	CS	Ventilation	None Identified	None Required
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe, Fuel Oil Storage Tank Vent	PB	CS	Ventilation	None Identified	None Required
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection
Pipe, Fuel Oil Storage Tank Suction	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Fuel Oil	Loss of Material	Chemistry Control Program
Pipe, Fuel Oil Storage Tank Suction	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Ventilation	None Identified	None Required
Pump Casing, Fuel Oil Transfer	PB	BZ	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pump Casing, Engine Fuel Oil	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Fuel Oil Sub-system (continued)					
Tank, Fuel Oil Storage	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection
Tank, Fuel Oil Storage	PB	CS	Ventilation	None Identified	None Required
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection
Tank, Fuel Oil Storage (manway) (MNS only)	PB	CS	Ventilation	None Identified	None Required
			Underground	Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection
Tank, Fuel Oil Day	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tank, Fuel Oil Day	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing, Fuel Oil (day tank)	PB	WI	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Fuel Oil Sub-system (continued)					
Tank, Fuel Oil Storage (manway (CNS only)	PB	CS	Ventilation	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing, Fuel Oil	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies, Fuel Oil	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies, Fuel Oil	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required
Valve Bodies, Fuel Oil	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Fuel Oil	Loss of Material	Chemistry Control Program
Valve Bodies, Fuel Oil (MNS Only)	PB	SS	Fuel Oil	Loss of Material	Chemistry Control Program
			Underground	Cracking	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection
				Loss of Material	Preventive Maintenance Activities – Condenser Circulating Water System Internal Coatings Inspection

Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Fuel Oil Sub-system (continued)					
Valve Bodies, Fuel Oil (MNS Only)	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies, Fuel Oil (duplex filters) (CNS Only)	PB	CS	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies, Fuel Oil	PB	BZ	Fuel Oil	Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Lubrication Oil Sub-system					
Filter, Lube Oil Bypass (housing)	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Filter, Lube Oil (mounting head)	PB	CS	Oil	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-44 Aging Management Review Results – Standby Shutdown Diesel:

(1) Component Function

HT	Provide heat transfer so that system and/or component operating temperatures are maintained.
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

AC	Acrylic
AL	Aluminum
BR	Brass
BZ	Bronze
CI	Cast Iron
CS	Carbon Steel
Cu	Copper
GL	Glass
SS	Stainless Steel
WI	Wrought Iron

(3) Cracking of carbon steel is only applicable to McGuire carbon steel components due to the use of nitrate corrosion inhibitors.

Table 3.3-45 Aging Management Review Results – Turbine Building Sump Pump System (Catawba Nuclear Station only)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Raw Water	Loss of Material	Sump Pump Systems Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-45 Aging Management Review Results – Turbine Building Sump Pump System (Catawba Nuclear Station only):

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel

Table 3.3-46 Aging Management Review Results – Turbine Building Ventilation System (McGuire Nuclear Station only)

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Air Handling Unit	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Ductwork	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Flexible Connectors	PB	Neoprene (Note 3)	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required
Plenum Section	PB	GS	Ventilation	None Identified	None Required
			Sheltered	None Identified	None Required

Notes for Table 3.3-46 Aging Management Review Results – Turbine Building Ventilation System (McGuire Nuclear Station only):

(1) **Component Function**

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) **Material**

GS Galvanized Steel

(3) Woven glass fabric double coated with neoprene.

Table 3.3-47 Aging Management Review Results – Waste Gas System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Flow Meters	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Hydrogen Recombiners	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Hydrogen Recombiner Heat Exchangers (Shell)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Hydrogen Recombiner Heat Exchangers (Tubes)	HT, PB	SS	Gas	None Identified	None Required
			Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
				Fouling	Chemistry Control Program
Hydrogen Recombiner Heaters	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Hydrogen Recombiner Phase Separators	PB, WR	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.3-47 Aging Management Review Results – Waste Gas System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Hydrogen Recombiner Phase Separators	PB, WR	SS	Treated Water	Cracking	Waste Gas System Inspection
				Loss of Material	Waste Gas System Inspection
			Sheltered	None Identified	None Required
Hydrogen Recombiner Safety Disc	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Orifices (Compressor Seal – CNS only)	PB	SS	Treated Water	Cracking	Waste Gas System Inspection
				Loss of Material	Waste Gas System Inspection
			Sheltered	None Identified	None Required
Orifices (Compressor Make-up – CNS only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices (MNS only)	PB,TH	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Pipe	PB	CS	Gas	Loss of Material	Galvanic Susceptibility Inspection Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-47 Aging Management Review Results – Waste Gas System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Waste Gas System Inspection
				Loss of Material	Waste Gas System Inspection
			Sheltered	None Identified	None Required
Pipe	PB	CS	Treated Water	Loss of Material	Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Strainers (CNS Only)	PB	SS	Treated Water	Cracking	Waste Gas System Inspection
				Loss of Material	Waste Gas System Inspection
			Sheltered	None Identified	None Required
Tubing	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies (CNS Only)	PB	BR	Treated Water	Loss of Material	Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	SS	Gas	None Identified	None Required
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Gas	Loss of Material	Galvanic Susceptibility Inspection Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-47 Aging Management Review Results – Waste Gas System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	SS	Treated Water	Cracking	Waste Gas System Inspection
				Loss of Material	Waste Gas System Inspection
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Loss of Material	Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Waste Gas Compressor Heat Exchangers (Tubes) (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Waste Gas System Inspection
				Loss of Material	Waste Gas System Inspection
Waste Gas Compressor Heat Exchangers (Tube Sheet) (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Treated Water	Cracking	Waste Gas System Inspection
				Loss of Material	Waste Gas System Inspection
Waste Gas Compressor Heat Exchangers (Channel Heads) (CNS Only)	PB	CS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.3-47 Aging Management Review Results – Waste Gas System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Waste Gas Compressor Heat Exchanger (Shell) (CNS only)	PB	CS	Treated Water	Loss of Material	Galvanic Susceptibility Inspection Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Waste Gas Decay Tanks	PB	CS	Gas	Loss of Material	Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Waste Gas Decay Tanks	PB	CS	Treated Water	Loss of Material	Waste Gas System Inspection
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.3-47 Aging Management Review Results – Waste Gas System:

(1) Component Function

HT Provide heat transfer so that system and/or component operating temperatures are maintained.
PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
WR Provide water removal so that sufficient moisture levels are maintained

(2) Material

BR Brass
CS Carbon Steel
SS Stainless Steel

This is the last page of Section 3.3.

3.4 AGING MANAGEMENT OF STEAM AND POWER CONVERSION SYSTEMS

Note: The aging management reviews for all steam and power conversion systems are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

The following mechanical systems are evaluated in the indicated tables in Section 3.4, Steam and Power Conversion Systems:

- Auxiliary Feedwater System (Table 3.4-1)
- Auxiliary Steam System (Table 3.4-2)
- Condensate System (Table 3.4-3)
- Condensate Storage System (Table 3.4-4)
- Feedwater System (Table 3.4-5)
- Feedwater Pump Turbine Exhaust System (Table 3.4-6)
- Main Steam System (Table 3.4-7)
- Main Steam Supply to Auxiliary Equipment System (Table 3.4-8)
- Main Steam Vent to Atmosphere System (Table 3.4-9)
- Turbine Exhaust System (Table 3.4-6)

3.4.1 AGING MANAGEMENT REVIEW RESULTS TABLES

The results of the aging management review for each system of this section are provided in a table, as indicated above. Information contained in each table was obtained in the following manner:

Column 1 – The component types listed in Column 1 were identified through the screening methodology described in Section 2.1.2 of this application and are on the marked plant drawings identified in Section 2.3.4 of this application.

Column 2 – The component functions listed in Column 2 were obtained from plant specific engineering documents using the screening methodology described in Section 2.1.2.

Column 3 – The materials listed in Column 3 were obtained from the drawings identified in Section 2.3.4 of this application and other plant specific engineering documents.

Column 4 – The internal and external environments listed in Column 4 were obtained from plant specific engineering documents. External environments are also noted on the drawings identified in Section 2.3.4 of this application. These environments are as follows:

- **Treated water –** Treated water is demineralized water that may be deareated, treated with a biocide or corrosion inhibitors, or a combination of these treatments. Treated water does not include borated water, which is evaluated separately.

- **Sheltered environment –** The ambient conditions within the sheltered environment may or may not be controlled. The sheltered environment atmosphere is a moist air environment. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry.

- **Reactor Building –** The Reactor Building environment is moist air. Components in systems with external surface temperatures the same or higher than ambient conditions due to normal system operation are expected to be dry.

- **Oil and Fuel Oil –** Lubricating oil is an organic fluid used to reduce friction between moving parts. Fuel oil is the fuel used for the emergency diesel generators.

Column 5 – The aging effects listed in Column 5 were obtained using the following aging effects identification process. The aging effects that require management during the period of extended operation have been determined by reviewing the plant-specific materials of construction (Column 3) and operating environments (Column 4) for each structure and component (Column 1) that is subject to an aging management review.

To provide reasonable assurance that the aging effects that require management for a specific material-environment combination are the only aging effects of concern for McGuire and Catawba, Duke also performed a review of industry experience and NRC generic communications relative to these structures and components. Finally, relevant McGuire and Catawba operating experience have been reviewed to provide further confidence that the set of aging effects for the specific material-environment combinations have been identified. Taken together, the steps of this methodology provide reasonable assurance that the aging effects that require management during the period of extended operation for McGuire and Catawba structures and components have been identified.

This aging effects identification process is consistent with that process used in Section 3.5 of the Oconee Nuclear Station license renewal application. Furthermore, in Section 3.1 of NUREG-1723 the staff concluded that based on its review of the information provided in Sections 3.5.1 and 3.5.2 of the Oconee application, "the applicant has identified the aging effects that are associated with mechanical systems components reviewed in [Section 3.5]." This aging effects identification process provides reasonable assurance that the aging effects that require management during the period of extended operation have been identified.

Column 6 – The aging management programs and activities listed in Column 6 are credited to manage the effects of aging for the period of extended operation.

3.4.2 AGING MANAGEMENT PROGRAMS

The following aging management programs and activities are credited to manage the effects of aging for the steam and power conversion systems listed in Section 3.4:

- Chemistry Control Program *
- Flow Accelerated Corrosion Program *
- Fluid Leak Management Program *
- Inspection Program for Civil Engineering Structures and Components *
- Flow Accelerated Corrosion Program *

* This aging management program/activity is equivalent or similar to the corresponding program/activity that has been previously reviewed and found acceptable by the NRC staff during the Oconee License Renewal review, as documented in NUREG-1723.

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in Tables 3.4-1 through 3.4-9 will be maintained consistent with the current licensing basis for the period of extended operation.

Table 3.4-1 Aging Management Review Results – Auxiliary Feedwater System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Eductors (CNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Motor-Driven CA Pump Casings	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Orifices	PB, TH	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (CNS Only)
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.4-1 Aging Management Review Results – Auxiliary Feedwater System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (CNS Only)
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required
Turbine-Driven CA Pump Casings	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.4-1 Aging Management Review Results – Auxiliary Feedwater System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
TDCAP Bearing Oil Cooler (Tubes)	PB, HT	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
				Fouling	Chemistry Control Program
			Oil	None Identified	None Required
TDCAP Bearing Oil Cooler (Tube Sheet)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Oil	None Identified	None Required
TDCAP Bearing Oil Cooler (Channel Heads)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
TDCAP Bearing Oil Cooler (Shell)	PB	SS	Oil	None Identified	None Required
			Sheltered	None Identified	None Required

Table 3.4-1 Aging Management Review Results – Auxiliary Feedwater System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note1)**	**Material (Note 2)**	**Internal Environment** / **External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (CNS Only)
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.4-1 Aging Management Review Results – Auxiliary Feedwater System:

(1) Component Function

HT Provide heat transfer so that system and/or component operating temperatures are maintained.

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS Carbon Steel
SS Stainless Steel

Table 3.4-2 Aging Management Review Results – Auxiliary Steam System

(Note are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Program**
Pipe	PB	CS	Treated Water / Steam	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (CNS Only)	PB	CS	Treated Water / Steam	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Yard (Trench)	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing (CNS Only)	PB	CU	Treated Water / Steam	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing (CNS Only)	PB	BR	Treated Water / Steam	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
					Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Tubing (MNS Only)	PB	SS	Treated Water / Steam	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.4-2 Aging Management Review Results – Auxiliary Steam System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Program**
Valve Bodies (CNS Only)	PB	BR	Treated Water / Steam	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program
Valve Bodies	PB	CS	Treated Water / Steam	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water / Steam	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.4-2 Aging Management Review Results – Auxiliary Steam System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

BR Brass
CS Carbon Steel
CU Copper
SS Stainless Steel

Table 3.4-3 Aging Management Review Results – Condensate System

(Notes are located at the end of this table)
(Catawba Nuclear Station only)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.4-3 Aging Management Review Results – Condensate System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel

Table 3.4-4 Aging Management Review Results – Condensate Storage System

(Notes are located at the end of this table)
(Catawba Nuclear Station only)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.4-4 Aging Management Review Results – Condensate Storage System:

(1) Component Function

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

(2) Material

CS Carbon Steel

Table 3.4-5 Aging Management Review Results – Feedwater System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Cavitating Venturies (CNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			(Note 3)	None Identified	None Required
Cavitating Venturies (CNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			(Note 3)	None Identified	None Required
Flow Nozzles (MNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Yard	None Identified	None Required
Orifices (CNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Orifices	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.4-5 Aging Management Review Results – Feedwater System

(continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe (CNS Only)	PB	AS	Treated Water	Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe (CNS Only)	PB	AS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	AS	Treated Water	Loss of Material	Chemistry Control Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.4-5 Aging Management Review Results – Feedwater System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Pipe (MNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Yard	None Identified	None Required
Pipe (MNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Pipe (CNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.4-5 Aging Management Review Results – Feedwater System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Program or Activities**
Reservoirs	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required
Tubing (MNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Yard	None Identified	None Required

Table 3.4-5 Aging Management Review Results – Feedwater System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies (CNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies (MNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Valve Bodies (CNS Only)	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Loss of Material	Chemistry Control Program
				Cracking	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.4-5 Aging Management Review Results – Feedwater System:

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
(2)	**Material**
AS	Low Alloy Steel
CS	Carbon Steel
SS	Stainless Steel
(3)	The cavitating venturies are located entirely inside the carbon steel pipe of the Feedwater System and are not exposed to the external environments.

Table 3.4-6 Aging Management Review Results – Feedwater Pump Turbine Exhaust System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Expansion Joint (CNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Expansion Joint (Bellows) (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices (CNS Only)	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.4-6 Aging Management Review Results – Feedwater Pump Turbine Exhaust System (continued)

Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effects	Aging Management Programs and Activities
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (MNS Only)
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program Flow Accelerated Corrosion Program (MNS Only)
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies (CNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Notes for Table 3.4-6 Aging Management Review Results – Feedwater Pump Turbine Exhaust System:

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure provide pressure reduction, or provide differential pressure.

(2)	**Material**
CS	Carbon Steel
SS	Stainless Steel

Table 3.4-7 Aging Management Review Results – Main Steam System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Orifices	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Orifices	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Orifices	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.4-7 Aging Management Review Results – Main Steam System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.4-7 Aging Management Review Results – Main Steam System (continued)

1	2	3	4	5	6
Component Type	Component Function (Note 1)	Material (Note 2)	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Valve Bodies (CNS Only)	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Reactor Building	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies (MNS Only)	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Yard	None Identified	None Required

Notes for Table 3.4-7 Aging Management Review Results – Main Steam System:

(1) Component Function

PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
TH	Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS	Carbon Steel
SS	Stainless Steel

Table 3.4-8 Aging Management Review Results – Main Steam Supply to Auxiliary Equipment System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Auxiliary Feedwater Pump Turbine	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Orifices	PB, TH	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.4-8 Aging Management Review Results – Main Steam Supply to Auxiliary Equipment System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Strainers (MNS Only)	PB, FI	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required

Table 3.4-8 Aging Management Review Results – Main Steam Supply to Auxiliary Equipment System (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.4-8 Aging Management Review Results – Main Steam Supply to Auxiliary Equipment System:

(1) Component Function

FI Provide filtration of process fluid so that downstream equipment and/or environments are protected.

PB Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.

TH Provide throttling so that sufficient flow and/or sufficient pressure is delivered, provide backpressure, provide pressure reduction, or provide differential pressure.

(2) Material

CS Carbon Steel
SS Stainless Steel

Table 3.4-9 Aging Management Review Results – Main Steam Vent to Atmosphere System

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material (Note 2)**	**Internal Environment / External Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Pipe	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Yard	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Tubing	PB	SS	Treated Water	Cracking	Chemistry Control Program
				Loss of Material	Chemistry Control Program
			Sheltered	None Identified	None Required
Valve Bodies	PB	CS	Treated Water	Loss of Material	Chemistry Control Program
			Sheltered	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.4-9 Aging Management Review Results – Main Steam Vent to Atmosphere System:

(1)	**Component Function**
PB	Maintain mechanical pressure boundary integrity so that sufficient flow and/or sufficient pressure are delivered, effect Containment isolation for fission product retention, or prevent physical interaction with safety-related equipment.
(2)	**Material**
CS	Carbon Steel
SS	Stainless Steel

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3.5 AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Note: The aging management reviews for all structures, structural components and component supports are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

The aging management review results for the Reactor Buildings, including the concrete shield building, the steel containment, the ice condenser components and all Reactor Building interior structural components, except component supports, are presented in Table 3.5-1 Aging Management Review Results – Reactor Building.

The aging management review results for structural components located within the following structures, except for component supports, are provided in Table 3.5-2 Aging Management Review Results – Other Structures:

- Auxiliary Building (including Control Building, Diesel Generator Buildings, Fuel Buildings, Groundwater Drainage System, and Main Steam Doghouses, and UHI Tank Building (Catawba only))
- Condenser Cooling Water Intake Structure – McGuire Nuclear Station fire pump rooms only
- Nuclear Service Water Structures
- Standby Nuclear Service Water Pond Dam
- Standby Shutdown Facility
- Turbine Buildings (including Service Building)
- Unit Vent Stack
- Yard Structures (including Low Pressure Service Water Intake Structure (Catawba only), Refueling Water Storage Tank foundation and missile wall, Reactor Makeup Water Storage Tank foundations (McGuire only), trenches, and Yard Drainage System (Catawba only))

The aging management review results for all equipment and component supports within the scope of license renewal are provided in Table 3.5-3 Aging Management Review Results – Component Supports.

3.5.1 AGING MANAGEMENT REVIEW RESULTS TABLES

The results of the aging management review for each structure of this section are provided in a table, as indicated above. Information contained in each table was obtained in the following manner:

Column 1 – The component types listed in Column 1 were identified through the screening methodology described in Section 2.1.2 of this application.

Column 2 – The component functions listed in Column 2 were obtained from plant specific engineering documents using the screening methodology described in Section 2.1.2.

Column 3 – The materials listed in Column 3 were obtained from plant specific engineering documents.

Column 4 – The external environments listed in Column 4 were obtained from plant specific engineering documents. These environments are as follows:

- **Below Grade –** Below grade portions of structures are exposed to back fill and groundwater. The groundwater at McGuire and Catawba is not aggressive. The McGuire groundwater pH ranges between 8.1 and 8.4; the chloride concentration is less than 20 ppm; and the sulfate concentration is less than 30 ppm. The Catawba groundwater pH ranges between 5.7 and 7.0; the chloride concentration is less than 25 ppm; and the sulfate concentration is less than 35 ppm.

- **Borated Water –** This environment is associated with the Spent Fuel Pool which contains an oxygen saturated borated water with a concentration of approximately 2000 to 4000 ppm boron.

- **Concrete –** Steel components located in concrete are protected by the alkaline environment of the concrete.

- **External –** External surfaces of structures are exposed to the external ambient environment.

- **Ice Condenser Environment –** The normal operating atmosphere in the ice condenser is at 10°F to 20°F and the absolute humidity is very low.

- **Raw Water –** This environment consists of lake water from either Lake Norman for McGuire Nuclear Station or Lake Wylie for Catawba Nuclear Station. The raw water at Lake Norman and Lake Wylie is not aggressive. The Lake Norman water pH ranges between 5.5 and 8.5; the chloride concentration is less than 15 ppm; and the sulfate concentration is less than 15 ppm. The Lake Wylie water pH ranges between 5.7 and 9.3; the chloride concentration is less than 30 ppm; and the sulfate concentration is less than 15 ppm.

- **Reactor Building –** The Reactor Building environment is moist air.

- **Sheltered –** The ambient conditions within the sheltered environment may or may not be controlled. The sheltered environment atmosphere is a moist air environment.

3.5.2 Aging Management Programs

The following aging management programs and activities are credited to manage the effects of aging for the structures and structural components listed in section 3.5:

- Battery Rack Inspections*
- Boraflex Monitoring Program* (McGuire only)
- Chemistry Control Program*
- Containment Inservice Inspection Plan – IWE*
- Containment Leak Rate Testing Program*
- Crane Inspection Program*
- Divider Barrier Seal Inspection and Testing Program
- Fire Protection Program*
- Flood Barrier Inspection Program (McGuire only)
- Fluid Leak Management Program*
- Ice Condenser Inspections
- Inservice Inspection Plan – IWF*
- Inspection Program for Civil Engineering Structures and Components*
- Standby Nuclear Service Water Pond Dam Inspection
- Technical Specification SR 3.6.16.3 Visual Inspection
- Underwater Inspection of Nuclear Service Water Structures*

** This aging management program/activity is equivalent to the corresponding program/activity that has been previously reviewed and found acceptable by the NRC staff during the Oconee License Renewal review, as documented in NUREG-1723.*

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in Tables 3.5-1 through 3.5-3 will be maintained consistent with the current licensing basis for the period of extended operation.

Table 3.5-1 Aging Management Review Results – Reactor Building

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Concrete Shield Building					
Dome	2, 3, 6, 7	Concrete	Reactor Building	None Identified	None Required
			External	Change in Material Properties due to leaching	Technical Specification SR 3.6.16.3 Visual Inspection
Foundation Dowels (McGuire only)	2, 7	Steel	Concrete	None Identified	None Required
Foundation Mat	2, 7, 11	Concrete	Below Grade	None Identified	None Required
Shell Wall	2, 3, 4, 6, 7, 11	Concrete	Reactor Building	None Identified	None Required
			Below Grade	None Identified	None Required
			External	Change in Material Properties due to leaching	Technical Specification SR 3.6.16.3 Visual Inspection

Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Containment					
Bellows (Penetration)	1	Stainless	Reactor Building	Cracking	Containment Leak Rate Testing Program
Electrical Penetrations	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Equipment Hatch	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Fuel Transfer Tube Penetration	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Mechanical Penetrations	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Personnel Air Locks	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program
Steel Containment Vessel	1	Steel	Reactor Building	Loss of Material	Containment Inservice Inspection Plan – IWE Containment Leak Rate Testing Program

Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Ice Condenser Components					
Ice Baskets	2, 7	Galvanized Steel	Ice Condenser	Loss of Material	Ice Condenser Inspections
Lattice Frames & Support Columns	2, 7	Steel	Ice Condenser	Loss of Material	Ice Condenser Inspections
Lower Inlet Doors, Intermediate Deck Doors, Top Deck Doors	2, 3, 7	Steel	Ice Condenser Reactor Building	Loss of Material	Ice Condenser Inspections
Lower Support Structure	2, 7	Steel	Ice Condenser	Loss of Material	Ice Condenser Inspections
Wear Slab	2, 7	Concrete	Ice Condenser	None Identified	None Required

Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reactor Building Interior Structural Components					
Anchorage	2, 7, 11	Steel	Concrete	None Identified	None Required
Anchorage (exposed surface)	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Checkered Plate	3	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Embedments	2, 7, 11	Steel	Concrete	None Identified	None Required
Embedments (exposed surface)	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Equipment Pads	2, 7, 11	Concrete	Reactor Building	None Identified	None Required
Expansion Anchors	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Flood Curbs	2, 8	Concrete	Reactor Building	None Identified	None Required
Flood Curbs (Steel)	2, 8	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components

Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reactor Building Interior Structural Components (continued)					
Flood, Pressure, & Specialty Doors	1, 3, 8	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Fuel Transfer Canal Liner Plate	1	Stainless	Reactor Building	None Identified	None Required
Hatches	3, 6, 11	Concrete	Reactor Building	None Identified	None Required
Missile Shields	3, 6	Concrete	Reactor Building	None Identified	None Required
			External (equipment hatch missile shield)	None Identified	None Required
Pressure Seals & Gaskets	1	EPDM (Note 2)	Reactor Building	Cracking Change in Material Properties	Divider Barrier Seal Inspection and Testing Program
Reinforced Concrete Beams, Columns, Floor Slabs, Walls	1, 2, 3, 4, 6, 7, 8, 10, 11	Concrete	Reactor Building	None Identified	None Required

Table 3.5-1 Aging Management Review Results – Reactor Building (continued)

1	2	3	4	5	6
Component Type	**Component Function (Note 1)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Reactor Building Interior Structural Components (continued)					
Structural Steel Beams, Columns, Plates & Trusses	2, 7, 11	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program Inspection Program for Civil Engineering Structures and Components
Sump Liner	1	Stainless	Reactor Building	None Identified	None Required
Sump Screens (recirculation intake screen)	2	Stainless	Reactor Building	None Identified	None Required
Sumps	2	Concrete	Reactor Building	None Identified	None Required

Notes for Table 3.5-1 Aging Management Review Results – Reactor Building:

(1) Component Function

1 Provides pressure boundary and/or fission product barrier.
2 Provides structural and/or functional support to safety-related equipment.
3 Provides shelter/protection to safety-related equipment.
4 Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
5 Provides Ultimate Heat Sink following a LOCA or loss of Lake Norman or Lake Wylie.
6 Serves as missile (internal or external) barrier.
7 Provides structural and/or functional support to non-safety related equipment where failure of this component could directly prevent satisfactory accomplishment of any of the required safety-related functions.
8 Provides a protective barrier for internal/external flood event.
9 Provides path for release of filtered and unfiltered gaseous discharge.
10 Provides heat sink during SBO or design basis accidents.
11 Provides structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events.

(2) EPDM is the acronym for ethylene propylene dienyl monomer.

Table 3.5-2 Aging Management Review Results – Other Structures

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Concrete Structural Components					
Equipment Pads	2, 7, 11	Concrete	Sheltered	None Identified	None Required
Fire Walls	4	Concrete	Sheltered	Cracking	Fire Protection Program
Flood Curbs	8	Concrete	Sheltered	None Identified	None Required
Foundation Cassions (MNS TB only)	11	Concrete	Below Grade	None Identified	None Required
Foundations	2, 7, 11	Concrete	Below Grade	None Identified	None Required
Hatches	3, 4, 6, 11	Concrete	Sheltered	None Identified	None Required
Manholes & Covers (CNS NSW only)	3	Concrete	Below Grade External	None Identified	None Required
Missile Shields (AB and NSW only)	3, 6	Concrete	Sheltered (AB only) External (AB and NSW only)	None Identified	None Required
			Raw Water (NSW only)	Loss of Material Cracking	Underwater Inspection of Nuclear Service Water Structures Inspection Program for Civil Engineering Structures and Components
Missile Shield (RWST Missile Shield Wall)	6	Concrete	External	Change in material properties due to leaching	Inspection Program for Civil Engineering Structures and Components

Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Concrete Structural Components (continued)					
Reinforced Concrete Beams, Columns, Floor Slabs, Walls	1(AB only), 2, 3, 4, 6, 7, 8 (AB only), 10 (AB only), 11	Concrete	Sheltered	None Identified	None Required
			External	Change in material properties due to leaching	Inspection Program for Civil Engineering Structures and Components
			Below Grade	None Identified	None Required
			Raw Water (NSW and LPSW (CNS))	Loss of Material Cracking	Inspection Program for Civil Engineering Structures and Components Underwater Inspection of Nuclear Service Water Structures
Roof Slabs	2, 3, 6, 7, 11	Concrete	External	Change in material properties due to leaching	Inspection Program for Civil Engineering Structures and Components
Sumps (AB only)	1, 2	Concrete	Sheltered	None Identified	None Required
Trenches (Yard only)	3, 11	Concrete	Below Grade	None Identified	None Required

Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Structural Components					
Anchorage	2, 7, 11	Steel	Concrete	None Identified	None Required
Anchorage (exposed surface)	2, 7, 11	Steel	Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
			External/Raw Water	Loss of Material	Inspection Program for Civil Engineering Structures and Components Underwater Inspection of Nuclear Service Water Structures
Checkered Plate	3, 11	Steel	Sheltered External	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Embedments	2, 7, 11	Steel	Concrete	None Identified	None Required
Embedments (exposed surface)	2, 7, 11	Steel	Sheltered External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Structural Components (continued)					
Expansion Anchors	2, 7, 11	Steel	Sheltered External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Fire Doors (AB and CNS NSW only)	4	Steel	Sheltered External (Yard only)	Loss of Material	Fire Protection Program
Flood Curbs	8	Steel	Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Flood, Pressure, & Specialty Doors (AB, TB, and CNS NSW only)	1, 3, 8	Steel	Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Foundation Dowels (MNS AB and CCW only)	2, 7, 11	Steel	Concrete	None Identified	None Required
			Below Grade	None Identified	None Required
Metal Siding (MNS Battery Rooms only)	1, 3	Steel	Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Steel Structural Components (continued)					
Roof (MNS Fire Pump enclosure roof cover)	11	Steel	External	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Spent Fuel Pool Liner Plate (AB only)	1, 3	Stainless	Borated Water	Loss of Material Cracking	Chemistry Control Program
Structural Steel Beams, Columns, Plates & Trusses	2, 7, 11	Steel	Sheltered External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Structural Steel and Plates	2, 7	Stainless	Borated Water (AB only)	Loss of Material Cracking	Chemistry Control Program
			External/Raw Water (NSW only)	Loss of Material	Underwater Inspection of Nuclear Service Water Structures Inspection Program for Civil Engineering Structures and Components

Table 3.5-2 Aging Management Review Results – Other Structures (continued)

Component Type (Note 1)	Component Function (Note 2)	Material	Environment	Aging Effects	Aging Management Programs and Activities
Steel Structural Components (continued)					
Sump Screens (AB only)	2	Steel	Sheltered	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Trash Rack and Screens (NSW only)	2	Stainless or Steel (CNS only)	Raw Water	Loss of Material	Underwater Inspection of Nuclear Service Water Structures
Unit Vent Stack	9	Steel	Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components
			External	Loss of Material	Inspection Program for Civil Engineering Structures and Components
Yard Drainage System (CNS only)	7	Steel	External	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Table 3.5-2 Aging Management Review Results – Other Structures (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Other Structural Components					
Boraflex Panels (MNS AB only)	2, 7	Boraflex	Borated Water	Degradation due to Gamma irradiation	Boraflex Monitoring Program
Earthen Embankment	2, 5	Soil	External	Loss of Material Cracking	Standby Nuclear Service Water Pond Dam Inspection
Fire Barrier Penetration Seals (AB and CNS NSW only)	4	Silicone	Sheltered	Cracking Separation	Fire Protection Program
		Rubber	Sheltered	Cracking	Fire Protection Program
Flood Seals	8	Rubber Silicon	Sheltered	Cracking Change in Material Properties	Flood Barrier Inspection (MNS only) Inspection Program for Civil Engineering Structures and Components (CNS Only)
Masonry Block Walls (AB, SSF, TB only)	2, 3, 4, 7, 11	Masonry	Sheltered	Cracking	Inspection Program for Civil Engineering Structures and Components
Metal Siding (Yard only)	3	Aluminum	External	None Identified	None Required
Roofing	3, 11	Composite	External	Loss of Material	Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.5-2 Aging Management Review Results – Other Structures:

(1) Location Abbreviations

AB = Auxiliary Building
CCW = Condenser Cooling Water Intake Structure (McGuire Fire Pump Rooms only)
CNS = Catawba Nuclear Station
LPSW = Low Pressure Service Water Intake Structure (Catawba)
MNS = McGuire Nuclear Station
NSW = Nuclear Service Water Structures
RB = Reactor Building
SNSWP = Standby Nuclear Service Water Pond Dam
SSF = Standby Shutdown Facility
TB = Turbine Buildings

(2) Component Function

1 Provides pressure boundary and/or fission product barrier.
2 Provides structural and/or functional support to safety-related equipment.
3 Provides shelter/protection to safety-related equipment.
4 Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
5 Provides Ultimate Heat Sink following a LOCA or loss of Lake Norman or Lake Wylie.
6 Serves as missile (internal or external) barrier.
7 Provides structural and/or functional support to non-safety related equipment where failure of this component could directly prevent satisfactory accomplishment of any of the required safety-related functions.
8 Provides a protective barrier for internal/external flood event.
9 Provides path for release of filtered and unfiltered gaseous discharge.
10 Provides heat sink during SBO or design basis accidents.
11 Provides structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events. (For McGuire CCW and SSF, Function 11 only applies; For Catawba LPSW and SSF, Function 11 only applies.)

(3) The Fluid Leak Management Program is applicable for structural components that are listed in this table that are only located in the Auxiliary Building.

Table 3.5-3 Aging Management Review Results – Component Supports

(Notes are located at the end of this table)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Battery Racks (AB, SSF only)	2, 11	Steel	Sheltered	Loss of Material	Battery Rack Inspections
Cable Tray & Conduit	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	None Identified	None Required
Cable Tray & Conduit Supports	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Class 1 (NSSS) Supports	2	Steel	Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inservice Inspection Plan – Subsection IWF Inspection Program for Civil Engineering Structures and Components
Control Boards (AB, SSF only)	2, 3, 7, 11	Steel	Sheltered	None Identified	None Required
Control Room Ceiling (AB only)	7	Steel	Sheltered	None Identified	None Required

Table 3.5-3 Aging Management Review Results – Component Supports (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Crane Rails & Girders (AB, RB only)	7	Steel	Sheltered Reactor Building	Loss of Material	Crane Inspection Program
Electrical & Instrument Panels & Enclosures	2, 3, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	None Identified	None Required
Equipment Component Supports	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Inservice Inspection Plan – Subsection IWF Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
			Raw Water (NSW only)	Loss of Material	Underwater Inspection of Nuclear Service Water Structures Inspection Program for Civil Engineering Structures and Components

Table 3.5-3 Aging Management Review Results – Component Supports (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
HVAC Duct Supports (RB, AB, SSF, and CNS NSW only)	2, 7, 11	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Instrument Racks & Frames	2, 7, 11	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Instrument Line Supports	2, 7, 11	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Lead Shielding Supports (RB and AB only)	7	Steel	Sheltered Reactor Building	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
New Fuel Storage Racks (AB only)	2	Steel	Sheltered	None Identified	None Required

Table 3.5-3 Aging Management Review Results – Component Supports (continued)

1	2	3	4	5	6
Component Type (Note 1)	**Component Function (Note 2)**	**Material**	**Environment**	**Aging Effects**	**Aging Management Programs and Activities**
Pipe Supports	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Inservice Inspection Plan – Subsection IWF Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components
Spent Fuel Storage Racks (AB only)	2	Stainless	Borated Water	Loss of Material Cracking	Chemistry Control Program
Stair, Platform, and Grating Supports	2, 7, 11	Steel	Sheltered Reactor Building External (Yard only)	Loss of Material	Fluid Leak Management Program (Note 3) Inspection Program for Civil Engineering Structures and Components

Notes for Table 3.5-3 Aging Management Review Results – Component Supports:

(1)	**Location Abbreviations**
	AB = Auxiliary Building
	CCW = Condenser Cooling Water Intake Structure (McGuire Fire Pump Rooms only)
	CNS = Catawba Nuclear Station
	MNS = McGuire Nuclear Station
	NSW = Nuclear Service Water Structures
	RB = Reactor Building
	SNSWP = Standby Nuclear Service Water Pond Dam
	SSF = Standby Shutdown Facility
	TB = Turbine Buildings
(2)	**Component Function**
1	Provides pressure boundary and/or fission product barrier.
2	Provides structural and/or functional support to safety-related equipment.
3	Provides shelter/protection to safety-related equipment.
4	Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
5	Provides Ultimate Heat Sink following a LOCA or loss of Lake Norman or Lake Wylie.
6	Serves as missile (internal or external) barrier.
7	Provides structural and/or functional support to non-safety related equipment where failure of this component could directly prevent satisfactory accomplishment of any of the required safety-related functions.
8	Provides a protective barrier for internal/external flood event.
9	Provides path for release of filtered and unfiltered gaseous discharge.
10	Provides heat sink during SBO or design basis accidents.
11	Provides structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events. (For McGuire CCW and SSF, Function 11 only applies; For Catawba LPSW and SSF, Function 11 only applies.)
(3)	The Fluid Leak Management Program is applicable for component supports only in the Auxiliary Building and the Reactor Building

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3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

Note: The aging management reviews for all non-EQ insulated cables and connections are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station unless otherwise stated.

Based on industry literature, plant operating experience and lessons learned from previous reviews performed for license renewal (1) aging effects caused by heat and radiation, (2) aging effects caused by moisture and voltage stress for medium-voltage cables, and (3) aging effects caused by boric acid ingress into connector pins are required to be included in the aging management review of non-EQ insulated cables and connections. This aging effects identification process specifically included a review of McGuire and Catawba Licensee Event Reports dealing with insulated cables and connections and is consistent with the process used in Section 3.6 of the Oconee Nuclear Station license renewal application. Details of the aging effects are provided in Table 3.6-1.

Table 3.6-1 Aging Effects Included in the Aging Management Review for Non-EQ Insulated Cables and Connections

Name	Subcomponent	Applicable Stressors	Degradation	Potential Effects
Aging effects caused by heat and radiation	Insulation material	Heat, oxygen	Embrittlement, cracking, melting, discoloration	Reduced insulation resistance (IR), electrical failure
		Radiation, oxygen	Embrittlement, cracking, discoloration, swelling	
Aging effects caused by moisture and voltage stress for inaccessible medium-voltage cables	Medium-voltage cable insulation material	Moisture and voltage stress	Formation of water trees, localized damage	Electrical failure (breakdown of insulation)
Aging effects caused by boric acid ingress into connector pins	Connector pins	Borated water	Corrosion	Connector failure

3.6.1 AGING EFFECTS CAUSED BY HEAT AND RADIATION

A bounding plant spaces approach is used to determine the required aging management programs and activities that will manage aging effects caused by heat and radiation such that the intended function non-EQ insulated cables and connections is maintained consistent with the current licensing basis for the period of extended operation.

The cable and connection material of interest for the aging management review is the primary conductor insulating material (hereafter referred to as insulation material). Using the plant spaces approach, cable and connection insulation materials with aging properties that bound the aging properties of other installed cable and connection insulation materials are identified and used in the aging management review. The bounding insulation materials for the review of aging effects caused by heat and radiation along with the insulation material 60-year service-limiting temperature and 60-year service-limiting radiation dose are listed in Table 3.6-2.

Table 3.6-2 Bounding Insulation Materials for the Review of Aging Effects Caused by Heat and Radiation

Bounding Insulation Materials	60-Year Service-Limiting Temperature	
EP, EPR, EPDM, FREP, Hypalon	Power	167°F (75°C)
PVC	I&C	112°F (44°C)
Bounding Insulation Materials	**60-Year Service-Limiting Radiation Dose**	
Hypalon, Nylon	Power, I&C	2×10^6 rads

The review of aging effects caused by heat and radiation includes the identification of the service conditions for insulated cables and connections. Service conditions include both ambient radiation and temperature with ohmic heat included for power applications. The service conditions for non-EQ insulated cables and connections are listed in Table 3.6-3. The service conditions identified in Table 3.6-3 are bounding values. These bounding values are greater than the actual values for most plant areas due to factors such as daily and seasonal temperature fluctuations and unit outages.

Table 3.6-3 Bounding Service Conditions for the Review of Aging Effects Caused by Heat and Radiation

Structure or Area	Design Ambient Temperature	Plus Ohmic Heat for Power Applications
Reactor Building Lower Containment	120°F (49°C)	162°F (72°C)
All other areas	110°F (43°C)	162°F (72°C)
Structure or Area	**Design Ambient Radiation Dose**	
Auxiliary and Reactor Buildings	6.8×10^9 rads	
All other areas	less than 3.0×10^4 rads	

Table 3.6-4 compares the temperature and radiation service conditions to the 60-year service-limiting temperature and radiation dose for the bounding insulation materials. The results of this comparison are provided in the right column of the table and are discussed following the table.

Table 3.6-4 Comparison of Service Conditions to Insulation Material 60-Year Service-Limiting Temperature and Radiation Dose

Bounding Insulation Materials	60-Year Service-Limiting Temperature and Radiation Dose	Bounding Temperature and Radiation Dose / Structure or Area	OK for 60 Years of Service
I&C applications PVC	112°F (44°C)	120°F (49°C) Reactor Buildings Lower Containment	No
Power applications EP, EPR, EPDM, FREP, Hypalon	167°F (75°C)	162°F (72°C) All Areas	Yes
Hypalon, Nylon	2×10^6 rads	6.8×10^9 rads Auxiliary and Reactor Buildings	No

There are plant areas where the bounding service conditions are greater than the 60-year service-limiting temperature or radiation dose; identified with a "No" in the right column of Table 3.6-4. This signifies that some insulation materials are not suited for the bounding service conditions for 60 years of service. Based on this finding, and choosing not to refine the service conditions for specific plant areas or to scope within insulated cables and connections, aging management is required to demonstrate reasonable assurance that the intended functions of non-EQ insulated cables and connections will be maintained consistent with the current licensing basis through the period of extended operation. A new program, the *Non-EQ Insulated Cables and Connections Aging Management Program*, will be implemented to demonstrate this reasonable assurance.

3.6.2 Aging Effects Caused by Moisture and Voltage Stress for Inaccessible Medium-Voltage Cables

Aging effects caused by moisture and voltage stress have been identified as potential aging effects for 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). Medium-voltage cables routed in conduit at Catawba is not a concern due to the design criteria documented in UFSAR Section 8.3.1.4.5.1, *Cable Installation*, that conduit runs are sloped for drainage.

In addition to being exposed to long-term, continuous standing water, to be identified as being susceptible to aging effects caused by moisture and voltage stress an inaccessible non-EQ medium-voltage cable must normally be energized more than 25% of the time. The time calculation includes outages.

The two criteria identified above are conservative and are used only as threshold values for an inaccessible non-EQ medium-voltage cable to be identified as susceptible to aging effects caused by moisture and voltage stress. A qualifier to these two criteria is that if an inaccessible non-EQ medium-voltage cable is designed or specified for the conditions described in these two criteria then the cable is not considered susceptible to aging effects caused by moisture and voltage stress.

To provide reasonable assurance that the intended functions of inaccessible non-EQ medium-voltage cables susceptible to aging effects caused by moisture and voltage stress will be maintained consistent with the current licensing basis through the period of extended operation a new program, the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program*, will be implemented.

3.6.3 Aging Effects Caused by Boric Acid Ingress into Connector Pins

During the Oconee License Renewal Application review potential boric acid ingress into connector pins was identified as causing aging effects that needs to be managed. The *Fluid Leak Management Program* (which includes boric acid leakage surveillance) will be credited for managing aging effects caused by boric acid ingress into non-EQ connector pins at McGuire and Catawba.

3.6.4 Aging Management Review Results Table

The results of the aging management review for non-EQ insulated cables and connections are provided in Table 3.6-5. Information in Table 3.6-5 was obtained from industry literature and plant specific engineering documentation as part of the aging management review.

Column 1 – The component types listed in Column 1 were obtained from plant specific engineering documentation.

Column 2 – The component function listed in Column 2 was obtained from plant specific engineering documentation.

Column 3 – The material information listed in Column 3 was obtained from plant specific engineering documentation.

Column 4 – The environment information listed in Column 4 was obtained from plant specific engineering documents.

Column 5 – The aging effects listed in Column 5 were obtained using the process described in Section 3.6 of this Application.

Column 6 – The aging management programs listed in Column 6 are credited to manage the aging effects for the period of extended operation as discussed in Sections 3.6.1, 3.6.2 and 3.6.3 or this Application.

3.6.5 AGING MANAGEMENT PROGRAMS

The following aging management programs and activities are credited to manage the effects of aging for non-EQ insulated cables and connectors listed in Section 3.6:

- Non-EQ Insulated Cables and Connections Aging Management Program*
- Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program*
- Fluid Leak Management Program*

** These aging management programs are equivalent to the corresponding programs that have been previously reviewed and found acceptable by the NRC staff during the Oconee License Renewal review, as documented in NUREG-1723.*

Based on the evaluations provided in Appendix B for the aging management programs and activities listed above, the aging effects will be adequately managed such that the intended functions of the components listed in Table 3.6-5 will be maintained consistent with the current licensing basis for the period of extended operation.

Table 3.6-5 Aging Management Review Results – Non-EQ power, Instrumentation, Control and Communication Insulated Cables and Connections

1	2	3	4	5	6
Component Type	**Component Function**	**Material**	**Environment**	**Aging Effect**	**Aging Management Programs and Activities**
Non-EQ insulated cables and connections	Maintain electrical connection to specified sections of an electrical circuit to deliver voltage, current or signals	Insulation materials - various organic polymers	Heat or radiation	Reduced insulation resistance (IR), electrical failure	Non-EQ Insulated Cables and Connections Aging Management Program
Inaccessible Non-EQ medium-voltage (4.16kV, 6.9kV, 13.8kV) cables	Maintain electrical connection to specified sections of an electrical circuit to deliver voltage, current or signals	Insulation materials - various organic polymers	Moisture and voltage stress	Electrical failure (breakdown of insulation)	Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program
Non-EQ connectors	Maintain electrical connection to specified sections of an electrical circuit to deliver voltage, current or signals	Connector pins - various metals	Borated water leakage	Connector failure	Fluid Leak Management Program

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4.0 TIME-LIMITED AGING ANALYSES

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

Note: The methodology to identify time-limited aging analyses described in Section 4.1 is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

4.1.1 BACKGROUND

Section 54.21(c) requires a list of time-limited aging analyses be provided as part of the application for a renewal license. Time-limited aging analyses are defined in §54.3 as those licensee calculations and analyses that meet six specific criteria.

§54.21 Contents of Application - technical information

(c) An evaluation of time-limited aging analyses.

(1) A list of time-limited aging analyses, as defined in §54.3, must be provided. The applicant shall demonstrate that —

(i) The analyses remain valid for the period of extended operation;
(ii) The analyses have been projected to the end of the period of extended operation; or
(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation

(2) A list must be provided of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in §54.3. The applicant shall provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

§54.3 Definitions

Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and

(6) Are contained or incorporated by reference in the CLB.

4.1.2 PROCESS TO IDENTIFY POTENTIAL MCGUIRE AND CATAWBA TIME-LIMITED AGING ANALYSES

The plant specific documents that were reviewed to identify potential TLAA for both McGuire and Catawba included the following:

- Duke / NRC Licensing Correspondence
- NUREG – 0422, as Supplemented, Safety Evaluation Report for McGuire Nuclear Station [Reference 4.1 - 1]
- NUREG – 0954, as Supplemented, Safety Evaluation Report for Catawba Nuclear Station [Reference 4.1 - 2]
- Updated Final Safety Analysis Report (for both McGuire and Catawba) [References 4.1 - 3, and 4.1 - 4]
- Improved Technical Specifications (for both McGuire and Catawba)
- Facility Operating Licenses (for both McGuire and Catawba)

The document set used for the search is contained in the Electronic Licensing Library (ELL). ELL contains over 30, 000 documents and consists of virtually all correspondence between Duke Energy (formerly Duke Power Company) and the NRC (and its predecessor the Atomic Energy Commission).

The information developed from the review of plant-specific source documents was reviewed to determine which calculations and analyses meet all six criteria of §54.3. The analyses and calculations that meet all six criteria were identified as either McGuire-specific or Catawba-specific time-limited aging analyses.

As required by §54.21(c)(1), an evaluation of each time-limited aging analyses must be performed to demonstrate that:

(1) the analyses remain valid for the period of extended operation;
(2) the analyses have been projected to the end of the period of extended operation; or
(3) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The results of the evaluations of the plant-specific time-limited aging analyses are presented in Sections 4.2 through 4.7 of this Application.

4.1.3 IDENTIFICATION OF EXEMPTIONS

Part 54 also requires that the application for a renewed license include a list of current plant-specific exemptions granted pursuant to §50.12 that are based on time-limited aging analyses as defined in §54.3.

A review of the McGuire docket has been performed and the results of this review identified that no §50.12 exemptions have been granted on the basis of a time-limited aging analysis as defined in §54.3.

A review of the Catawba docket has been performed and the results of this review identified that no §50.12 exemptions have been granted on the basis of a time-limited aging analysis as defined in §54.3.

4.1.4 References for Section 4.1

4.1 - 1. NUREG-0422, *Safety Evaluation Report Related to the Operation of the McGuire Nuclear Station, Units 1 and 2*, March 1978, as supplemented, U. S. Nuclear Regulatory Commission, Docket Nos. 50-369 and 50-370.

4.1 - 2. NUREG-0954, *Safety Evaluation Report Related to the Operation of the Catawba Nuclear Station, Units 1 and 2*, February 1983, as supplemented, U. S. Nuclear Regulatory Commission, Docket Nos. 50-413 and 50-414.

4.1 - 3. *McGuire Nuclear Station Updated Final Safety Analysis Report*, as revised.

4.1 - 4. *Catawba Nuclear Station Updated Final Safety Analysis Report*, as revised.

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The regulations governing reactor vessel integrity are in 10 CFR Part 50:

- Section 50.60 requires all light-water reactors meet the fracture toughness, pressure-temperature limits, and material surveillance program requirements for the reactor coolant boundary as set forth in Appendices G and H of 10 CFR Part 50.
- Section 50.61 contains fracture toughness requirements for protection against pressurized thermal shock.

The design bases of McGuire Nuclear Station and Catawba Nuclear Station contain calculations and analyses addressing the effects of neutron irradiation embrittlement of the reactor vessels. The calculations that had been previously in effect for the initial 40-year license have been revised and updated to address the additional twenty years of operation being requested for license renewal. These updated calculations assume that licensed activities will continue to be conducted in accordance with the facilities' current licensing basis (e.g., use of low enriched uranium dioxide fuel only). The results of these updated calculations are summarized below.

4.2.1 UPPER-SHELF ENERGY

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 §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.* These results meet the requirement of §54.21(c)(ii).

McGuire Nuclear Station –The USE values for McGuire Units 1 and 2 reactor vessel beltline materials at 54 Effective Full Power Years (EFPY) are presented in Table 4.2-1 for McGuire Unit 1 and in Table 4.2-2 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. Forty-eight (48) EFPY is a traditional value assumed to be the effective full power years that would be reached at the end of the period of extended operation using an assumed capacity factor. Fifty-four (54) EFPY was used as a conservative estimate of EFPY at the time this analysis was completed.

Catawba Nuclear Station – The USE values for Catawba Units 1 and 2 reactor vessel beltline materials at 54 EFPY are presented in Table 4.2-3 for Catawba Unit 1 and in Table 4.2-4 for Catawba Unit 2. All of the beltline materials in the Catawba reactor vessels have USE above the 50 ft-lb limit. The nozzle shell forging 06 is the most limiting material for Catawba Unit 1 with a 54 EFPY USE value of 61 ft-lbs. The nozzle shell plate B8804-3 is the most limiting material for Catawba Unit 2 with a 54 EFPY USE value of 51 ft-lbs. Forty-eight (48) EFPY is a traditional value assumed to be the effective full power years that would be reached at the end of the period of extended operation using an assumed capacity factor. Fifty-four (54) EFPY was used as a conservative estimate of EFPY at the time this analysis was completed.

Table 4.2-1 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 4.2-2 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

Table 4.2-3 Evaluation of Upper Shelf Energy for Catawba 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)
Nozzle Shell Forging 06	0.25	1.88	101	40	61
Intermediate Shell Forging 05 (Using Surveillance Capsule Data)	0.09	1.88	134	10	121
Lower Shell Forging 04	0.04	1.88	134	22	105
Intermediate to Lower Shell Circumferential Weld (Using Surveillance Capsule Data)	0.04	1.88	130	8	120
Nozzle Shell to Intermediate Shell Weld	0.03	1.88	92	22	72

Table 4.2-4 Evaluation of Upper Shelf Energy for Catawba 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)
Intermediate Shell Plate B8605-1	0.08	1.88	96	6.6	90
Intermediate Shell Plate B8605-2	0.08	1.88	82	22	64
Intermediate Shell Plate B8616-1	0.05	1.88	92	22	72
Lower Shell Plate B8806-1	0.06	1.88	83	22	65
Lower Shell Plate B8806-2	0.06	1.88	102	22	80
Lower Shell Plate B8806-3	0.06	1.88	105	22	82
Nozzle Shell Plate B8604-1	0.11	1.88	96	24	73
Nozzle Shell Plate B8604-2	0.11	1.88	89	24	68
Nozzle Shell Plate B8604-3	0.07	1.88	65	22	51
Nozzle Shell Longitudinal Weld Seams 101-122A, B, C	0.15	1.73 1.72	112	33	75
Nozzle Shell to Intermediate Shell Circumferential Weld Seam	0.13	1.88	102	31	70
Intermediate Shell Longitudinal Weld Seams 101-142A, B, C	0.04	1.13 1.88 1.88	146	10 11 11	131 130 130
Intermediate Shell to Lower Shell Circumferential Weld Seams	0.04	1.88	146	11	130
Lower Shell Longitudinal Weld Seams 101-124A, B, C	0.04	1.88 1.13 1.88	146	11 10 11	130 131 130

4.2.2 PRESSURIZED THERMAL SHOCK

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 §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. These results meet the requirement of §54.21(c)(ii).

McGuire Nuclear Station – The RT_{PTS} results for all beltline materials are presented in Table 4.2-5 for McGuire Unit 1 and in Table 4.2-6 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 & 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.

Catawba Nuclear Station – The RT_{PTS} results for all beltline materials are presented in Table 4.2-7 for Catawba Unit 1 and in Table 4.2-8 for Catawba Unit 2. All the beltline materials in the Catawba 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 forging 04 is the most limiting material for Catawba Unit 1 with a 54 EFPY PTS value of 55°F. The intermediate shell plate B8605-2 is the most limiting material for Catawba Unit 2 with a 54 EFPY PTS value of 133°F.

Table 4.2-5 RT PTS Calculations for McGuire Unit 1 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	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 4.2-6 RT PTS Calculations for McGuire Unit 2 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	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

Table 4.2-7 RT PTS Calculations for Catawba Unit 1 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	FF	$RT_{NDT(U)}$	Δ RT_{PTS}	M	RT_{PTS} °F
Intermediate Shell Forging 05	58	3.12	1.3	-8	75.4	34	101
→ Using Surveillance Capsule Data	28.4	3.12	1.3	-8	36.9	17	46
Lower Shell Forging 04	26	3.12	1.3	-13	33.8	33.8	55
Circumferential Weld Metal	54	3.12	1.3	-51	70.2	56	75
→ Using Surveillance Capsule Data	23.2	3.12	1.3	-51	30.2	28	7

Table 4.2-8 RT PTS Calculations for Catawba Unit 2 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	FF	$RT_{NDT(U)}$	Δ RT_{PTS}	M	RT_{PTS} °F
Intermediate Shell Plate B8605-1	51	3.16	1.3	15	66.3	34	115
→ Using Surveillance Capsule Data	44	3.16	1.3	15	57.2	17	89
Intermediate Shell Plate B8605-2	51	3.16	1.3	33	66.3	34	133
Intermediate Shell Plate B8616-1	31	3.16	1.3	12	40.3	34	86
Lower Shell Plate B8806-1	37	3.16	1.3	6	48.1	34	88
Lower Shell Plate B8806-2	37	3.16	1.3	-10	48.1	34	72
Lower Shell Plate B8806-3	37	3.16	1.3	8	48.1	34	90
Intermediate, Lower and Intermediate to Lower Shell Weld Seams	37.3	3.16	1.3	-80	48.5	48.5	17
→ Using Surveillance Capsule Data	33.4	3.16	1.3	-80	43.4	28	-9

4.2.3 PRESSURE-TEMPERATURE (P-T) LIMITS

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 have been performed on each reactor vessel to address projected operation during the period of extended operation. These results meet the requirement of §54.21(c)(ii).

Forty-eight (48) EFPY is a traditional value assumed to be the effective full power years that would be reached at the end of the extended period of operation using an assumed capacity factor. When these analyses were completed, slightly more conservative assumptions were made regarding EFPY to ensure that if actual capacity factors were greater than those assumed, the analysis results would remain bounding.

McGuire Nuclear Station – 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.

Catawba Nuclear Station – For Catawba Unit 1 and Unit 2 the heatup and cooldown limit curves for normal operation at 51 EFPY provide a predicted operating window that is sufficient to conduct heatups and cooldowns.

4.3 METAL FATIGUE

4.3.1 ASME SECTION III, CLASS 1 COMPONENT FATIGUE

The issue of design assumptions associated with thermal fatigue of Reactor Coolant System components has been identified as a time-limited aging analysis for both McGuire Nuclear Station and Catawba Nuclear Station. Metal fatigue is a time-limited aging analysis because all six of the criteria contained in §54.3 are met. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation by the *Thermal Fatigue Management Program* for each station.

4.3.1.1 Thermal Fatigue Management Program

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

The *Thermal Fatigue Management Program* is an existing plant program. Its purpose is to manage the thermal fatigue basis for those component evaluations that include analyses containing explicit thermal cycle count assumptions. To assure these analyses remain valid for the period of extended operation, specific programmatic actions are required.

The components managed by this program are the ones shown to be acceptable by analyses that explicitly addressed thermal fatigue transient limits. As defined for license renewal, the *Thermal Fatigue Management Program* is a prevention program in that it seeks to preclude cracking due to low-cycle thermal fatigue by managing the thermal fatigue basis. Management of the thermal fatigue basis is accomplished by continually showing that the severity and number of occurrences of the transients actually occurring are enveloped by the severity and number of occurrences of the analyzed transients.

Scope – The scope of the *Thermal Fatigue Management Program* is the set of transients for the component scope requiring thermal fatigue basis confirmation for license renewal. The component scope from which the transient set is determined is:

- Reactor Coolant System Class 1 components (including piping connected to the RCS falling under the purview of NRC I.E. Bulletins 88-08 and 88-11). Fatigue environmental effects will also be addressed for the Class 1 components.

- The replacement steam generators (RSG) where the analysis set includes not only transients associated with the Class 1 portion of these steam generators, but also the transients applicable to certain non-Class 1 portions of these steam generators.

- Components falling within the *Inservice Inspection Plan* that contain flaws detected during inservice inspection (ISI) that exceeded acceptance standards, but were shown to be acceptable by fracture mechanics analyses (FMA) explicitly addressing an assumed set of thermal transient cycle limits.

- For Catawba only, four non-Class 1 heat exchangers whose design basis was assumed to be established by considering the specific thermal transient cycle limits established for the Reactor Coolant System.

The set of transients that have been assembled for this scope of components addresses each transient described in each plant's UFSAR:

McGuire Updated Final Safety Analysis Report (UFSAR) Section 5.2 and listed in UFSAR Table 5-2, "Summary of Reactor Coolant Design Transients" and Table 5-49, "BWI Replacement Steam Generator." [Reference 4.3 - 1]

Catawba Updated Final Safety Analysis Report (UFSAR) Section 3.9.1.1 and listed in UFSAR Table 3-50, "Design Transients for ASME Code Class I Piping." [Reference 4.3 - 2]

Preventive Actions – Cracking due to thermal fatigue of locations specifically designed to preclude such cracking is prevented by assuring that the thermal fatigue basis remains valid for the period of extended operation. The actions taken by the *Thermal Fatigue Management Program* are based on reliance in the standards established in ASME Section III and ASME Section XI.

Parameters Monitored or Inspected – The *Thermal Fatigue Management Program* contains specific actions which will assure the thermal fatigue basis remains valid for the period of extended operation. These actions are discussed in detail under **Monitoring & Trending** and/or **Corrective Action & Confirmation Process**.

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

Monitoring & Trending – The three key actions of the *Thermal Fatigue Management Program* are:

1. Determining the Thermal Cycles to be Monitored and Their Character and Number of Allowed Occurrences: As described under **Scope,** the set of transient events to be managed by the *Thermal Fatigue Management Program* is derived from the associated component information. Included are their thermal and pressure profile characteristics and the minimum of the numbers of occurrences used in the evaluations. As updates occur to

associated component information such as analyzed conditions, operational practices, inservice inspection results, or, fatigue environmental effect modifications required for the extended period of operation (after 40 years), the set of transients and their limits may require revision.

2. Monitoring the Thermal Cycles Experienced: From continual monitoring of plant operating conditions, the responsible engineer will discover plant conditions that meet the definition of a transient cycle defined by this program. Upon discovery of each transient cycle required to be documented by the program, the responsible engineer will tabulate the cycle count and enter it into a database. For those events that are logged, the *Thermal Fatigue Management Program* specifies appropriate parameters such as minimum/maximum temperature limits and rates of temperature change that are assumed in the analysis. The logging process captures these values for review.

3. Comparison of Observed Events to Allowable Events: When the responsible engineer determines which transients have occurred since the previous assessment, two evaluations are performed to determine if parameters are within limits. The first evaluation compares the observed values for those parameters applicable to each transient to the limits described in the *Thermal Fatigue Management Program* (e.g. a maximum or minimum temperature limit). The second evaluation is a comparison to the allowable number of occurrences. The database information allows a comparison of the accumulated cycles to the overall allowable cycles.

Acceptance Criteria – The program acceptance criteria are to maintain the actual thermal cycle transient count within overall allowable limits for the defined transients and for each transient, to maintain actual transient thermal and pressure profile characteristics within the limits analyzed.

Corrective Action & Confirmation Process – Should the thermal and pressure profile for a specific transient be outside of the parameters defined for that transient set or should an allowable cycle count limit for a transient cycle set be approached or exceeded, the program requires that the responsible engineer identify the issue to the appropriate engineering group(s) for resolution. The corrective action program is triggered immediately if profile values are exceeded. Similarly, the corrective action program is triggered if the number of events is expected to exceed the thermal fatigue basis limits within a manageable time period. A manageable time period is the time needed to complete actions to ensure the affected components stay within acceptable cycle count limits.

Similarly, if the cycle count limit is found to be actually lower than tabulated in the UFSAR and/or other source locations, (as through ISI activities, plant modifications or re-analyses), then the corrective action program is again used to document such discrepancies and track the resolution.

Administrative Controls – The *Thermal Fatigue Management Program* actions are implemented by station work processes. These actions are required by McGuire Technical Specification 5.5.6 for Reactor Coolant System Class 1 components covered in the McGuire UFSAR, Section 5.2 and by Catawba Technical Specification 5.5.6 for Reactor Coolant System Class 1 components covered in the Catawba UFSAR Section 3.9.1.1.

Operating Experience – Thermal fatigue transients have been tracked since operation began at both McGuire and Catawba. Operating experience associated with the *Thermal Fatigue Management Program* has demonstrated that that the program continues to monitor plant transients and track the accumulation of these transients consistent with the requirements in Technical Specification 5.5.6.

Conclusions – The *Thermal Fatigue Management Program* has been demonstrated to be capable of programmatically managing the set of transients for the component scope requiring thermal fatigue basis confirmation for license renewal. The *Thermal Fatigue Management Program* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 4.2.3 Reference [4.3 - 3]. Based on the above review, the continued implementation of the *Thermal Fatigue Management Program* provides reasonable assurance that thermal fatigue will be managed such that the components will continue to perform their intended function(s) for the period of extended operation. This result meets the requirement of §54.21(c)(iii).

4.3.1.2 Fatigue Environmental Effects

Note: The process to evaluate the environmental effects of fatigue is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

4.3.1.2.1 BACKGROUND

The topic of fatigue environmental effects were previously associated with NRC Generic Safety Issue (GSI) 190, *Fatigue Evaluation of Metal Components for 60-Year Life*, more informally known as environmentally-assisted fatigue (EAF). This issue began under GSI-166, *Adequacy of Fatigue Life of Metal Components*, concerning all operating plants. In the December 31, 1997 version of NUREG-0933, *A Prioritization of Generic Safety Issues*, GSI-166 was closed for the current operating period (i.e. 40 year plant life) and was re-issued as GSI-190 for plants considering the 20 additional years associated with license renewal. The environmental fatigue issue for license renewal was finalized during the close-out of GSI-190 in December 1999. In an internal NRC memorandum [Reference 4.3 - 4], it was concluded that environmental effects would have a negligible impact on core damage frequency, and as such, no generic regulatory action was required. However, since NUREG/CR-6674 [Reference 4.3 - 5] indicated that fatigue reactor coolant environmental effects would result in an increased frequency of pipe leakage, the NRC required that utilities applying for license renewal address the effects of reactor water environment on fatigue usage in affected components.

Duke, in cooperation with the Nuclear Energy Institute (NEI), EPRI and a number of other utilities, has engaged the NRC in a dialogue associated with a means to address the concerns outlined above. In December 2000, a draft EPRI report entitled *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* [Reference 4.3- 6] aimed at establishing a standard approach to addressing these concerns was submitted to NRC. [Reference 4.3- 7]

4.3.1.2.2 DUKE APPROACH

The draft EPRI report, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application,* offers two methods for applicant utilities to address EAF for license renewal. Duke selects Method 2, Fatigue Assessment Using Environmental Factors. Briefly, Duke will follow the steps in EPRI Method 2, paraphrased and adjusted as follows:

1. Choose 6-10 plant locations for assessment. Locations for consideration will include the NUREG/CR-6260 locations and other locations expected to have high usage factors when considering EAF (See Note 1).
2. For an evaluation period, determine the EAF-adjusted CUF at these locations, using defined transient severities and/or assumed occurrences either bounding or coinciding with

realistic expectations (See Notes 2 and 3). CUF calculations may use methods less conservative than original design as appropriate. Transient definitions may coincide with design bases or may deviate as judged appropriate. Reconciliation with actual plant behavior is required in a following step.

3. Within the evaluation period, continually track the fatigue accumulating at the locations. This will be done by either tracking the incidences of the occurring transients or by taking the observed plant parameters and calculating a resulting EAF-adjusted CUF.
4. Compare either the recorded incidences of occurring transients with the number used in step 2, or compare the calculated EAF-adjusted CUF with that predicted in step 2 (See Note 3).
5. Make future projections of either the EAF-adjusted CUF or the count of transient occurrences to determine the remaining time to reaching the allowables (See Note 3). If the projections predict exceeding the allowables within a time period needed for the implementation of corrective actions to prevent this, then initiate those corrective actions at this time (See Note 1).

Note 1: The objective to meet in choosing locations will be to ensure by example that no plant location will have an EAF-adjusted CUF that exceeds 1.0 in actual operation. This will dictate that for locations chosen, any assumptions or practices followed in determining that EAF-adjusted CUFs remain below 1.0 will be applicable to those not chosen as well. For example, if a later ASME Code, or more rigorous analysis methods, or less severe transients are applied in order to show that for a given location the EAF-adjusted CUF remains < 1.0, then those same actions must be appropriate for other similar locations not explicitly chosen. If not, then representatives from the remaining population would need to be included. This includes Alternate Fatigue Management options: 3.3.1 Reanalysis, 3.3.2 Partial Cycle Counting, 3.3.3 Fatigue Monitoring, or 3.3.5 Modified Plant Operations of Reference 4.3 - 6. Employment of option 3.3.4 Flaw Tolerance and Inspection for any location shall require that similar EAF fatigue sensitive locations be found bounded by fatigue crack growth and end of period K_I (or size) for that location.

Should it not be possible to show that a projected transient accumulation rate remains less than allowable for the evaluation period chosen, corrective actions may include but not be limited to those described in Section 3.3 Alternate Fatigue Management, of Reference 4.3 - 6.

Note 2: The procedure to be used for determination of EAF-adjusted CUFs is given in Appendix B of Reference 4.3 - 6. This procedure employs an F_{en} process with an adjustment for moderate environmental effects, i.e. usage for each transient load pair is multiplied by F_{en}, where F_{en} is $F_{en,nom}/Z$, with Z=3.0 for carbon and low alloy, and 1.5

for wrought and cast stainless steels. A strain rate weighted average F_{en} may be used as described in Reference 11 of Reference 4.3 - 6.

Note 3: The choice of evaluation period will not be less than one fuel cycle, but may be as long as the time remaining to the end of the 60 year license renewal period. The objective will be to arrive at an end of period EAF-adjusted CUF that is less than 1.0, using the transient cycles expected within that time period. Care will be taken that the transient cycles used will be conservative enough that exceedance of severity or occurrences will not occur within the frequency of data collection and evaluation. Otherwise the plant will experience an un-analyzed transient (either severity or number) and plant operability evaluation and associated efforts will be triggered.

As described for the *Thermal Fatigue Management Program*, Duke will employ a monitoring scheme based on detecting, assessing, and counting plant transient events (as opposed to CUF monitoring). As such, the allowable number of occurrences for each transient then becomes the quantity assumed for the chosen evaluation period, which in turn was limited by each location's EAF-adjusted CUF remaining less than or equal to 1.0. The severity of transients are also monitored and compared to the as analyzed values. EAF-dependent variables will also either be similarly monitored and compared or be chosen such that non-exceedance is judged assured. As such, the *Thermal Fatigue Management Program* will be adjusted to include the effects of EAF only through the lowering of the allowable number of transient events and the inclusion of EAF parameters to monitor and compare.

The exercise of the above procedure will be at a time prior to the end of the 40th year of plant operation. This lead time shall be sufficient to ensure that implementation of corrective actions will prevent the exceedance of 1.0 of EAF-adjusted CUF within the extended period of operation. No requirement exists that any resulting adjustments in allowables be applied prior to the end of the initial 40 years of operation. It is recognized that a discontinuity exists at the 40 year point in the need to apply this adjustment.

Duke may choose to exercise a different course of action should the NRC approve a less restrictive approach in the future, either through agreement with the industry, or individually with us.

4.3.2 ASME Section III, Class 2 and 3 Piping Fatigue

Note: The assessment of thermal fatigue for non-Class 1 mechanical components is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

Thermal fatigue of mechanical systems is considered to be a time-limited aging analysis because all six of the criteria contained in §54.3 are satisfied. Thermal fatigue of non-Class 1 components is considered to be an effect of aging and involves time-limited assumptions defined by the current term (e.g., 7000 cycles). Thermal fatigue is relevant in making a safety determination and involves conclusions related to the capability of the component to perform its intended function. The mechanical system design requirements are contained in the applicable design Code of Record. This section addresses the non-Class 1 mechanical system components falling within the scope of license renewal. The Reactor Coolant System Class 1 components and certain other non-Class 1 components are addressed in Section 4.3.1.1**.**

As background, McGuire and Catawba have a number of systems within the scope of license renewal that were designed to the requirements of ANSI B31.1 or ASME Section III Subsection NC or Subsection ND. Piping systems designed to these requirements include a stress range reduction factor to provide conservatism in the design to account for cyclic conditions due to operations. The stress range reduction factor is 1.0 as long as the location does not exceed 7000 full temperature thermal cycles during its operation.

In order to identify the specific locations where extended operation could invalidate the existing stress range reduction factor in the piping analysis, an engineering review process was developed that considered the design temperatures and operating conditions of these McGuire and Catawba mechanical systems. These mechanical systems were reviewed to determine which ones would be likely to see 7000 equivalent full temperature thermal cycles during plant operations. Results of this engineering review determined that two specific system locations at McGuire and two specific system locations at Catawba could reach 7000 equivalent full temperature thermal cycles during the period of extended operation. Each of these locations was re-evaluated to consider a higher number of cycles and the analyses were found to be acceptable. The analyses for all other locations are valid for the period of extended operation.

For license renewal, all thermal cycle count assumptions for the non-Class 1 mechanical systems are conservatively revalidated for 60 years of operation. The conservative projection of the thermal transient cycle count assumptions to the end of the period of extended operation when coupled with the resolution of the four specific locations means that the analyses are valid for the period of extended operation. The results described above are equivalent to the corresponding results described and evaluated in NUREG-1723, Section 4.2.7 [Reference 4.3 - 5]. Therefore, the full temperature thermal cycle count assumptions in the existing non-Class 1 piping analyses for mechanical components within the scope of license renewal are valid for the period of extended operation. This result meets the requirement of §54.21(c)(i).

4.3.3 References for Section 4.3

4.3 - 1. McGuire Nuclear Station Updated Final Safety Analysis Report, as revised.

4.3 - 2. Catawba Nuclear Station Updated Final Safety Analysis Report, as revised.

4.3 - 3. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station*, Units 1, 2, and 3, March 2000, U. S. Nuclear Regulatory Commission, Docket Nos. 50-269, 50-270, and 50-287.

4.3 - 4. A. C. Thadani (NRC-RES) Memorandum to W. D. Travers (NRC-EDO) dated December 26, 1999, *Closeout of Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."*

4.3 - 5. NUREG-6674, *Fatigue Analysis of Components for 60-Year Plant Life*, June 2000, U. S. Nuclear Regulatory Commission.

4.3 - 6. Draft EPRI Report, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application,* Electric Power Research Institute.

4.3 - 7. D. J. Walters (NEI) letter dated December 13, 2000, *Industry Guidance for Addressing Fatigue Environmental Effects in a License Renewal Application*.

4.4 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

Note: Discussions of environmental qualification (EQ) of electrical components in this section are generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

The Nuclear Regulatory Commission (NRC) has established nuclear station environmental qualification (EQ) requirements in 10 CFR 50 Appendix A, Criterion 4 and in 10 CFR 50.49 [Reference 4.4 - 1]. Section 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (that is, those areas of the plant that could be subject to the harsh environmental effects of a loss-of-coolant-accident [LOCA], high energy line breaks [HELBs] or post-LOCA radiation) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. Section 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

4.4.1 MCGUIRE AND CATAWBA *ENVIRONMENTAL QUALIFICATION PROGRAM* BACKGROUND

The McGuire and Catawba *Environmental Qualification Program* meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. Section 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics and the environmental conditions to which the components could be subjected. Section 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. Section 50.49(e) also requires replacement or refurbishment of component not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. Section 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions. Sections 50.49(k) and (l) permit different qualification criteria to apply based on plant and component vintage. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in the DOR Guidelines, Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors [Reference 4.4 -2]; NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment [Reference 4.4 - 3]; and Regulatory Guide 1.89, Rev. 1, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants [Reference 4.4 - 4]. Compliance with 10 CFR 50.49 provides reasonable assurance that the component can perform its intended functions during accident conditions after experiencing the effects of inservice aging.

The McGuire and Catawba *Environmental Qualification Program* manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

Aging evaluations for electrical components in the McGuire and Catawba *Environmental Qualification Program* that specify a qualification of at least 40 years are time-limited aging analyses (TLAAs) for license renewal because all of the criteria contained in 10 CFR 54.3 are met.

Under 10 CFR §54.21(c)(1)(iii), the McGuire and Catawba *Environmental Qualification Program*, which implements the requirements of 10 CFR 50.49 (as further defined and clarified by the DOR Guidelines, NUREG-0588 and Regulatory Guide 1.89, Rev. 1), at the plants is viewed as an aging management program for license renewal. Reanalysis of an aging evaluation to extend the qualifications of components is performed on a routine basis as part of the McGuire and Catawba *Environmental Qualification Program*. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met).

TLAA demonstration option (iii), which states that the effects of aging will be adequately managed for the period of extended operation, is chosen and the McGuire and Catawba *Environmental Qualification Program* will manage the aging effects of the components associated with the environmental qualification TLAA. Section 4.4.2.1.3 of the draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR) (August 2000) states that the staff evaluated the EQ program (10 CFR 50.49) and determined that it is an acceptable aging management program to address environmental qualification according to 10 CFR 54.21(c)(1)(iii). The evaluation referred to in the draft SRP-LR contains sections on *EQ Component Reanalysis Attributes* and *Evaluation and Technical Basis*, which is the basis of the description provided below.

4.4.2 McGuire and Catawba EQ Component Reanalysis Attributes

The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the McGuire and Catawba *Environmental Qualification Program*. While a component life limiting condition may be due to thermal, radiation or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, an unrealistically low activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to McGuire and Catawba quality assurance program requirements, which requires the verification of assumptions and conclusions. As already noted, important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed below.

Analytical Methods – The McGuire and Catawba *Environmental Qualification Program* uses the same analytical models in the reanalysis of an aging evaluation as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60 year normal radiation dose is to multiply the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection & Reduction Methods – Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis per the McGuire and Catawba *Environmental Qualification Program*. Temperature data used in an aging evaluation should be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are to be justified on a plant-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.

Underlying Assumptions – McGuire and Catawba *Environmental Qualification Program* EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria & Corrective Action – Under the McGuire and Catawba *Environmental Qualification Program*, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner (that is, sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).

4.4.3 MCGUIRE AND CATAWBA *ENVIRONMENTAL QUALIFICATION PROGRAM* EVALUATION AND TECHNICAL BASIS

Scope – The McGuire and Catawba *Environmental Qualification Program* includes certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49.

Preventive Actions – Section 50.49 does not require actions that prevent aging effects. McGuire and Catawba *Environmental Qualification Program* actions that could be viewed as preventive actions include (a) establishing the component service condition tolerance and aging limits (for example, qualified life or condition limit), and (b) where applicable, requiring specific installation, inspection, monitoring or periodic maintenance actions to maintain component aging effects within the bounds of the qualification basis.

Parameters Monitored or Inspected – The qualified life of a component in the McGuire and Catawba *Environmental Qualification Program* is not based on condition or performance monitoring. However, pursuant to Regulatory Guide 1.89 Rev. 1, such monitoring programs are an acceptable basis to modify a qualified life through reanalysis. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

Detection of Aging Effects – Section 50.49 does not require the detection of aging effects for in-service components. As implemented by the McGuire and Catawba *Environmental Qualification Program*, monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

Monitoring and Trending – Section 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. McGuire and Catawba *Environmental Qualification Program* actions that could be viewed as monitoring include monitoring how long qualified components have been installed. Monitoring or inspection of certain environmental, condition, or component parameters may be used to ensure that a component is within the bounds of its qualification basis or as a means to modify the qualification.

Acceptance Criteria – Section 50.49 acceptance criteria, as implemented by the McGuire and Catawba *Environmental Qualification Program*, are that an inservice EQ component is maintained within the bounds of its qualification basis, including (a) its established qualified life and (b) continued qualification for the projected accident conditions. Section 50.49 requires refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device. When monitoring is used to modify a component qualified life, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification methods.

Corrective Action & Confirmation Process – If a component in the McGuire and Catawba *Environmental Qualification Program* is found to be outside the bounds of its qualification basis, corrective actions are implemented in accordance with the station's corrective action program. When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. When an emerging industry aging issue is identified that affects the qualification of an EQ component, the affected component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. Confirmatory actions, as needed, are implemented as part of the McGuire and Catawba corrective action program, pursuant to 10 CFR 50, Appendix B.

Administrative Controls – The McGuire and Catawba *Environmental Qualification Program* is implemented through the use of station policy, directives, and procedures. The McGuire and Catawba *Environmental Qualification Program* will continue to comply with 10 CFR 50.49 throughout the renewal period, including development and maintenance of qualification documentation demonstrating reasonable assurance that a component can perform required functions during harsh accident conditions. McGuire and Catawba *Environmental Qualification Program* documents identify the applicable environmental conditions for the component locations. McGuire and Catawba *Environmental Qualification Program* qualification files are maintained at McGuire and Catawba in an auditable form for the duration of the installed life of the component. McGuire and Catawba *Environmental*

Qualification Program documentation is controlled under the station's quality assurance program.

Operating Experience – EQ programs include consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended functions during accident conditions after experiencing the effects of inservice aging.

Conclusion

The McGuire and Catawba *Environmental Qualification Program* has been demonstrated to be capable of programmatically managing the qualified lives of the components falling within the scope of the program for license renewal. The McGuire and Catawba *Environmental Qualification Program* described above is equivalent to the corresponding program described and evaluated in Section 4.2.8 of NUREG-1723 [Reference 4.4 - 5]. Based on the above review, the continued implementation of the McGuire and Catawba *Environmental Qualification Program* provides reasonable assurance that the aging effects will be managed and that EQ components will continue to perform their intended functions for the period of extended operation. This result meets the requirement of §54.21(c)(iii).

4.4.4 References for Section 4.4

4.4 - 1. 10 CFR 50.49, *Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants*, Office of the Federal Register, National Archives and records Administration, 2000.

4.4 - 2. DOR Guidelines, *Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors*, November 1979.

4.4 - 3. NUREG-0588, *Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment*, U. S. Nuclear Regulatory Commission, July 1981.

4.4 - 4. NRC Regulatory Guide 1.89, Rev. 1, *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,* U. S. Nuclear Regulatory Commission, June 1984.

4.4 - 5. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station*, Units 1, 2, and 3, March 2000, U. S. Nuclear Regulatory Commission, Docket Nos. 50-269, 50-270, and 50-287.

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4.5 CONCRETE CONTAINMENT TENDON PRESTRESS (NOT APPLICABLE)

The prestressing tendons in prestressed concrete containments lose their prestressing forces with time due to creep and shrinkage of concrete, and relaxation of the prestressing steel. During the design phase, engineers estimate these losses to arrive at the predicted prestressing forces at the end of operating life, normally forty years. The loss of tendon prestress analysis is a time-limited aging analysis for only prestressed concrete containments.

This topic is not applicable to either the McGuire Nuclear Station Ice Condenser Containments or the Catawba Nuclear Station Ice Condenser Containments. Ice condenser containments do not use prestressed tendons.

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4.6 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATION FATIGUE ANALYSIS

The containment liner plates, metal containments, penetration sleeves (including dissimilar metal welds), and penetration bellows may be designed in accordance with requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. If a plant's code of record requires a fatigue analysis, then this fatigue analysis may be a TLAA and must be evaluated to ensure that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

4.6.1 CONTAINMENT LINER PLATES

Prestressed concrete containments have containment liner plates. McGuire and Catawba are ice condenser containments and do not have containment liner plates. Therefore, the topic of a fatigue analysis for containment liner plate is not applicable for either McGuire Nuclear Station or Catawba Nuclear Station.

4.6.2 METAL CONTAINMENTS

The McGuire and Catawba ice condensers containments are metal containments, also know as Steel Containment Vessels (SCV). These Steel Containment Vessels are described in the Section 2.4 of this Application. The design Code of Record for the McGuire SCV is Section B, Section III, of the ASME Boiler and Pressure Vessel Code, 1968 Edition, including all addenda and code cases through Summer 1970 [Reference 4.6 - 1, Section 3.8.2.2]. For Catawba, the design Code of Record is Subsection NE, Section III, of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including all addenda through the Summer of 1972 [Reference 4.6 - 2, Section 3.8.2.2].

Typical hot penetration assemblies are shown on UFSAR Figure 3-68 for McGuire and Figure 3-13 for Catawba. The hot penetration consists of three major components: a) process line and flued head, b) guard pipe, and c) expansion bellows [Reference 4.6 - 1, Section 3.9.2.8.2 and Reference 4.6 - 2, Section 3.6.2.4.2]. Of these three major components, the bellows design accommodates displacements between the SCV and Reactor Building due to thermal, seismic, and containment test conditions [Reference 4.6 - 1, Section 3.9.2.8.2 and Reference 4.6 - 2, Section 3.6.2.4.2]. Fatigue is a progressive failure of a structural part under repeated, cyclic, or fluctuating loads. Because of the bellows design, the piping loads which could cause fatigue are not transferred to the SCV. No fatigue analysis was required for the SCV; therefore, containment fatigue is not a time-limited aging analysis for either McGuire Nuclear Station or Catawba Nuclear Station.

4.6.3 BELLOWS

Mechanical penetrations are provided with bellows to accommodate differential movement between the containment and the Reactor Building for thermal, seismic, and containment test conditions [Reference 4.6 - 1, Section 3.9.2.8.2 and Reference 4.6 - 2, Section 3.6.2.4.2]. Typical details of bellows are shown in McGuire UFSAR Figure 3-68 and Catawba Figure 3-13. All bellows expansion joints are of two-ply construction with a wire mesh between plys for testability of bellows and bellows weld to piping [Reference 4.6 - 1, Section 3.9.2.8.1 and Reference 4.6 - 2, Section 3.6.2.4.1]. The design and time-limited aging analysis evaluation are plant specific as described below.

4.6.3.1 McGuire Design and Time-Limited Aging Analysis Evaluation

The McGuire bellows assemblies are manufactured, installed, and examined in accordance with ASME Section III 1971 edition paragraph NC-3649. Design requirements for the bellows are contained in McGuire engineering documents.

The 1971 Code required the manufacturer to consider the combined stress due to pressure and deflection versus the stress to failure at the design cyclic life as a part of the bellows design. The cyclic life data was based on actual experiments in which bellows designs, similar to those installed, were cycled to failure. In addition, the design code included a 175% safety margin in the bellows design. This requirement assured a conservative design. A search of Duke engineering records did not locate any manufacturer’s records for a fatigue calculation on the original design for McGuire.

During later modifications at McGuire, the bellow’s manufacturer reviewed the bellows design for revised feedwater penetration movements. The manufacturer calculated the new cycle life for the Feedwater bellows to be 32,782 cycles. Although not part of the original design, this cycle life is well beyond what the bellows would see during normal operation. In order for the bellows to cycle 32,782 times over 60 years, the bellows would have to cycle more than once a day.

Fatigue of bellows is not a time-limited aging analysis for McGuire Nuclear Station because Criterion (4) of §54.3 is not met. The bellows fatigue analysis was determined not to be relevant by Duke in making any safety determination.

While fatigue of bellows has not been determined to be a time-limited aging analysis for McGuire, the aging effect which could result from cyclic fatigue, cracking, has been identified as an aging effect requiring management for the bellows for the period of extended operation. Local leak rate testing has been identified as the program that manages cracking of the bellows. The local leak rate testing is discussed as part of *Containment Leak Rate Testing Program*.

4.6.3.2 Catawba Design and Time-Limited Aging Analysis Evaluation

The Catawba bellows assemblies are manufactured, installed, and examined in accordance with ASME Section III 1974 edition paragraph NC-3649. Design requirements for the bellows are contained in Catawba engineering documents.

The Code requirements for bellows design evolved throughout the design of McGuire and Catawba. The 1974 edition ASME Section III paragraph NC-3649 added the requirement that the cumulative effect of stress cycles be evaluated for the cyclic life. The additional requirements were adopted from the Expansion Joint Manufacturers Association (EJMA), Inc. These requirements also assured a conservative design.

The manufacturer provided calculations for the cyclic life evaluation for each type of penetration. These cycle life values were used by the manufacturer to demonstrate that the design met the Code requirements.

Fatigue of bellows is not a time-limited aging analysis for Catawba Nuclear Station because Criterion (4) of §54.3 is not met. Although a cyclic life evaluation was performed by the vendor for the bellows, the bellows fatigue analysis was determined not to be relevant by Duke in making any safety determination.

While fatigue of bellows has not been determined to be a time-limited aging analysis for Catawba, the aging effect which could result cyclic fatigue, cracking, has been identified as an aging effect requiring management for the bellows for the period of extended operation. Local leak rate testing has been identified as the program that manages cracking of the bellows. The local leak rate testing is discussed as part of *Containment Leak Rate Testing Program*.

4.6.4 REFERENCES FOR SECTION 4.6

4.6 - 1. *McGuire Nuclear Station Updated Final Safety Analysis Report*, as revised.

4.6 - 2. *Catawba Nuclear Station Updated Final Safety Analysis Report*, as revised.

4.7 OTHER PLANT SPECIFIC TIME-LIMITED AGING ANALYSES

4.7.1 REACTOR COOLANT PUMP FLYWHEEL FATIGUE

The reactor coolant pump motors at McGuire and Catawba are of the same design. The reactor coolant pump motors are large, vertical, squirrel cage, induction motors. The motors have flywheels to increase rotational-inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing and inside the motor frame. The aging effect of concern is fatigue crack initiation in the flywheel bore key way from stresses due to starting the motor. Therefore, this topic is considered to be a time-limited aging analysis for license renewal because all of the criteria contained in §54.3 are met.

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. Reaching 6000 starts in 60 years would require a pump start on average every 3.7 days. Since a pump start normally occurs every 200 to 300 days, on average, the design of the reactor coolant pump flywheels is conservative. In addition, crack growth from postulated flaws in each flywheel is only a few mils [Reference 4.7 - 1]. The existing analysis is valid for the period of extended operation, meeting the requirement of §54.21(c)(i).

4.7.2 LEAK-BEFORE-BREAK ANALYSES

Leak-before-break (LBB) 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 both McGuire Nuclear Station and Catawba Nuclear Station.

As background, the successful application of LBB to the McGuire Reactor Coolant System primary loop piping is described in WCAP-10585, *Technical Basis For Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis For McGuire Units 1 and 2* [Reference 4.7 - 2]. Likewise, the successful application of LBB to the Catawba Reactor Coolant System primary loop piping is described in WCAP-10546, *Technical Basis For Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis For Catawba Units 1 and 2* [Reference 4.7 - 3]. These reports provide the technical basis for evaluating postulated flaw growth in the main Reactor Coolant System piping under normal plus faulted loading conditions.

The first analysis consideration that could be influenced by time is the material properties of the cast austenitic stainless steel. Cast austenitic stainless steels used in the Reactor Coolant System are subject to thermal aging during service. This thermal aging causes an elevation in the yield strength of the material and a degradation of the fracture toughness, the degree of degradation being a function of the level of ferrite in the material. Thermal aging in these stainless steels will continue until a saturation or fully aged point is reached. WCAP-10456, *The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems* [Reference 4.7 - 4] presented a detailed study of the effects of thermal aging on piping integrity. This report concluded that the thermal aging process does not significantly change the failure characteristics of the cast stainless steel piping. WCAP-10585 (McGuire) and WCAP-10546 (Catawba) used the findings of this report to make the determination that the material properties in WCAP-10456 were bounding for McGuire and Catawba. Fully aged, lower bounding data was used in performing the leak-before-break evaluation. Additionally during the license renewal review, the lower bound data in WCAP-10456 was compared to the lower bound data in NUREG 6177 [Reference 4.7 - 5] and found to be comparable. Therefore, because the original analysis supporting leak-before-break relied on fully aged stainless steel material properties, the analysis does not have a material property time-dependency that requires further evaluation for license renewal.

The second analysis consideration that could be influenced by time is the accumulation of actual fatigue transient cycles over time that could invalidate the fatigue flaw growth analysis that was done as part of the leak-before-break analysis. A review of the accumulation of the applicable fatigue transient cycles is considered to meet the TLAA definition. This review has been done within the scope of the *Thermal Fatigue Management Program*. The *Thermal Fatigue Management Program* has been demonstrated to be capable of programmatically managing the Class 1 thermal fatigue design basis, including the assumptions in the leak-before-break analysis, for the period of extended operation. The *Thermal Fatigue Management Program* is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 4.2.3. The continued implementation of the *Thermal Fatigue Management Program* provides reasonable assurance that thermal fatigue will be managed for the Class 1 components such that they will continue to perform their intended function(s) for the period of extended operation. This result meets the requirement of §54.21(c)(iii) for both McGuire Nuclear Station and Catawba Nuclear Station.

4.7.3 DEPLETION OF NUCLEAR SERVICE WATER POND VOLUME DUE TO RUNOFF

McGuire Nuclear Station – The depletion of Nuclear Service Water Pond Volume due to runoff time-limited aging analysis is not applicable to McGuire. The drainage area serving the McGuire Nuclear Service Water Pond is such that the run-off and resulting sedimentation are negligible. The volume of the McGuire Nuclear Service Water Pond been previously reviewed and accepted by the NRC in the initial McGuire Safety Evaluation Report, Section 4.2 [Reference 4.7 - 6].

Catawba Nuclear Station – The Standby Nuclear Service Water (SNSW) Pond is a nuclear safety related impoundment constructed by placing a dam across a small cove of Lake Wylie. Because of the design of the Standby Nuclear Service Water Pond, an analysis was performed to predict the total loss of volume in the pond due to sedimentation during the 40 year plant life. This analysis is described in the Catawba UFSAR Section 2.4.8 [Reference 4.7 - 7] and the Catawba SER Section 2.4.4.2 [Reference 4.7 - 8]. The analysis estimated that the Standby Nuclear Service Water Pond volume would be depleted by about 10 acre-feet of sediment during the 40-year plant life.

Because all of the criteria contained in §54.3 are met, the sedimentation of the Standby Nuclear Service Water Pond over time is a time-limited aging analysis for Catawba Nuclear Station. TLAA demonstration option (iii), which states that the effects of aging will be adequately managed for the period of extended operation, is chosen to manage the Standby Nuclear Service Water Pond sedimentation TLAA. The *Standby Nuclear Service Water Pond Volume Program* manages the volume of water in the pond.

Catawba Technical Specification [SR] 3.7.9.1 requires that the water level of the SNSW pond remain greater than or equal to 571 feet mean seal level. This requirement ensures that a sufficient volume of water is available to allow the nuclear service water system to operate for at least 30 days following the design basis LOCA. SNSW Pond level is monitored and makeup water is provided should the pond level drop to 571.5 feet. Technical Specification 3.7.9 requires immediate makeup to restore level or the station is shutdown. The minimum allowable includes margin to account for evaporation and the use of SNSW Pond water for fire protection, assured auxiliary feedwater, assured component cooling makeup, and assured fuel pool makeup for the full 30 days after a postulated accident [Reference 4.7 - 7, Section 9.2.5.4].

Catawba UFSAR Figure 9-54 contains the area volume curves which are used in the thermal analysis for the ultimate heat sink. The UFSAR also includes a commitment that soundings will be taken around the SNSW intake structure at 5 year intervals to assure that sediment deposits will not adversely affect the operation of the nuclear service water system. Although an earlier calculation for the volume of the Standby Nuclear Service Water (SNSW) Pond was documented, more recent calculations have been performed which validate the volume of water in the SNSW Pond.

Scope – The scope of the *Standby Nuclear Service Water Pond Volume Program* includes the volume of water in the SNSW Pond.

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

Parameters Monitored or Inspected – The *Standby Nuclear Service Water Pond Volume Program* requires a topographic survey of the pond to determine the topography of the bottom of the SNSW Pond. Calculations are then performed using the survey data to determine the volume of water within the SNSW Pond.

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

Monitoring & Trending – The design parameter (volume of water within the SNSW Pond) is validated using the *Standby Nuclear Service Water Pond Volume Program*. Conventional methods of surveying and volume calculation are used. A contour map with a known scales is developed as a result of the survey. Areas at the different elevations are determined. Using the contour intervals and the area at each contour interval, volumes are computed for each contour elevation. The computed (surface) areas and the volume of water (below the specified pond surface elevations) at each contour elevation are compared to the areas and volumes in Figure 9-54 in the Catawba UFSAR to ensure that an adequate volume of water is available.

The *Standby Nuclear Service Water Pond Volume Program* is performed once every three years.

The *Standby Nuclear Service Water Pond Volume Program* is documented and retained in sufficient detail to permit adequate confirmation of the results. The accountable engineer is responsible for reviewing the findings and determining whether or not the results are acceptable.

Acceptance Criteria – The acceptance criteria are contained in the area-volume curve shown in Catawba UFSAR Figure 9-54. Calculated areas and volumes are compared to the criteria in Figure 9-54.

Corrective Action & Confirmation Process – Where the calculations do not meet the acceptance criteria, the results are deemed unacceptable and documented under the corrective action program.

Administrative Controls – The level of the SNSW Pond is governed by Technical Specification (SR) 3.7.9.1. The survey of the SNSW Pond is implemented in accordance with plant procedures as required by Technical Specification 5.4.

Operating Experience – Previous surveys and calculations have verified that the (surface) area and volume of water in the Standby Nuclear Service Water Pond is sufficient.

Conclusion

The design parameter (volume of water within the SNSW Pond) is validated using the *Standby Nuclear Service Water Pond Volume Program*. For license renewal, the TLAA associated with the volume of water in the Standby Nuclear Service Water Pond is resolved in accordance with §54.21(c)(1)(iii). Based on the above review, it is reasonable to expect that the continued implementation of the *Standby Nuclear Service Water Pond Volume Program* will verify that the volume of water within the SNSW Pond is consistent with the current licensing basis throughout the period of extended operation for Catawba Nuclear Station.

4.7.4 REFERENCES FOR SECTION 4.7

4.7 - 1. WCAP-14535A, November 1996, *Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination*, Section 4.3.1, Westinghouse Electric Corporation.

4.7 - 2. WCAP-10585, *Technical Basis For Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis For McGuire Units 1 and 2,* June 1984, Westinghouse Electric Corporation.

4.7 - 3. WCAP-10546, *Technical Basis For Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis For Catawba Units 1 and 2,* June 1984, Westinghouse Electric Corporation.

4.7 - 4. WCAP-10456, *The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems*, November, 1983, Westinghouse Electric Corporation.

4.7 - 5. NUREG-6177, *Assessment of Thermal Embrittlement of Cast Stainless Steels,* May 1994, U. S. Nuclear Regulatory Commission.

4.7 - 6. NUREG-0422, *Safety Evaluation Report Related to the Operation of the McGuire Nuclear Station, Units 1 and 2,* March 1978, as supplemented, U. S. Nuclear Regulatory Commission, Docket Nos. 50-369 and 50-370.

4.7 - 7. Catawba Nuclear Station Updated Final Safety Analysis Report, as revised.

4.7 - 8. NUREG-0954, *Safety Evaluation Report Related to the Operation of the Catawba Nuclear Station, Units 1 and 2,* February 1983, as supplemented, U. S. Nuclear Regulatory Commission, Docket Nos. 50-413 and 50-414.

Appendix A

UFSAR Supplements

Appendix A1 – McGuire Nuclear Station UFSAR Supplement

Appendix A2 – Catawba Nuclear Station UFSAR Supplement

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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.

Changes to Existing Chapters 3 and 5

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	$RT_{NDT(U)}$	Δ RT $_{PTS}$	M	RT $_{PTS}$ ^oF
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

Changes to Existing Chapters 3 and 5

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.

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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 5^{th} and 6^{th} 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.

Appendix A-2

Catawba Nuclear Station UFSAR Supplement

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Changes to Existing Chapters 2, 3 and 5

The following paragraph is contained in Section 2.4.8, Cooling Water Canals and Reservoirs:

> "Using the grain size analysis for site soils, it is calculated that approximated two-thirds of the sediment entering the pond would settle rapidly (L < 310 ft). This would lead logically to the formation of delta deposits in the head water areas of the pond. This conclusion is substantiated by data from existing reservoirs, presented in Reference 20. The one-third or approximately 1.7 ac-ft of sediment which does not deposit in the upper regions of the pond can be assumed to distribute uniformly in the form of bottom set deposits in the vicinity of the dam and SNSW intake structure. This amount of deposition should have no adverse impact on the operation of the SNSW intake structure. Soundings will be taken around the SNSW intake structure prior to fuel loading and at 5 year intervals thereafter to assure that sediment deposits will not adversely affect the operation of the NSW System."

This paragraph provides a summary description of the Standby Nuclear Service Water Pond Volume Program. No changes are necessary for license renewal.

Changes to Existing Chapters 2, 3 and 5

Make the following changes to Section 3.5.1.2.1, 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 2, 3 and 5

Add the following paragraphs to Section 3.9.1, Special Topics for Mechanical Components:

Fatigue Evaluation for License Renewal
Catawba Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 3.9.1, to assure that components are maintained within design limits. This requirement is managed by the Catawba *Thermal Fatigue Management Program*. For license renewal, continuation of the Catawba *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 Catawba. The leak before break analyses have been determined to be acceptable for the period of extended operation.

Changes to Existing Chapters 2, 3 and 5

Add the following paragraph to Section 3.9.3, ASME Code Class 2 and 3 Components:

Evaluation for License Renewal

Catawba 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 2, 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 2, 3 and 5

Add the following paragraphs to Section 5.3.3, Reactor Vessel Integrity:

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 Catawba Unit 1 and in Table 5-X for Catawba Unit 2. All the beltline materials in the Catawba 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 forging 04 is the most limiting material for Catawba Unit 1 with a 54 EFPY PTS value of 55°F. The intermediate shell plate B8605-2 is the most limiting material for Catawba Unit 2 with a 54 EFPY PTS value of 133°F.

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

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	FF	$RT_{NDT(U)}$	Δ RT_{PTS}	M	RT_{PTS} °F
Intermediate Shell Forging 05	58	3.12	1.3	-8	75.4	34	101
→ Using Surveillance Capsule Data	28.4	3.12	1.3	-8	36.9	17	46
Lower Shell Forging 04	26	3.12	1.3	-13	33.8	33.8	55
Circumferential Weld Metal	54	3.12	1.3	-51	70.2	56	75
→ Using Surveillance Capsule Data	23.2	3.12	1.3	-51	30.2	28	7

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

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	FF	$RT_{NDT(U)}$	ΔRT_{PTS}	M	RT_{PTS} °F
Intermediate Shell Plate B8605-1	51	3.16	1.3	15	66.3	34	115
→ Using Surveillance Capsule Data	44	3.16	1.3	15	57.2	17	89
Intermediate Shell Plate B8605-2	51	3.16	1.3	33	66.3	34	133
Intermediate Shell Plate B8616-1	31	3.16	1.3	12	40.3	34	86
Lower Shell Plate B8806-1	37	3.16	1.3	6	48.1	34	88
Lower Shell Plate B8806-2	37	3.16	1.3	-10	48.1	34	72
Lower Shell Plate B8806-3	37	3.16	1.3	8	48.1	34	90
Intermediate, Lower and Intermediate to Lower Shell Weld Seams	37.3	3.16	1.3	-80	48.5	48.5	17
→ Using Surveillance Capsule Data	33.4	3.16	1.3	-80	43.4	28	-9

Changes to Existing Chapters 2, 3 and 5

Add the following paragraphs to Section 5.3.3, Reactor Vessel Integrity:

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 Catawba Units 1 and 2 reactor vessel beltline materials at 54 EFPY are presented in Table 5-Y for Catawba Unit 1 and in Table 5-Z for Catawba Unit 2. All of the beltline materials in the Catawba reactor vessels have USE above the 50 ft-lb limit. The nozzle shell forging 06 is the most limiting material for Catawba Unit 1 with a 54 EFPY USE value of 61 ft-lbs. The nozzle shell plat B8804-3 is the most limiting material for Catawba Unit 2 with a 54 EFPY USE value of 51 ft-lbs.

Table 5-Y Evaluation of Upper Shelf Energy for Catawba 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)
Nozzle Shell Forging 06	0.25	1.88	101	40	61
Intermediate Shell Forging 05 (Using Surveillance Capsule Data)	0.09	1.88	134	10	121
Lower Shell Forging 04	0.04	1.88	134	22	105
Intermediate to Lower Shell Circumferential Weld (Using Surveillance Capsule Data)	0.04	1.88	130	8	120
Nozzle Shell to Intermediate Shell Weld	0.03	1.88	92	22	72

Table 5-Z Evaluation of Upper Shelf Energy for Catawba 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)
Intermediate Shell Plate B8605-1	0.08	1.88	96	6.6	90
Intermediate Shell Plate B8605-2	0.08	1.88	82	22	64
Intermediate Shell Plate B8616-1	0.05	1.88	92	22	72
Lower Shell Plate B8806-1	0.06	1.88	83	22	65
Lower Shell Plate B8806-2	0.06	1.88	102	22	80
Lower Shell Plate B8806-3	0.06	1.88	105	22	82
Nozzle Shell Plate B8604-1	0.11	1.88	96	24	73
Nozzle Shell Plate B8604-2	0.11	1.88	89	24	68
Nozzle Shell Plate B8604-3	0.07	1.88	65	22	51
Nozzle Shell Longitudinal Weld Seams 101-122A, B, C	0.15	1.73 1.72	112	33	75
Nozzle Shell to Intermediate Shell Circumferential Weld Seam	0.13	1.88	102	31	70
Intermediate Shell Longitudinal Weld Seams 101-142A, B, C	0.04	1.13 1.88 1.88	146	10 11 11	131 130 130
Intermediate Shell to Lower Shell Circumferential Weld Seams	0.04	1.88	146	11	130
Lower Shell Longitudinal Weld Seams 101-124A, B, C	0.04	1.88 1.13 1.88	146	11 10 11	130 131 130

Changes to Existing Chapters 2, 3 and 5

Add the following paragraph to Section 5.3.2, Pressure-Temperature Limits:

Pressure – Temperature Limits 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 Catawba have been performed on each reactor vessel to address projected operation during the period of extended operation. For Catawba Unit 1 and Unit 2, the heatup and cooldown limit curves for normal operation at 51 EFPY provide a predicted operating window that is sufficient to conduct heatups and cooldowns.

New Chapter 18

Insert new UFSAR Chapter 18 to read as follows:

18 Aging Management Programs and Activities

18.1 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

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 Table 18-1 identifies where the description of the Program, Activity, or TLAA is located in either the Catawba UFSAR or in the Catawba Improved Technical Specifications (ITS).

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

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

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 3.8.4.4 SLC 16.7-9.2 SLC 16.7-9.4
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
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
Flow Accelerated Corrosion Program	B.3.14	18.2.9

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

Topic	*Application Location*	*UFSAR / ITS Location*
Fluid Leak Management Program	B.3.15	18.2.10
Galvanic Susceptibility Inspection	B.3.16	18.2.11
Heat Exchanger Preventive Maintenance Activities	B.3.17	18.2.12
Ice Condenser Inspections	B.3.18	ITS 3.6.12 18.2.13
Inaccessible Non-EQ Medium Voltage Cables Aging Management Program	B.3.19	18.2.14
Inservice Inspection Plan	B.3.20	18.2.15
Inspection Program for Civil Engineering Structures and Components	B.3.21	18.2.16
Leak-Before-Break	4.7.2	3.9.1
Liquid Waste System Inspection	B.3.22	18.2.17
Metal Fatigue	4.3	3.9.1, 3.9.3
Non-EQ Insulated Cables and Connections Aging Management Program	B.3.23	18.2.18
Preventive Maintenance Activities	B.3.24	18.2.19
Reactor Coolant Pump Flywheel	4.7.1	3.5.1.2.1
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	18.2.20
Reactor Vessel Internals Inspection	B.3.27	18.2.21
Reactor Vessel Neutron Embrittlement	4.2	5.3.3

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

Topic	*Application Location*	*UFSAR / ITS Location*
Selective Leaching Inspection	B.3.28	18.2.22
Service Water Piping Corrosion Program	B.3.29	18.2.23
Standby Nuclear Service Water Pond Dam Inspection	B.3.30	ITS SR 3.7.8.3
Standby Nuclear Service Water Pond Volume Program	4.7.3	2.4.8
Steam Generator Surveillance Program	B.3.31	ITS 5.5.9
Sump Pump Systems Inspection	B.3.32	18.2.24
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	18.2.25
Underwater Inspection of Nuclear Service Water Structures	B.3.35	18.2.26
Waste Gas System Inspection	B.3.36	18.2.27

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 Catawba, this review will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1). The results of this review will be incorporated into the unit specific inservice inspection (ISI) plans for the ISI intervals during the period of extended operation.

18.2.2 BORATED WATER SYSTEMS STAINLESS STEEL INSPECTION

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

- Containment Spray
- Refueling Water

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

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

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

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

One of twelve possible locations at Catawba will be inspected using a volumetric technique. If no parameters are known that would distinguish the susceptible locations at Catawba, one of the twelve available at Catawba 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 Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

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

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

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

18.2.3 BOTTOM-MOUNTED INSTRUMENTATION THIMBLE TUBE INSPECTION PROGRAM

Scope – The scope of the *Bottom Mounted Instrumentation Thimble Tube Inspection Program* includes all 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 Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Due to length of time in operation, it is expected that Unit 1 results will provide a leading indicator for Unit 2. 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 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 Action & Confirmation Process – Structures and components that do not meet the acceptance criteria are evaluated by engineering for continued service and repaired as required. Structures and components which are deemed unacceptable are documented under the corrective action program. Specific corrective actions and confirmatory actions are implemented in accordance with the corrective action program.

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

18.2.8 FIRE PROTECTION PROGRAM

Elements of the *Fire Protection Program* that serve to manage aging are implemented in accordance with Selected Licensee Commitments (See Table 18-1). 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 Catawba, this sample disassembly inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

18.2.9 FLOW ACCELERATED CORROSION PROGRAM

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

- Auxiliary Feedwater
- Auxiliary Steam
- Boron Recycle
- Feedwater
- Liquid Radwaste
- Steam Generator Blowdown Recycle
- Turbine Exhaust

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

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

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

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

Monitoring & Trending – The program is consistent with the basic guidelines or recommendations provided by EPRI document NSAC-202L [Reference 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 Catawba component design code of record.

Defined inspection locations exist in the several systems within the scope of license renewal. Auxiliary Feedwater and Feedwater and Steam Generator Blowdown Recycle each contain multiple inspection locations in susceptible regions. Other defined inspection locations cover several systems that are exposed to the same steam supply environment. Auxiliary Steam, Boron Recycle and Liquid Radwaste systems are all part of the same steam supply that spans these several systems. The steam is supplied from Auxiliary Steam and several inspection locations exist in this run of piping.

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

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

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

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

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

18.2.10 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 – 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.

18.2.11 GALVANIC SUSCEPTIBILITY INSPECTION

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

- Condenser Circulating Water
- Diesel Generator Room Sump Pump
- Exterior Fire Protection
- Interior Fire Protection
- Liquid Radwaste
- Nuclear Service Water
- Waste Gas

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

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

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

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

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

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

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

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

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

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

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

18.2.12 HEAT EXCHANGER ACTIVITIES

18.2.12.1 COMPONENT COOLING HEAT EXCHANGERS

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

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

18.2.12.2 CONTAINMENT SPRAY HEAT EXCHANGERS

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

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

18.2.12.3 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

18.2.12.4 Control Area Chilled Water

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

18.2.12.5 Diesel Generator Engine Starting Air

The purpose of the *Heat Exchanger Preventive Maintenance Activities – Diesel Generator Engine Starting Air* is to manage loss of material for parts of the diesel generator engine starting air aftercoolers exposed to 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 ICE CONDENSER ENGINEERING INSPECTION

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

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

Scope – The scope of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* includes inaccessible (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 Catawba, the first test per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criteria for each test performed per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* are defined by the specific type of test performed and the specific cable tested.

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

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

18.2.15 INSERVICE INSPECTION PLAN

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

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

- ASME Section XI, Subsection IWB and IWC (secondary side of steam generators) Inspections
- ASME Section XI, Subsection IWF Inspections
- Small Bore Piping

SMALL BORE PIPING

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

The risk-informed approach is based on WCAP 14572 Revision 1 - NP-A [Reference 18 - 10] and consists of the following two essential elements: (1) a degradation mechanism evaluation is performed to assess the failure potential of the piping under consideration, and (2) a consequence evaluation is performed to assess the impact on plant risk in the event of a piping failure. As is required by WCAP 14572 Revision 1- NP-A, the Catawba risk-informed submittals will provide equivalent or better risk coverage for the Risk-Informed-Inservice Inspection Scope.

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

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 - 11] 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 Catawba.

18.2.16 INSPECTION PROGRAM FOR CIVIL ENGINEERING STRUCTURES AND COMPONENTS

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

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

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

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*.

18.2.17 LIQUID WASTE SYSTEM INSPECTION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

18.2.18 NON-EQ INSULATED CABLES AND CONNECTIONS AGING MANAGEMENT PROGRAM

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

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

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

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

Monitoring & Trending – Accessible non-EQ insulated cables and connections installed in the Reactor Buildings, Auxiliary Building and Turbine Building are visually inspected per the *Non-EQ Insulated Cables and Connections Aging Management Program* at least once every 10 years. EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments* [Reference 18 - 12], is used as guidance in performing the inspections.

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

For Catawba, the first inspection per the *Non-EQ Insulated Cables and Connections Aging Management Program* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

18.2.19 PREVENTIVE MAINTENANCE ACTIVITIES

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 REACTOR VESSEL INTEGRITY PROGRAM

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

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

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

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

Monitoring & Trending – Each reactor vessel had six specimen capsules located in guide baskets welded to the outside of the neutron shield pads and were positioned directly opposite the center portion of the core. Each capsule contains reactor vessel steel specimens oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting shell plate located in the core region of the Catawba Unit 2 reactor vessel. Catawba Unit 1 reactor vessel specimen are oriented both parallel and normal to the major working direction of the limiting core region shell forging. Associated weld metal and weld heat affected zone metal specimens are also included in each capsule. Capsule withdrawal schedules for the Catawba Units are provided in Table 18-2.

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

Fluence Received by the Specimens – Dosimeters such as Ni, Cu, Fe, Co-Al, shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are contained in the capsules. The dosimeters permit evaluation of the flux seen by the specimens. In addition, thermal monitors made of low melting point alloys are included to monitor the temperature of the specimens. A description of the methodology used to evaluate fluence received by the specimens using dosimetry measurements and fluence calculations, assuming the same neutron spectrum at the specimens and the vessel inner wall, is described in Catawba UFSAR, Sections 5.3.1.6.1 and 5.3.1.6.2 [Reference 18 - 13]. 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 - 14]. 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 18-2

Catawba Reactor Vessel Capsule Withdrawal Schedule

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

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

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

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

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

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

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

18.2.22 SELECTIVE LEACHING INSPECTION

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

- Exterior Fire Protection
- Interior Fire Protection

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

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

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

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

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

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

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

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

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

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

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

18.2.23 SERVICE WATER PIPING CORROSION PROGRAM

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

- Exterior Fire Protection
- Interior Fire Protection
- Nuclear Service Water

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

- Containment Spray
- Control Area Chilled Water
- Diesel Generator Cooling Water
- Diesel Generator Engine Starting Air

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 - 15] 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.

18.2.24 SUMP PUMP SYSTEMS INSPECTION

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

- Diesel Generator Room Sump Pump System
- Groundwater Drainage System
- Turbine Building Sump Pump System

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

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

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

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

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

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

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

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

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

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

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

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

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

18.2.25 TREATED WATER SYSTEMS STAINLESS STEEL INSPECTION

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

- Containment Valve Injection Water
- Drinking Water
- Solid Radwaste

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

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

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

Monitoring & Trending – The *Treated Water Systems Stainless Steel Inspection* at Catawba will inspect stainless steel components, welds, and heat affected zones, as applicable, in the Drinking Water System. The Drinking Water System receives water from the local municipality that has contaminants in excess of limits below which a concern would not exist for cracking and loss of material in stainless steel. Because of the higher starting level of contaminants, the environment in the Drinking Water System is more likely to lead to cracking or loss of material if it is occurring and bounds the environments of the Containment Valve Injection Water and Solid Radwaste Systems. In addition to the volumetric examination, a visual examination of the interior of a valve will be conducted to determine the presence of pitting corrosion. Therefore, the inspection results will serve as a leading indicator and can be applied to the Containment Valve Injection Water and Solid Radwaste Systems.

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

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

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

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

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

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

18.2.26 UNDERWATER INSPECTION OF NUCLEAR SERVICE WATER STRUCTURES

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

- Low Pressure Service Water Intake Structure
- Nuclear Service Water Intake Structure
- Nuclear Service Water Pump Structure
- Standby Nuclear Service Water Discharge Structures
- Standby Nuclear Service Water Intake Structure
- Standby Nuclear Service Water Pond Outlet

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

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

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

Monitoring & Trending – The *Underwater Inspection of Nuclear Service Water Structures* detects aging effects through visual examination. The inspection is performed every Unit 1 refueling outage for Catawba Nuclear Service Water and Standby Nuclear Service Water Intake structures and five years for other Catawba structures. No actions are taken as part of this program to trend inspection or test results.

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.

18.2.27 WASTE GAS SYSTEM INSPECTION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

18 - 13. *Catawba Nuclear Station Updated Final Safety Analysis Report,* as revised.

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

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

Appendix B

Aging Management Programs and Activities

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APPENDIX B: AGING MANAGEMENT PROGRAMS AND ACTIVITIES

B.1 INTRODUCTION

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

§54.29 Standards for issuance of a renewed license

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

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

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

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

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

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

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

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B.2 PROGRAM AND ACTIVITY ATTRIBUTES

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

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

B.2.1 TYPES OF PROGRAMS AND ACTIVITIES

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

One-time Inspections

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

For McGuire, these new inspections will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, these new inspections will be completed following issuance of the renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1). Completion of these new inspections prior to the end of the initial licenses is consistent with conclusions previously made by the Nuclear Regulatory Commission in NUREG-1723.

Prevention Programs

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

Mitigation Programs

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

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

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

B.2.2 ATTRIBUTE DEFINITIONS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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B.3 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

B.3.1 ALLOY 600 AGING MANAGEMENT REVIEW

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

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

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

For McGuire, this review will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this review will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1). The results of this review will be incorporated into the unit specific inservice inspection (ISI) plans for the ISI intervals during the period of extended operation.

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B.3.2 BATTERY RACK INSPECTIONS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

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B.3.3 BORAFLEX MONITORING PROGRAM

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

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B.3.4 BORATED WATER SYSTEMS STAINLESS STEEL INSPECTION

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

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

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

- Containment Spray
- Refueling Water

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

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

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

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

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

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

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

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

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

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

Conclusion

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

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B.3.5 BOTTOM-MOUNTED INSTRUMENTATION THIMBLE TUBE INSPECTION PROGRAM

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

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

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

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

Parameters Monitored or Inspected – The *Bottom Mounted Instrumentation Thimble Tube Inspection* monitors tube wall degradation of the BMI thimble tubes. Failure of the thimble tubes would result in a breach of the reactor coolant pressure boundary.

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

Monitoring & Trending – Inspection of the BMI thimble tubes is performed using eddy current testing. All of the thimble tubes are inspected. The frequency of examination is based on an analysis of the data obtained using wear rate relationships that are predicted based on Westinghouse research that is presented in WCAP-12866, *Bottom Mounted Instrumentation Flux Thimble Wear* [Reference B - 11]. These wear rates, as well as the results of the eddy current examinations are documented in site specific calculations. The eddy current results are trended and inspections are planned prior to the refueling outage in which thimble tube wear is predicted to exceeding the ***Acceptance Criteria***, below. This ensures that the thimble tubes continue to perform their pressure boundary function.

Acceptance Criteria – The acceptance criteria for the BMI thimble tubes is 80% through wall (thimble tube wall thickness is not less than 20% of initial wall thickness). This acceptance criteria was developed by Westinghouse in WCAP 12866, "Bottom Mounted Instrumentation Flux Thimble Wear," and reported to the NRC by Duke [Reference B - 10].

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

Administrative Controls – Data are collected and evaluated using written procedures. The data are evaluated and the timing for the next inspection are determined using engineering calculations using methodology based on the information Westinghouse developed in WCAP-12866 [Reference B - 11].

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

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

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

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

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

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

Conclusion

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

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B.3.6 CHEMISTRY CONTROL PROGRAM

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

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B.3.7 CONTAINMENT INSERVICE INSPECTION PLAN – IWE

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

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

The *Containment Inservice Inspection Plan – IWE* was developed to implement applicable requirements of 10 CFR 50.55a. Section 50.55a(g)(4) requires that throughout the service life of nuclear power plants, components which are classified as either Class MC or Class CC pressure retaining components and their integral attachments must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME Code and Addenda that are incorporated by reference in §50.55a(b). Furthermore, §50.55a(g)(4)(v)(A) requires that metal containment pressure retaining components and their integral attachments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC. These requirements are subject to the limitation listed in paragraph (b)(2)(vi) and the modifications listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of §50.55a, to the extent practical within the limitations of design, geometry and materials of construction of the components [Reference B - 20]. The *Containment Inservice Inspection Plan - IWE* is a condition monitoring program.

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

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

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

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

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

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

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

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

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

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

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

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

Operating Experience –

McGuire Operating Experience

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

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

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

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

Catawba Operating Experience

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

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

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

Conclusion

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

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B.3.8 CONTAINMENT LEAK RATE TESTING PROGRAM

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

<u>Conclusion</u>

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

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

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

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

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

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

Parameters Monitored or Inspected – The *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* monitors cracking of nickel based alloy nozzles with partial penetration welds in the reactor vessel closure head.

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

Monitoring & Trending – The *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* will inspect the control rod drive mechanism type penetrations, the head vent penetration and the auxiliary head vent penetration. This program will consist of both visual and volumetric examinations.

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

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

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

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

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

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

Corrective Action & Confirmation Process – For the visual inspection, if leakage is detected the leakpath will be determined and repairs completed. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

For the volumetric examination, indications detected during volumetric examination which can not be justified for continued service by analysis will be repaired in accordance with ASME Section XI. Flaws which can be justified for continued service will be managed by the station specific *Thermal Fatigue Management Program* described in Section 4.3.1 of this Application. Specific corrective actions and confirmation will be implemented in accordance with the *Thermal Fatigue Management Program*.

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

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

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

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

Conclusion

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

B.3.10 CRANE INSPECTION PROGRAM

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

Loss of material has been identified as an aging effect requiring management for crane rails and girders for the period of extended operation. The *Crane Inspection Program* is credited with managing loss of material for the steel rails and girders within the scope of license renewal. This program has been in effect for many years at the Duke nuclear plants and is based on the guidance contained in ANSI B30.2.0 [Reference B - 27] for cranes, ANSI B30.16 [Reference B - 28] for hoists, and the requirements contained in 29 CFR Chapter XVII, 1910.179 [Reference B - 29]. The *Crane Inspection Program* is a condition monitoring program.

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

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

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

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

Monitoring & Trending – The *Crane Inspection Program* detects aging effects through visual examination of the crane rails and girders. Inspection procedures for cranes and hoists are identified in plant procedures and are in accordance with industry standards, plant experience, and other industry experience. Each crane and hoist is subject to several inspections. Prior to initial use, all new, reinstalled, altered, modified, extensively repaired and newly erected cranes are inspected and the results of the inspections are documented. Additional inspections are conducted prior to crane operation, quarterly, and/or annually depending on the specific crane or hoist. The inspection frequencies for the cranes and hoists are based on the guidance provided by ANSI B30.2.0 and ANSI B30.16 and are considered acceptable. Plant experience supports the established frequency as being timely and effective. No actions are taken as part of this program to trend inspection or test results.

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

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

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

Operating Experience –

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

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

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

B.3.11 DIVIDER BARRIER SEAL INSPECTION AND TESTING PROGRAM

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

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

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

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

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

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

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

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

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

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

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

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

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

Operating Experience –

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

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

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

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

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B.3.12 FIRE PROTECTION PROGRAM

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

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

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

B.3.12.1 Fire Barrier Inspections

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

B.3.12.2 Mechanical Fire Protection Component Tests and Inspections

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

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

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

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

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

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

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

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

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

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

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

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

For McGuire, this sample disassembly inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this sample disassembly inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

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

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

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

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

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

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

B.3.13 FLOOD BARRIER INSPECTION

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

B.3.14 FLOW ACCELERATED CORROSION PROGRAM

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

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

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

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

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

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

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

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

Monitoring & Trending – The program is consistent with the basic guidelines or recommendations provided by EPRI document NSAC-202L [Reference B - 34]. Component wall thickness is measured using volumetric examinations such as ultrasonic testing and radiography. Visual examinations are also employed when access to interior surfaces is allowed by component design. Component wall thickness acceptability is judged in accordance with the McGuire and Catawba component design code of record.

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

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

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

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

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

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

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

Conclusion

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

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B.3.15 FLUID LEAK MANAGEMENT PROGRAM

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

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

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

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

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

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

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

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

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

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

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

Acceptance Criteria – The external surfaces of structures and components within the scope of the *Fluid Leak Management Program*, including surroundings (e.g., insulation and floor areas), are expected to be free from pitting and corrosion, abnormal discoloration or accumulated residues that may be evidence of leakage from proximate borated water systems.

Corrective Action & Confirmation Process – When the programmatic activities described in the *Fluid Leak Management Program* lead to detection of an unacceptable condition, the following corrective actions are required:

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

Specific corrective actions are implemented in accordance with the *Fluid Leak Management Program* or the corrective action program. These programs apply to all structures and components within the scope of the *Fluid Leak Management Program*.

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

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

Conclusion

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

B.3.16 GALVANIC SUSCEPTIBILITY INSPECTION

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

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

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

- Condenser Circulating Water
- Containment Ventilation Cooling Water (MNS Only)
- Diesel Generator Room Sump Pump
- Exterior Fire Protection
- Interior Fire Protection
- Liquid Radwaste (CNS Only)
- Nuclear Service Water
- Waste Gas

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

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

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

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

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

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

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

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

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

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

Conclusion

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

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B.3.17 HEAT EXCHANGER ACTIVITIES

B.3.17.1 Component Cooling Heat Exchangers

B.3.17.1.1 PERFORMANCE TESTING ACTIVITIES – COMPONENT COOLING HEAT EXCHANGERS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

B.3.17.1.2 HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – COMPONENT COOLING

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

B.3.17.2 Containment Spray Heat Exchangers

B.3.17.2.1 PERFORMANCE TESTING ACTIVITIES – CONTAINMENT SPRAY HEAT EXCHANGERS

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

B.3.17.2.2 HEAT EXCHANGER PREVENTIVE MAINTENANCE ACTIVITIES – CONTAINMENT SPRAY

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

B.3.17.3 Diesel Generator Engine Cooling Water Heat Exchangers

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

<u>Conclusion</u>

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Conclusion

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

B.3.18 ICE CONDENSER INSPECTIONS

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

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

- Ice Basket Inspection
- Ice Condenser Engineering Inspection

B.3.18.1 Ice Basket Inspection

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

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

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

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

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

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

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

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

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

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

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

Operating Experience –

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

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

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

<u>Conclusion</u>

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

B.3.18.2 Ice Condenser Engineering Inspection

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

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

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

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

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

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

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

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

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

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

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

Operating Experience –

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

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

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

Conclusion

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

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

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

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

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

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

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

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

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

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

For McGuire, the first test per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, the first test per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criteria for each test performed per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* are defined by the specific type of test performed and the specific cable tested.

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

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

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

Conclusion

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

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B.3.20 INSERVICE INSPECTION PLAN

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

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

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

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

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

The Inservice Inspection Plan includes the following inspections and activities:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

<u>Conclusion</u>

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

B.3.20.1.1 McGUIRE UNIT 1 COLD LEG ELBOW

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

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

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

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

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

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

B.3.20.1.2 SMALL BORE PIPING

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

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

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

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

B.3.20.2 ASME Section XI, Subsection IWF Inspections

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

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

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

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

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

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

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

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

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

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

Conclusion

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

B.3.21 INSPECTION PROGRAM FOR CIVIL ENGINEERING STRUCTURES AND COMPONENTS

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

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

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

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

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

<u>McGuire Nuclear Station</u>

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

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

Catawba Nuclear Station

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Operating Experience –

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

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

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

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

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

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B.3.22 LIQUID WASTE SYSTEM INSPECTION

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

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

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

- Component Cooling System (MNS only)- the portion of the Component Cooling System of concern is the stainless steel waste evaporator package exposed to an unmonitored treated water environment of the Liquid Waste Recycle System;
- Liquid Waste Recycle System (MNS)- stainless steel components exposed to an unmonitored borated water environment;
- Liquid Radwaste System (CNS)- stainless steel components exposed to an unmonitored borated water, unmonitored treated water, or a raw water environment; carbon steel and cast iron components exposed to a raw water environment.

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

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

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

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

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

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

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

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

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

For carbon steel, cast iron, and stainless steel components at Catawba exposed to raw water environments, the sample population will include components located in and around the

Liquid Radwaste System sumps. The inspection results will be applied to carbon steel, cast iron, and stainless steel components in the raw water environments.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

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

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

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

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

Conclusion

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

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

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

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

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

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

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

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

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

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

For McGuire, the first inspection per the *Non-EQ Insulated Cables and Connections Aging Management Program* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, the first inspection per the *Non-EQ Insulated Cables and Connections Aging Management Program* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

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

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

Administrative Controls – The *Non-EQ Insulated Cables and Connections Aging Management Program* will be controlled by plant procedures.

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

Conclusion

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

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B.3.24 Preventive Maintenance Activities

B.3.24.1 Condenser Circulating Water System Internal Coating Inspection

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Corrective Action & Confirmation Process – Engineering evaluation is performed to determine whether the coating and base metal continue to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls –*Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* is controlled by plant procedures and work processes. The procedures and work processes provide steps for performance of the activities and require documentation of the results.

Operating Experience –

McGuire Nuclear Station
At McGuire, one complete inspection has been performed on Units 1 and 2 intake and discharge piping and on the Low-Level Intake piping from Cowans Ford Dam through the Low-Level Intake Structure to the Main Intake within the last five years. The internal coating

was observed to be in good condition with random minor defects and corrosion. The Condenser Circulating Water System intake and discharge piping has experienced two leaks. One leak was a crack that developed in a weld near the low-level intake pumps as a result of one or two water hammer events. A pinhole was discovered during a visual inspection of the low-level intake piping. The diameter of the pinhole was larger on the outside diameter than the inside diameter, indicating that the corrosion initiated on the external surface of the pipe. No inspection was performed on the external surface of the pipe. The pinhole was repaired with a steel pipe plug.

Catawba Nuclear Station

At Catawba, the Condenser Circulating Water System is scheduled to be entered every outage for blasting and recoating and/or a walkdown of areas that are not recoated. This work is being performed because the original interior coating was not properly applied and is failing. These recoating and walkdown inspections have not identified any through-wall pits originating from the exterior of the pipe. Upon completion of the recoating work, Catawba will go to a five-year inspection frequency.

External Surface

During the Catawba Unit 1 outage in the fall of 2000, piping in the Nuclear Service Water System was cleaned to remove the fouling buildup from the pipe walls. Internal inspection of accessible areas after the cleaning discovered a row of small through-wall pits. Excavation of the pipe and examination of the external coating revealed that the coating had been cut during construction allowing the underground environment to contact the external surface. Except for the cut, plant personnel noted the external coating was in good shape. Plant personnel have also identified other instances of externally generated through-wall leaks of buried components by other means that have been attributed to construction-related damage.

<u>Conclusion</u>

The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* has been demonstrated to be capable of managing loss of material of the internally coated carbon steel intake and discharge piping of the Condenser Circulating Water System by maintaining an intact protective coating. The above description has also demonstrated that this inspection is capable of managing loss of material and cracking of externally coated buried components by providing symptomatic evidence of the condition of the external coating and its ability to protect the external surfaces. The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.2.10 [Reference B - 5]. Based upon the above review, the continued implementation of the *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* provides reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

B.3.24.2 Refueling Water Storage Tank Internal Coating Inspection

Note: The PREVENTIVE MAINTENANCE ACTIVITIES – REFUELING WATER STORAGE TANK INTERNAL COATING INSPECTION is applicable only to McGuire Nuclear Station.

The purpose of the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* is to manage loss of material of the internal surfaces of the carbon steel refueling water storage tanks. The internal carbon steel surfaces of the refueling water storage tank are coated with a phenolic epoxy paint that prevents borated water and air from contacting the internal surfaces. Continued presence of an intact coating precludes loss of material of the internal surfaces of the carbon steel refueling water storage tank that could lead to loss of pressure boundary function. This preventive maintenance activity inspects the internal coating of the refueling water storage tanks to check the condition of the coating and to identify coating failures. The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* is a condition monitoring program.

Scope – The scope of the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* is the internal surface of the McGuire Units 1 and 2 carbon steel refueling water storage tanks in the Refueling Water System.

The comparable refueling water storage tanks at Catawba are constructed of stainless steel. The *Borated Water Systems Stainless Steel Inspection* (B.3.4) and the *Chemistry Control Program* (B.3.6) are credited with managing the aging effects of the stainless refueling water storage tanks at Catawba.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* inspects the phenolic epoxy paint for signs of blistering, chipping, peeling, and missing paint as well as signs of corrosion of the underlying carbon steel tank.

Detection of Aging Effects – In accordance with the information provided under **Monitoring & Trending** below, the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* will detect loss of material prior to loss of the component intended function.

Monitoring & Trending – The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* visually inspects the internal phenolic epoxy paint every ten years using an underwater video camera. The inspection looks for signs of blistering, chipping, peeling, and missing paint as well as signs of corrosion of the underlying carbon steel tank. Detection of defects in the internal coating results in draining of the tank for further inspection and evaluation of the defects.

No actions are taken as part of this activity to trend inspection results.

Acceptance Criteria – The acceptance criteria for the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* are no visual indications of coating defects that have led to corrosion of the underlying carbon steel tank surfaces.

Corrective Action & Confirmation Process – Engineering evaluation is performed to determine whether the coating and base metal continue to be acceptable. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* is controlled by plant procedures and work processes. The procedures and work processes provide steps for performance of the activities and require documentation of the results.

Operating Experience – Recently, the internal surfaces of the refueling water storage tanks for McGuire Units 1 and 2 were inspected during outage 1EOC-13 and 2EOC-12, respectively, using an underwater camera. Video results showed some second coating blistering, so the tanks were drained, visually inspected, and repainted in the necessary locations. No bare metal was exposed as a result of the blistering. A layer of the coating remained in the blistered location. The submerged portion of the tanks showed little to no degradation. However, the roof, which is not a part of the pressure boundary of the tank, did show evidence of coating concerns and was blasted and repainted in several locations. This operating experience demonstrates that this activity when continued through the extended period of operation will continue to be effective in managing loss of material of the carbon steel tank by maintaining the effectiveness of the phenolic epoxy paint.

Conclusion

The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* has been demonstrated to be capable of managing loss of material of the internal carbon steel by maintaining an intact protective coating. The *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* described above is similar to the corresponding program described and evaluated in NUREG-1723, Section 3.2.10 [Reference B - 5]. Based upon the above review, the continued implementation of the *Preventive Maintenance Activities – Refueling Water Storage Tank Internal Coating Inspection* provides reasonable assurance that the aging effects will be managed and that the component will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

B.3.25 REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE MONITORING PROGRAM

Note: The REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE MONITORING PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

The purpose of the *Reactor Coolant System Operational Leakage Monitoring Program* is to provide an additional line of defense against aging effects that may result in leakage due to cracking and loss of mechanical closure integrity. McGuire and Catawba have a continual Reactor Coolant System Technical Specification leakage limit and system surveillance requirement as defined in their Technical Specifications. The *Reactor Coolant Operational Leakage Monitoring Program* is a condition monitoring program that provides reasonable assurance that leakage will be detected prior to loss of reactor coolant system function.

Scope – The scope of the *Reactor Coolant System Operational Leakage Monitoring Program* is all Reactor Coolant components that contain coolant; however it is specifically credited with managing aging of bolted closures on the steam generators, pressurizer, and reactor coolant pumps as well as the Inconel penetrations on the reactor vessel head and steam generator tubes.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The *Reactor Coolant System Operational Leakage Monitoring Program* monitors Reactor Coolant System operational leakage and steam generator primary to secondary leakage.

Detection of Aging Effects – In accordance with the information provided in **Monitoring & Trending**, the *Reactor Coolant System Operational Leakage Monitoring Program* will detect cracking of the Reactor Coolant System pressure boundary and loss of mechanical closure integrity of bolted closures in cases where leakage is occurring.

Monitoring & Trending – The method for monitoring reactor coolant system operational leakage is specified in McGuire and Catawba Technical Specifications 3.4.13, *RCS Operational LEAKAGE*, and 3.4.15, *RCS Leakage Detection Instrumentation*.

GDC 30 of Appendix A to 10CFR50 requires means for detecting and, to the extent practical, identifying the location of the source of Reactor Coolant System leakage. Regulatory Guide 1.45 [Reference B - 45] describes acceptable methods for selecting leakage detection systems.

The primary method of detecting leakage into the Containment is measurement of the Containment floor and equipment level sump level. The sump level rate of change is calculated by the plant computer and can detect a 1 gpm leak within an hour. Leakage from the Reactor Coolant, Main Steam and Feedwater Systems can be detected this way. The containment ventilation unit condensate drain tank level change is another method of detecting leakage that is capable of detecting a 1 gpm leak. Radioactivity monitoring of particulate and gaseous radiation levels are also indicative of Reactor Coolant System leakage because the activity levels contained within the Reactor Coolant System during operation of the plant. Primary to secondary leakage from steam generator tubes can be detected by effluent monitoring (for activity) within the secondary steam and feedwater systems.

A reactor coolant water inventory balance is performed every 72 hours at steady state operation as specified in plant technical specifications to verify that leakage is within allowable limits.

Steam Generator primary to secondary leakage is monitored continuously using an operator aid computer point, radiation monitors, condensate steam air ejector off gas or secondary tritium samples depending on monitoring equipment availability and operating Mode.

Acceptance Criteria – The acceptance criteria are provided in the McGuire and Catawba Technical Specifications LCO 3.4.13, *RCS Operational LEAKAGE*.

Corrective Action & Confirmation Process – Corrective actions for this program are specified in the McGuire and Catawba Technical Specifications 3.4.13, *RCS Operation LEAKAGE*, and 3.4.15, "RCS Leakage Detection Instrumentation." Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The *RCS Operational Leakage Monitoring Program* is implemented by written procedures as required by Technical Specification 5.4.1.

Operating Experience – A search of Licensee Event Reports (LER) was performed to demonstrate the effectiveness of *the Reactor Coolant System Operational Leakage Monitoring Program* for McGuire and Catawba Nuclear Stations. Many of the LERs were maintenance issues, however several identified age-related events. Some of the issues were leakage due to loose valve bonnet bolts, leakage from an incore thermocouple fitting, a leaking compression fitting and a weld failure due to fatigue resulting from cavitation. In all of the above cases, a determination was made that the events had no significance regarding the health and safety of the public.

Another use of this program, especially prior to steam generator replacement, is monitoring of primary to secondary leakage through the steam generators. Leakage that is still within allowable limits can be monitored and a determination regarding timing of shutdown and repair of steam generator tubes can be made.

Conclusion

The *Reactor Coolant System Operational Leakage Monitoring Program* has been demonstrated to be capable of providing an additional line of defense against aging effects that may result in leakage due to cracking and loss of mechanical closure integrity. The *Reactor Coolant System Operational Leakage Monitoring Program* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *Reactor Coolant System Operational Leakage Monitoring Program* provides reasonable assurance that the aging effects will be managed and that the reactor coolant pressure boundary will continue to perform its intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

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B.3.26 REACTOR VESSEL INTEGRITY PROGRAM

Note: The Reactor Vessel Integrity Program is generically applicable to both McGuire and Catawba Nuclear Stations, except as otherwise noted.

The purpose of the *Reactor Vessel Integrity Program* is to manage the reduction of fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor vessel beltline is maintained for the period of extended operation. The program includes an evaluation of radiation damage based on pre-irradiation and post irradiation testing of Charpy V-notch and tensile specimens. The *Reactor Vessel Integrity Program* is a condition monitoring program.

Scope – The scope of the *Reactor Vessel Integrity Program* includes all reactor vessel beltline materials as defined by 10 CFR 50.61(a)(3).

Preventive Actions - No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The *Reactor Vessel Integrity Program* monitors reduction of fracture toughness of reactor vessel beltline materials by irradiation embrittlement.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending** the *Reactor Vessel Integrity Program* will detect the effects of reduction of fracture toughness prior to loss of the reactor vessel intended functions.

Monitoring & Trending – Each reactor vessel had six specimen capsules located in guide baskets welded to the outside of the neutron shield pads and were positioned directly opposite the center portion of the core. McGuire Unit 1 and Catawba Unit 2 capsules contain reactor vessel steel specimens oriented both parallel and normal (longitudinal and transverse) to the principal rolling direction of the limiting shell plate located in the core region. McGuire Unit 2 and Catawba Unit 1 reactor vessel specimens are oriented both parallel and normal to the major working direction of the limiting core region shell forging. Associated weld metal and weld heat affected zone metal specimens are also included in each capsule. Capsule withdrawal schedules for the McGuire and Catawba Units are provided in Table B.3.26-1 and Table B.3.26-2, respectively. The limiting weld material is not contained in a McGuire Unit 1 surveillance capsule, but is contained in a sister plant surveillance capsule and integrated into the McGuire Unit 1 surveillance program.

Surveillance capsule specimens are tested in accordance with approved industry standards. The test results from the encapsulated specimens represent the actual behavior of the material in the vessel. Data from testing of the surveillance capsule specimens are used to analyze Pressurized Thermal Shock, Upper Shelf Energy and to generate pressure-temperature curves for future operation of each unit. Additional information that is used to perform these analyses is as follows:

Fluence Received by the Specimens – Dosimeters such as Ni, Cu, Fe, Co-Al, shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are contained in the capsules. The dosimeters permit evaluation of the flux seen by the specimens. In addition, thermal monitors made of low melting point alloys are included to monitor the temperature of the specimens. A description of the methodology used to evaluate fluence received by the specimens using dosimetry measurements and fluence calculations, assuming the same neutron spectrum at the specimens and the vessel inner wall, is described in each station's UFSAR (McGuire UFSAR, Sections 5.4.3.7.1 and 5.4.3.7.2 [Reference B - 38] and Catawba UFSAR, Sections 5.3.1.6.1 and 5.3.1.6.2 [Reference B - 39]). The correlations have indicated good agreement and form the bases for ensuring that the calculations of the integrated flux at the vessel wall are conservative WCAP-14040 [Reference B - 46]. Projections of neutron exposure at the vessel wall to end of life are based on the assumption that irradiation data from three previous fuel cycles are representative of all future fuel cycles.

Effective Full Power Years – The effective full power years of plant operation are based on reactor vessel incore power readings. The Operator Aid Computer collects incore instrument data and reactor engineers determine effective full power year values by comparing burnup to the thermal power to calculated burnup. This data is collected continuously for all four units.

Cavity Dosimetry –The cavity dosimetry provides a method for verification of fast neutron exposure distribution within the reactor vessel beltline region and establishes a mechanism to enable long term monitoring of neutron exposure once all of the capsules have been removed from the vessel.

Monitoring of Plant Changes – Actions will be taken to ensure that the capsule data tested during the current term of operation remains valid during the period of extended operation by monitoring changes to design and operation such as the neutron spectra relative to the conditions of existing capsule data or the reactor vessel inlet temperature. These types of changes will be assessed and the applicable analyses will be updated as necessary.

Acceptance Criteria – The acceptance criteria for the *Reactor Vessel Integrity Program* are:

- Charpy specimens removed from the surveillance capsules will be laboratory tested to ensure reactor vessel fracture toughness properties exhibit upper shelf energy greater than 50 ft-lbs.
- Calculations of reference temperature for pressurized thermal shock (RT_{PTS}) must be below the screening criteria of 270°F for plates, forgings, and longitudinal welds and 300°F for circumferential welds, respectively.
- Acceptable pressure-temperature curves for heatup and cooldown of the units must be maintained in Technical Specifications
- Capsules included in the *Reactor Vessel Integrity Program* must be withdrawn as scheduled.

Corrective Action & Confirmation Process – Specific corrective action and confirmation will be implemented as follows:

- If the Charpy upper-shelf energy drops below 50 ft-lbs, it must be demonstrated that margins of safety against fracture are equivalent to those of Appendix G of ASME Section XI.
- If the projected reference temperature exceeds the screening criteria, licensees are required to submit an analysis and/or schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, licensees shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed.
- If the pressure-temperature curves are not maintained current, actions are taken as required by Technical Specifications.
- If a capsule is not withdrawn as scheduled, the NRC will be notified and a revised withdrawal schedule will be updated and submitted to the NRC.

Administrative Controls – The administrative controls that apply to *the Reactor Vessel Integrity Program* are:

- Submittal of reports required by 10 CFR Part 50 Appendix H which include a capsule withdrawal schedule, a summary report of capsule withdrawal and test results within one year of capsule withdrawal and if needed a date when a Technical Specification change will be made to change pressure-temperature limits or procedures to meet pressure-temperature limits.
- RT_{PTS} analysis will be updated as required by 10 CFR 50.61.
- Pressure-Temperature curves are maintained in the plant Technical Specifications.

- As surveillance capsules are withdrawn and either tested or stored, documentation will be updated accordingly and submitted to the NRC in accordance with 10 CFR 50, Appendix G.

Operating Experience – The *Reactor Vessel Integrity Program* monitors neutron embrittlement to assure their acceptability in accordance with NRC Regulations 10 CFR 50.60 and 10 CFR 50.61. NRC Regulation 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," requires that all light water nuclear power reactors meet the requirements of Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of Part 50. Appendix G specifies fracture toughness requirements for the reactor coolant pressure boundary to provide margins of safety against fracture during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. Appendix H requires monitoring the changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment.

Fracture toughness requirements for protection against pressurized thermal shock events are provided in 10 CFR 50.61.

McGuire and Catawba comply with the requirements of 10 CFR 50.60, Appendices G and H and 10 CFR 50.61, through the *Reactor Vessel Integrity Program*.

Conclusion

The *Reactor Vessel Integrity Program* has been demonstrated to be capable of ensuring that reactor vessel degradation is identified and corrective actions are taken prior to exceeding allowable limits. The *Reactor Vessel Integrity Program* described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *Reactor Vessel Integrity Program* provides reasonable assurance that the aging effects will be managed and that the reactor vessel will continue to perform its intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

Table B.3.26-1

McGuire Reactor Vessel Capsule Withdrawal Schedule

Unit	Capsule	Withdrawal End of Cycle (EOC)	Projected EOC Date	Estimated Fluence ($n/cm^2 \times 10^{19}$)	Reference
Unit 1	U	1	2/24/84	0.405	WCAP-10786
Unit 1	X	5	10/12/88	1.50[a]	WCAP-12354
Unit 1	V	8	3/12/93	2.08 [b][c]	WCAP-13949
Unit 1	Y	11	2/14/97	2.86 [d]	WCAP-14993
Unit 1 (dosimetry analysis & storage)	Z	8	3/12/93	2.38	WCAP-13949
Unit 1	W	16	4/5/04	4.52	STANDBY
Ex-vessel Cavity Dosimetry	N/A	12	5/29/98	1.58	WCAP-15253
Unit 2	V	1	1/25/85	0.323	WCAP-11029
Unit 2	X	5	7/5/89	1.47 [a]	WCAP-12556
Unit 2	U	7	1/9/92	2.04 [b][c]	WCAP-13516
Unit 2	W	10	4/5/96	3.07 [d]	WCAP-14799
Unit 2 (dosimetry analysis & storage)	Z	8	7/1/93	2.41	WCAP-14231
Unit 2 (dosimetry analysis & storage)	Y	8	7/1/93	2.08 [b]	WCAP-14231
Ex-vessel Cavity Dosimetry	N/A	12	3/12/99	--	WCAP-15334

a. Approximate fluence at vessel 1/4 thickness location, at 32 EFPY
b. Approximate fluence at vessel inner wall location, at 32 EFPY
c. Approximate fluence at vessel 1/4 thickness location, at 54 EFPY
d. Approximate fluence at vessel inner wall location at 54 EFPY

Table B.3.26-2

Catawba Reactor Vessel Capsule Withdrawal Schedule

Unit	Capsule	End of Cycle (EOC)	Projected EOC Date	Estimated Fluence ($n/cm^2 \times 10^{19}$)	Reference
Unit 1	Z	1	8/8/86	0.299	WCAP -1527
Unit 1	Y	6	7/10/92	1.318 [a]	WCAP-13720
Unit 1	W	14	11/29/03	3.0 [d]	--
Unit 1 (dosimetry analysis & storage)	X	10	12/20/97	2.439	WCAP-15117
Unit 1 (dosimetry analysis & storage)	U	10	12/20/97	2.439	WCAP-15117
Unit 1	V	10	12/20/97	2.334 [b][c]	WCAP-15117
Ex-vessel Cavity Dosimetry	N/A	13	5/18/02	--	--
Unit 2	Z	1	12/23/87	0.323	WCAP-11941
Unit 2	X	5	1/23/93	1.23 [a]	WCAP-13875
Unit 2	W	14	3/9/06	3.0 [d]	--
Unit 2 (dosimetry analysis & storage)	U	Standby	Standby	---	--
Unit 2 (dosimetry analysis & storage)	Y	9	9/13/98	2.38	WCAP-15243
Unit 2	V	9	9/13/98	2.38 [b][c]	WCAP-15243
Ex-vessel Cavity Dosimetry	N/A	13	10/18/04	--	--

a. Approximate fluence at vessel 1/4 thickness location, at 32 EFPY
b. Approximate fluence at vessel inner wall location, at 32 EFPY
c. Approximate fluence at vessel 1/4 thickness location, at 54 EFPY
d. Approximate fluence at vessel inner wall location at 54 EFPY

B.3.27 REACTOR VESSEL INTERNALS INSPECTION

Note: The REACTOR VESSEL INTERNALS INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

The purpose of the *Reactor Vessel Internals Inspection* is to inspect the condition of reactor vessel internals items in order to assure that the applicable aging effects will not result in loss of the intended functions of the reactor vessel internals during the period of extended operation. The reactor vessel internals stainless steel items may be separated into three groups – (1) items comprised of plates, forgings, and welds, (2) bolting (baffle-to-baffle, baffle-to-former, and barrel-to-former), and (3) items fabricated from cast austenitic stainless steel (CASS). Different aging effects will affect the various parts.

Similar reactor vessel internals inspections will be performed at other nuclear plants. Specifically, inspections are planned at Oconee Nuclear Station. In addition, the characterization of the internals aging effects through activities of EPRI and other industry groups focussed on reactor vessel internals will ensure a better understanding of the identified aging effects. These inspections and industry activities will provide significant insights prior to McGuire or Catawba entering their respective periods of extended operation.

McGuire Unit 1 will be Duke's lead Westinghouse plant for reactor vessel internals inspection since it is expected to have the most hours of operation among the four units. After the results of the McGuire Unit 1 and the Oconee inspections are evaluated, additional actions may be taken regarding McGuire Unit 2 and Catawba Units 1 and 2. The Oconee Reactor Vessel Internals Inspection is described in Oconee UFSAR Section 18.3.20 and has been evaluated in NUREG-1723, Section 3.4.3.3.

The *Reactor Vessel Internals Inspection* will supplement *The Inservice Inspection Plan* to assure that aging effects, potentially requiring additional management, will not result in loss of the intended functions of the reactor vessel internals during the period of extended operation.

Scope – The scope of the *Reactor Vessel Internals Inspection* consists of the reactor vessel internals stainless steel items that may be separated into three groups – (1) items comprised of plates, forgings, and welds, (2) bolting (baffle-to-baffle, baffle-to-former, and barrel-to-former), and (3) items fabricated from cast austenitic stainless steel (CASS).

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The *Reactor Vessel Internals Inspection* monitors the following parameters:

Visual inspections will be performed for items comprised of plates, forgings, and welds to detect cracking which could be initiated by irradiation assisted stress corrosion enhanced reduction of fracture toughness due irradiation embrittlement, and dimensional changes due to void swelling.

Volumetric inspections will be performed for bolting to detect cracking due to irradiation assisted stress corrosion enhanced by reduction of fracture toughness due to irradiation embrittlement, and loss of preload by stress relaxation due to irradiation creep.

For items fabricated from CASS, crack propagation of existing flaws caused by reduction of fracture toughness by thermal embrittlement and irradiation embrittlement.

Dimensional changes due to void swelling will be monitored in lead components for items comprised of plates, forgings, welds, and bolting.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending,** the *Reactor Vessel Internals Inspection* will detect cracking, reduction of fracture toughness, dimensional changes, and loss of preload prior to loss of the reactor vessel internals intended function(s).

Monitoring & Trending – The *Reactor Vessel Internals Inspection* includes the following inspection activities:

For plates, forgings, and welds, a visual inspection will be performed to detect the effects of cracking by irradiation assisted stress corrosion cracking enhanced by reduction of fracture toughness by irradiation embrittlement.

For baffle bolts, a volumetric inspection will be performed at McGuire Unit 1 to assess cracking.

For items fabricated from CASS, an analytical approach to assess the effect of reduction of fracture toughness on the applicable reactor vessel internals items will be performed. The specific inspection method will depend on the results of these analyses.

McGuire Unit 1 will be inspected in the fifth inservice inspection interval. The decision to perform inspections on McGuire Unit 2, Catawba Unit 1 and Catawba Unit 2 and when to perform such inspections will depend on an evaluation of the results of the internals inspections performed at Oconee and on McGuire Unit 1.

With respect to dimensional changes due to void swelling, McGuire and Catawba will rely on the results of inspections to be performed at Oconee. Items comprised of plates, forgings, and welds will be inspected at all three Oconee Units to assess the effects of void swelling. Activities are in progress to develop and qualify the inspection method. The results of the Oconee inspections will be used to determine if change in dimensions due to void swelling is a concern for the reactor vessel internals of McGuire Unit 1, McGuire Unit 2, Catawba Unit 1 and Catawba Unit 2 and if additional inspections are necessary.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The *Reactor Vessel Internals Inspection* includes the following acceptance criteria:

For the items comprised of plates, forgings, and welds, critical crack size will be determined by analysis prior to the inspection.

For baffle bolts, any detectable crack indication is unacceptable for a particular baffle bolt. The number of baffle bolts needed to be intact and their locations will be determined by analysis.

For items fabricated from CASS, critical crack size will be determined by analysis. Acceptance criteria for all aging effects will be developed prior to the inspection.

Corrective Action & Confirmation Process – If the results of the inspection are not acceptable, then actions will be taken to repair or replace the affected items or to determine by analysis the acceptability of the items. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The *Reactor Vessel Internals Inspection* will be implemented by plant procedures and the work management system.

Operating Experience – The *Reactor Vessel Internals Inspection* is a new inspection for which there is no operating experience. However, a similar inspection was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

Conclusion

The *Reactor Vessel Internals Inspection* described above is similar to and builds upon the corresponding reactor vessel internals inspection described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the implementation of the *Reactor Vessel Internals Inspection* will assure the reactor vessel internals will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operation license).

B.3.28 SELECTIVE LEACHING INSPECTION

Note: The SELECTIVE LEACHING INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

The purpose of the *Selective Leaching Inspection* is to characterize any loss of material due to selective leaching of system components exposed to raw water environments. Uncertainty exists as to whether long term exposure to raw water environments could cause loss of material due to selective leaching in brass and cast iron components such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect brass and cast iron components exposed to raw water to detect the presence and extent of any loss of material due to selective leaching. The *Selective Leaching Inspection* is a one-time inspection.

Scope – The scope of the *Selective Leaching Inspection* is the brass and cast iron components exposed to raw water in the following McGuire and Catawba systems:

- Conventional Wastewater Treatment (MNS Only)
- Diesel Generator Room Sump Pump (MNS Only)
- Exterior Fire Protection
- Groundwater Drainage (MNS Only)
- Interior Fire Protection
- Nuclear Service Water (MNS Only)

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameter inspected by the *Selective Leaching Inspection* is the hardness of the wetted surface of cast iron pump casings and brass valve bodies. Selective leaching (a form of galvanic corrosion) is the dissolution of one metal in an alloy at the metal surface which leaves a weakened network of corrosion products that is revealed by a Brinnell Hardness check or equivalent as reduction in material hardness.

Detection of Aging Effects – The *Selective Leaching Inspection* is a one-time inspection that will detect the presence and extent of any loss of material due to selective leaching.

Monitoring & Trending – Of the cast iron components in the systems above, the *Selective Leaching Inspection* will perform a Brinnell Hardness Test or an equivalent test on one cast iron pump casing in the Exterior Fire Protection System at each site. The Brinnell Hardness Test or an equivalent test is most easily performed on a pump casing and will be indicative of all cast iron components in the systems listed above. The Exterior Fire Protection System contains a raw water environment that is susceptible to selective leaching and will be bounding for the other environments in the other systems. If no parameters are known that would distinguish among the pump casings, one of the three cast iron pump casings in the

Exterior Fire Protection System at each site will be examined based on accessibility and operational concerns. The results of this inspection will be applied to the other cast iron components exposed to raw water environments in the systems listed above.

The *Selective Leaching Inspection* will also perform a Brinnell Hardness Test or an equivalent test on a sample of brass valves at each site in the Interior Fire Protection System. Valves selected for inspection should be continuously exposed to stagnant or low flow raw water environments. If no parameters are known that would distinguish the susceptible locations at each site, a select set of susceptible locations will be examined based on accessibility, operational, and radiological concerns. The results of this inspection will be applied to the brass components exposed to raw water environments in the systems listed above.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

No actions are taken as part of this program to trend inspection results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The acceptance criteria for the *Selective Leaching Inspection* is no unacceptable loss of material due to selective leaching that could result in a loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effect will not cause a loss of the component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the applicable aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The *Selective Leaching Inspection* will be implemented in accordance with controlled plant procedures.

Operating Experience – The *Selective Leaching Inspection* is a new one-time inspection for which there is no operating experience. However, a similar inspection was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

Conclusion

The *Selective Leaching Inspection* described above is similar to the Selective Leaching Inspection described and evaluated in NUREG-1723, Section 3.5.3.2 [Reference B - 5]. Based on the above review, implementation of the *Selective Leaching Inspection* will adequately verify that no need exists to manage the aging effect on the components or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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B.3.29 SERVICE WATER PIPING CORROSION PROGRAM

Note: The SERVICE WATER PIPING CORROSION PROGRAM is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

The purpose of the *Service Water Piping Corrosion Program* is to manage loss of material in order to maintain the pressure boundary function of specific raw water system components within the scope of license renewal. The program will manage the more uniform loss of material such as that due to general corrosion as well as particulate erosion in areas of higher flow velocity. Components subject to these generalized effects are made from carbon and galvanized steel, cast and ductile iron, and copper alloys in the McGuire and Catawba raw water systems. The program also will serve to manage loss of material due to localized corrosion of components made from carbon and galvanized steel, cast and ductile iron, copper alloys and stainless steel. The *Service Water Piping Corrosion Program* is a condition monitoring program.

Scope – For license renewal, the *Service Water Piping Corrosion Program* is credited with managing loss of material for components in the following systems:

- Containment Ventilation Cooling Water (MNS only)
- Exterior Fire Protection
- Interior Fire Protection
- Nuclear Service Water

Additionally, the *Service Water Piping Corrosion Program* is credited with managing loss of material for heat exchanger sub-components in the following systems:

- Containment Spray
- Control Area Chilled Water
- Diesel Generator Cooling Water
- Diesel Generator Engine Starting Air (CNS only)

Preventive Actions – No actions are taken as part of the *Service Water Piping Corrosion Program* to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The *Service Water Piping Corrosion Program* inspections are focused on carbon steel piping components exposed to raw water. Among the installed component materials, carbon steel is the more susceptible to general loss of material and serves as a leading indicator of the general material condition of the system components. Inspection of carbon steel piping provides symptomatic evidence of loss of material of other components and other materials exposed to raw water. The specific parameter monitored is pipe wall thickness as an indicator of loss of material.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending** below, the *Service Water Piping Corrosion Program* will detect the more uniform loss of material such as that due to general corrosion as well as particulate erosion that may occur in areas of higher flow velocity. The program will also detect loss of material due to localized corrosion due to crevice, pitting, and microbiologically influenced corrosion (MIC).

Monitoring & Trending – The *Service Water Piping Corrosion Program* manages all of the system components within license renewal that are susceptible to the various corrosion mechanisms and is not focused on individual components within each specific system. The intent of the *Service Water Piping Corrosion Program* is to inspect a number of locations with conditions that are characteristic of the conditions found throughout the raw water systems above. The results of these inspection locations would then be applied to similar locations throughout all the raw water systems within the scope of license renewal. This characteristic-based approach recognizes the commonality among the component materials of construction and the environment to which they are exposed.

Monitoring under the *Service Water Piping Corrosion Program* focuses on carbon steel pipe.. For components constructed of cast and ductile iron, galvanized steel and copper alloys, experience has shown that loss of material for these components will occur at a rate somewhat less than the carbon steel pipe. Therefore, the results of the carbon steel pipe inspections will provide a leading indicator of the condition of these materials.

For the carbon and galvanized steel, cast and ductile iron, and copper alloy component materials that can experience loss of material from both uniform and localized mechanisms, it is the gross material loss due to uniform mechanisms that is of primary concern under the *Service Water Piping Corrosion Program*. Gross wall loss can lead to structural instability concerns and could directly impact component intended function. Monitoring for uniform loss of material is accomplished using ultrasonic test techniques, supplemented by visual inspections if access to the interior surfaces is allowed such as during plant modifications.

When pipe wall thickness is determined by volumetric wall thickness measurements using ultrasonic testing, several measurements are taken around the circumference of the piping. These measurements are then assessed in relation to the specific acceptance criteria for that location. Because the phenomena is slow-acting, inspection frequency varies for each location. The frequency of re-inspection depends on previous inspection results, calculated rate of material loss, piping analysis review, pertinent industry events, and plant operating experience. Refer to **Acceptance Criteria** for additional details. Component results are catalogued and future inspection or component replacement schedules are determined as a part of the program.

Localized corrosion due to pitting and MIC will reveal itself through pinhole leaks in the piping components. The geometry of the pinholes means that they are not a structural integrity concern. Further, these pinhole leaks cannot individually lead to loss of the component intended function, since sufficient flow at prescribed pressures can still be provided by the system. These localized concerns will lead to structural integrity concerns only when a significant number of pinholes are present. A trend of indications of through-wall leaks due to pitting corrosion or MIC will provide evidence when localized corrosion may become a structural integrity concern and will trigger corrective actions by the *Service Water Piping Corrosion Program*. Methods in place to identify incidents of through-wall leaks are system walkdowns, operator rounds, system testing, and maintenance activities. This relevant operating experience will form the basis for any future programmatic actions with respect to pitting corrosion and MIC concerns.

While the emphasis of the *Service Water Piping Corrosion Program* remains on gross material loss, the loss of material due to localized corrosion of component materials exposed to raw water will be managed by the monitoring and trending of relevant plant operating experience of non-structural, through-wall leaks identified during various plant activities.

Acceptance Criteria – The *Service Water Piping Corrosion Program* manages loss of material for nuclear safety related and non-nuclear safety related components.

For nuclear safety-related components designed to ASME Section III, Class 3 rules, acceptance criteria are defined as meeting ASME code requirements [Reference B - 47] in order to assure structural integrity. Several factors are used to determine structural integrity at an inspection location. These factors include consideration of actual as-found wall thickness, calculated rate of material loss, use of the piping stress analyses to determine a minimum required thickness and projected time to reach the minimum wall thickness which, in turn, will establish the re-inspection interval or component replacement schedule.

For the non-nuclear safety related components that have no seismic design requirements, the acceptance criterion is the minimum wall thickness calculated on a location-specific basis. These minimum values have been determined based on design pressure or structural loading using the piping design code of record and then applying additional conservatism.

Corrective Action & Confirmation Process – Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The *Service Water Piping Corrosion Program* is governed by site specifications and implemented using controlled plant procedures and work orders. The procedures and work processes provide steps for performance of the activities and require the documentation of the results.

Operating Experience – The *Service Water Piping Corrosion Program* was formalized at each site in the early 1990's. The early investigations were conducted as a part of the efforts to address NRC Generic Letter 89-13 [Reference B - 48]. Test results have indicated mostly pitting corrosion problems. Typical corrosion rates have ranged from 3 to 5 mils per year average wall loss, but vary depending on line size and flow regime. Test locations continue to be monitored and evaluated for continued service. Piping replacements have not been required to date based on corrosion rate projections.

Refinement of the predictive capabilities of the program have been made over time and now include monitoring and trending to determine calculated rate of material loss to schedule the next inspection. Operating experience has demonstrated that using measured corrosion rates provides adequate information on the extent of loss of material to predict when replacement of components might be necessary. Stress analysis of components has been able to refine acceptance criteria and extend the life of some pipe sections. Overall the program continues to successfully manage loss of material in the raw water systems of McGuire and Catawba.

Conclusion

The *Service Water Piping Corrosion Program* has been demonstrated to be capable of managing loss of material in components exposed to a raw water environment. The program described above is equivalent to the corresponding program described and evaluated in NUREG-1723, Section 3.2.13 [Reference B - 5]. Based on the above review, the continued implementation of the *Service Water Piping Corrosion Program* provides reasonable assurance that the aging effect will be managed and that the structure(s) or component(s) will continue to perform its intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

B.3.30 STANDBY NUCLEAR SERVICE WATER POND DAM INSPECTION

Note: The STANDBY NUCLEAR SERVICE WATER POND DAM INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

Loss of material and cracking of earthen embankments have been identified as aging effects requiring management for the Standby Nuclear Service Water Pond Dam for the extended period of operation. The *Standby Nuclear Service Water Pond Dam Inspection* is credited with managing these aging effects. The *Standby Nuclear Service Water Pond Dam Inspection* is implemented per Technical Specification (SR) 3.7.8.3 for McGuire and Technical Specification (SR) 3.7.9.3 for Catawba. The *Standby Nuclear Service Water Pond Dam Inspection* is a condition monitoring program.

Scope – The scope of the *Standby Nuclear Service Water Pond Dam Inspection* includes the upstream and downstream slopes, the spillway overflow/outlet works, the area near the right and left abutments, and the toe of the dam.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – Parameters monitored during the *Standby Nuclear Service Water Pond Dam Inspection* are cracking and loss of material. The examination guidelines for the inspection are in accordance with Regulatory Guide 1.127.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending**, the *Standby Nuclear Service Water Pond Dam Inspection* will detect loss of material and cracking of the earthen embankment prior to loss of structure or component intended functions.

Monitoring & Trending – Aging effects are detected through visual examination of the dam. The *Standby Nuclear Service Water Pond Dam Inspection* visually examines the SNSW Pond Dam for erosion, settlement, slope stability, seepage, drainage systems, integrity of rip-rap, and environmental conditions. The *Standby Nuclear Service Water Pond Dam Inspection* is performed on an annual basis as required by McGuire Technical Specification (SR) 3.7.8.3 and Catawba Technical Specification (SR) 3.7.9.3. In addition, the results of the piezometric readings and settlement monitoring are reviewed. Piezometers are located on the dam to monitor foundation pore pressure. The piezometers are read quarterly. Survey monuments are located on the crest along the entire length of the dam to provide information on settlement. Surveys of the monuments are performed annually.

Inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. In particular, these records identify the inspection team and provide review of past inspection results, the results of the current inspection, whether the results were acceptable, discrepancies and their cause, and any corrective action resulting from the inspection.

Acceptance Criteria – Acceptance criteria are no visual indications of abnormal degradation, vegetation growth, erosion, or excessive seepage that would affect the Standby Nuclear Service Water Pond Dam operability.

Corrective Action & Confirmation Process – Structures and components which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repaired as required. Each inspection notes recommendations concerning repairs or studies. Structures and components which are deemed unacceptable are documented under the corrective action program. Specific corrective actions and confirmatory actions, as needed, are implemented in accordance with the corrective action program. All prior inspection reports are reviewed to ensure implementation of recommended corrective actions.

Administrative Controls – The *Standby Nuclear Service Water Pond Dam Inspection* is governed by the McGuire Technical Specification 3.7.8.3 and Catawba Technical Specification 3.7.9.3.

Operating Experience –

McGuire Operating Experience
Standby Nuclear Service Water Pond Dam Inspections have been performed for the Standby Nuclear Service Water (SNSW) Pond Dam and appurtenances since 1980. Previous inspections have revealed the dam to be in good condition. No conditions were identified that represent immediate danger to the safety and permanence of the SNSW Pond Dam and appurtenances. The inspection also includes a review of the instrumentation (piezometers, seepage monitor) for any abnormal conditions or trends. Available data show no significant changes in the trends for both the piezometers and observation wells. Each inspection performed a review of the previous inspection recommendations and documents the current status such as repair implemented or continual monitoring. The majority of the inspection recommendations were to spray the riprap on the upstream face and downstream toe of the dam to kill any vegetation, repair ruts, and re-seed. Structurally, cracks found in the vicinity of the concrete drainage ditch have been cleaned out and sealed with appropriate sealer.

In addition to the *Standby Nuclear Service Water Dam Inspection*, previous dam safety audits conducted by the NRC in 1994 and 1998 have concluded that there were no conditions observed that would indicate an immediate or adverse threat to the safety and permanence of the Standby Nuclear Service Water Pond Dam [References B - 49 and B - 50]. An independent consultant also performs inspections every five years per NCUC Docket No. E-100, Subpart 23. These inspections have also revealed similar findings.

Catawba Operating Experience

The Standby Nuclear Service Water Pond Dam is inspected annually in compliance with Technical Specification Surveillance Requirement 3.7.9.3. Based upon review of the results of previous inspections, no conditions were observed which have adverse effects on the intended function of the Standby Nuclear Service Water Pond Dam. Most common recurring recommendations are to clear vegetation from the concrete drainage ditches, pack soil and gravel along the sides of the concrete drainage ditch, and monitor any signs of erosion along the sides of the concrete drainage ditch. Most corrective items are routine and are addressed by regularly scheduled maintenance activities.

In addition to the credited *Standby Nuclear Service Water Pond Dam Inspection*, dam safety audits conducted by the NRC in 1997 and 1999 have concluded that there were no conditions that would indicate an immediate or adverse threat to the safety and permanence of the Standby Nuclear Service Water Pond Dam [References 51 and 52].

Conclusion

The *Standby Nuclear Service Water Pond Dam Inspection* has been demonstrated to be capable of detecting and managing loss of material and cracking of the dams. Based on the above review, the continued implementation of the *Standby Nuclear Service Water Pond Dam Inspection* provides reasonable assurance that loss of material and cracking will be managed such that the intended functions of the dams will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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B.3.31 STEAM GENERATOR SURVEILLANCE PROGRAM

Note: The STEAM GENERATOR SURVEILLANCE PROGRAM is generically applicable to both McGuire and Catawba Nuclear Stations, except as otherwise noted.

The purpose of the *Steam Generator Surveillance Program* is to provide comprehensive examinations of the steam generator tubes to ensure that degradation is identified and corrective actions are taken prior to exceeding allowable limits. The *Steam Generator Surveillance Program* is a condition monitoring program that is credited with managing loss of material and cracking of Alloy 600 and 690 steam generator tubes.

Scope – The scope of the *Steam Generator Surveillance Program* includes all steam generator tubes (including plugs and sleeves) in each steam generator and internal support structures.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The *Steam Generator Surveillance Program* monitors steam generator tube wall degradation and support plate locations.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending**, the *Steam Generator Surveillance Program* will detect loss of material and cracking prior to loss of component intended functions.

Monitoring & Trending – The minimum frequency of inspection for the steam generator tubes is specified in Technical Specification 5.5.9.3, Inspection Frequencies.

The minimum sample size of steam generator tubes to be inspected is specified in Technical Specification 5.5.9.2, Steam Generator Tube Sample Selection and Inspection.

In addition to the Technical Specification requirements, inspection of the steam generators follow the recommendations of the NEI 97-06 [Reference B - 53] and EPRI PWR Steam Generator Examination Guidelines [Reference B - 54].

Inspection of tubes and rolled plugs are done by eddy current examination. Tube plugs that cannot be examined by eddy current are visually inspected.

Acceptance Criteria – The acceptance criteria for the *Steam Generator Surveillance Program* is provided in Technical Specification 5.5.9.4, Acceptance Criteria. In addition, data are evaluated to determine that all structural and leakage criteria were met during the past operating cycle and a projection is made to determine that all tubes left in service will continue to meet operability requirements until the next examination.

Corrective Action & Confirmation Process – Corrective actions for the *Steam Generator Surveillance Program* are specified in Table 5.5-2, Steam Generator Tube Inspection of the Technical Specifications. Specific corrective actions and confirmation are implemented in accordance with the corrective action program.

Administrative Controls – The *Steam Generator Surveillance Program* is implemented using approved engineering programs and procedures and meets the minimum requirements of Technical Specification 5.5.9 for both McGuire and Catawba Nuclear Stations.

Operating Experience –

McGuire Operating Experience
The McGuire Steam Generators were replaced in May, 1997 for Unit 1 and December, 1998 for Unit 2. Since the replacement of the steam generators the only mechanism of tube degradation that has been identified in the McGuire Units 1 and 2 steam generators is wear at the secondary side U-bend fan bar and lattice grid supports.

Catawba Operating Experience
The Catawba Unit 1 steam generators were replaced in October, 1996. Wear has been identified at the secondary side U-bend fan bar supports.

The Catawba Unit 2 steam generators have not been replaced. The degradation mechanism that has been identified in the Unit 2 steam generators is wear. Wear has occurred at the edge of anti vibration bars and in the preheater section. Wear at the secondary side structures is a slow process and is readily detectable by eddy current testing before it is severe enough to detect tube structural integrity. Current wear rate is <5% per cycle for anti vibration bars and pre-heater tubes based on a review of eddy current data.

Conclusion

The *Steam Generator Surveillance Program* has been demonstrated to be capable of ensuring that steam generator degradation is identified and corrective actions are taken prior to exceeding allowable limits. The *Steam Generator Surveillance Program* described above is equivalent to the *Steam Generator Tube Surveillance Program* described and evaluated in NUREG-1723, Section 3.4.3.3 [Reference B - 5]. Based on the above review, the continued implementation of the *Steam Generator Surveillance Program* provides reasonable assurance that the aging effects will be managed and that the steam generator will continue to perform its intended function for the period of extended operation (i.e., 20-years from the end of the initial operation license).

B.3.32 SUMP PUMP SYSTEMS INSPECTION

Note: The SUMP PUMP SYSTEMS INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

The purpose of the *Sump Pump Systems Inspection* is to characterize any loss of material of the internal and external surfaces of a limited set of mechanical components exposed to sump environments. Sump environments may contain leakage from a variety of systems but are considered to be raw water environments with alternate wetting and drying as sump levels change. Uncertainty exists as to whether long term exposure to these sump environments could cause loss of material of system components such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect components constructed of various materials to detect the presence and extent of any loss of material from exposure to raw water, including alternate wetting and drying. The *Sump Pump Systems Inspection* is a one-time inspection.

Scope – The scope of the *Sump Pump Systems Inspection* is a limited set of mechanical components constructed of carbon steel, cast iron, and stainless steel exposed to sump environments in the following McGuire and Catawba systems:

- Diesel Generator Room Sump Pump System
- Conventional Waste Water Treatment System (MNS Only)
- Groundwater Drainage System
- Turbine Building Sump Pump System (CNS Only)

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameter inspected by the *Sump Pump Systems Inspection* is wall thickness as a measure of loss of material.

Detection of Aging Effects – The *Sump Pump Systems Inspection* is a one-time inspection that will detect the presence and extent of loss of material due to crevice, general, pitting, and microbiologically influenced corrosion.

Monitoring & Trending – The *Sump Pump Systems Inspection* will inspect sump components at each site located within the Diesel Generator Room Sump Pump System using a volumetric examination technique. The Diesel Generator Room Sump Pump System was selected for inspection because the system contains a representation of all of the materials present within the other sump environments. The sump environment in the Diesel Generator Room Sump Pump System is a potential combination of leakage of raw water, fuel oil, and treated water. Inspection of the Diesel Generator Room Sump Pump System will provide a representative review of the condition of mechanical component materials subject to a sump environment.

Inspection locations will be at piping low points, pump casings, and valve bodies where materials are continuously wetted by the raw water environment or subject to alternate wetting and drying. The results of this inspection will be applied to the mechanical components in the Conventional Waste Water Treatment (MNS Only), Groundwater Drainage, and Turbine Building Sump Pump Systems (CNS Only).

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

No actions are taken as part of this activity to trend inspection or test results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

The Groundwater Drainage System contains raw water that is considered to be relatively pure and not subject to mixing with treated water or contaminants from other plant systems. This environment is considered to be less severe than the other sump pump environments. Additionally, the system contains a limited selection of materials within the system boundaries at each station. Therefore, the results of the *Sump Pump Systems Inspection* are encompassing and will be applied to the Groundwater Drainage System components subject to a raw water environment.

The portion of the Catawba Turbine Building Sump Pump System within the scope of license renewal is carbon steel piping connecting the Liquid Waste System to the sump. This system was not selected for inspection because it is only applicable to one material and only at the Catawba station. Therefore, the results of the *Sump Pump Systems Inspection* are encompassing and will be applied to the Turbine Building Sump Pump System components subject to a raw water environment.

Acceptance Criteria – The acceptance criterion for the *Sump Pump Systems Inspection* is no unacceptable loss of material that could result in the loss of the component intended function(s), as determined by engineering evaluation.

Corrective Action & Confirmation Process – If the engineering evaluation determines that continuation of the aging effect will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The *Sump Pump Systems Inspection* will be implemented in accordance with controlled plant procedures.

Operating Experience – The *Sump Pump Systems Inspection* is a new one-time inspection for which there is no operating experience.

Conclusion

Based on the above review, the implementation of the *Sump Pump Systems Inspection* will adequately verify that no need exists to manage the aging effects on the components or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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B.3.33 TECHNICAL SPECIFICATION SR 3.6.16.3 VISUAL INSPECTION

Note: The TECHNICAL SPECIFICATION SR 3.6.16.3 VISUAL INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

Change in material property due to leaching has been identified as an aging effect requiring programmatic management for the walls and dome of the concrete Reactor Building for the extended period of operation. Technical Specification SR 3.6.16.3 requires that a visual inspection be performed on the exposed interior and exterior surfaces of the Reactor Building three times every ten years. The purpose of such visual inspections is to uncover evidence of deterioration which could affect the Reactor Building structural integrity. The *Technical Specification SR 3.6.16.3 Visual Inspection* is a condition monitoring program.

Scope – The scope of the *Technical Specification SR 3.6.16.3 Visual Inspection* includes accessible surface areas of the walls and dome of the concrete Reactor Building.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The parameter monitored by the *Technical Specification SR 3.6.16.3 Visual Inspection* is change in material property due to leaching.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending**, the *Technical Specification SR 3.6.16.3 Visual Inspection* will detect change in material properties due to leaching prior to loss of structure or component intended function.

Monitoring & Trending – Loss of material due to leaching is detected through visual examination. From the Technical Specification bases, this surveillance requirement provides advance indication of deterioration of the concrete structural integrity of the Reactor Building. The frequency of the inspection is three times every ten years. *Technical Specification SR 3.6.16.3 Visual Inspection* does not include a requirement to monitor or trend degradation. Unacceptable conditions are noted for correction or further evaluation.

Acceptance Criteria – Acceptance criteria are based on visual indication of structural damage or degradation.

For concrete, the acceptance criterion is no unacceptable indication of change in material property due to leaching.

Corrective Action & Confirmation Process – Structures and components which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repaired as required. Structures and components which are deemed unacceptable are documented under the corrective action program. Specific corrective actions are implemented in accordance with the corrective action program. Confirmatory actions, as needed, are implemented as part of the corrective action program.

Administrative Controls – McGuire and Catawba Technical Specifications SR 3.6.16.3 govern these inspections. The *Technical Specification SR 3.6.16.3 Visual Inspection* is implemented by plant procedures as required by Technical Specification 5.4.

Operating Experience – *Technical Specification SR 3.6.16.3 Visual Inspections* have been performed at the specified frequencies since initial operation. The inspection results are documented in station procedures. These inspections have revealed only minor degradation of concrete at McGuire and Catawba. Observations include minor hairline surface cracking and minor leaching. Leaching has been observed on the interior of the reactor building domes at McGuire near the dome-to-shell interface, and maintenance has been planned for the dome exterior to minimize water intrusion. Adverse conditions are reinspected during subsequent inspections. The observed aging effects are relatively minor and have no impact on the ability of the concrete Reactor Building to perform its intended functions.

Conclusion

The *Technical Specification SR 3.6.16.3 Visual Inspection* has been demonstrated to be effective in managing aging of the concrete Reactor Building during the period of extended operation. Based on the above review, the continued implementation of the *Technical Specification SR 3.6.16.3 Visual Inspection* provides reasonable assurance that change in material property due to leaching of Reactor Building concrete will be managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

B.3.34 TREATED WATER SYSTEMS STAINLESS STEEL INSPECTION

Note: The TREATED WATER SYSTEMS STAINLESS STEEL INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

The purpose of the *Treated Water Systems Stainless Steel Inspection* is to characterize any loss of material or cracking of stainless steel components resulting from exposure to unmonitored treated water environments. An unmonitored treated water environment is one that may contain conditions that can concentrate existing levels of contaminants or that may simply start with a higher level of contaminants than those systems routinely monitored by the Chemistry Control Program. Examples of contaminants are halogens, sulfates, and dissolved oxygen. Uncertainty exists as to whether exposure of stainless steel components located in an unmonitored treated water environment could lead to loss of material or cracking such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect stainless steel components to detect the presence and extent of any loss of material or cracking. The *Treated Water Systems Stainless Steel Inspection* is a one-time inspection.

Scope – The scope of *Treated Water Systems Stainless Steel Inspection* is stainless steel components exposed to unmonitored treated water environments in the following McGuire and Catawba systems:

- Containment Valve Injection Water (CNS only)
- Drinking Water (CNS only)
- Nuclear Solid Waste Disposal (MNS only)
- Solid Radwaste (CNS only)

Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – The parameters inspected by the *Treated Water Systems Stainless Steel Inspection* are pipe wall thickness, as an indicator of loss of material, and evidence of cracking.

Detection of Aging Effects – The *Treated Water Systems Stainless Steel Inspection* is a one-time inspection that will detect the presence and extent of any loss of material or cracking of stainless steel components exposed to unmonitored treated water environments.

Monitoring & Trending – The *Treated Water Systems Stainless Steel Inspection* at McGuire will inspect stainless steel components, welds, and heat affected zones, as applicable, in the McGuire Nuclear Solid Waste Disposal System. The McGuire Nuclear Solid Waste Disposal System components within the scope of license renewal is a mixture of unmonitored treated water and spent resins sluiced from demineralizers in various systems. The environment is expected to contain contaminants in excess of the limits below which a concern would not exist for cracking and loss of material in stainless steel. A concentration of any contaminants present would occur in areas of low flow or stagnant conditions. As a result, inspections will be performed in stagnant and low flow lines around the spent resin storage tanks using volumetric techniques. In addition to the volumetric examination, a visual examination of the interior of a valve will be conducted to determine the presence of pitting corrosion.

The *Treated Water Systems Stainless Steel Inspection* at Catawba will inspect stainless steel components, welds, and heat affected zones, as applicable, in the Drinking Water System. The Drinking Water System receives water from the local municipality that has contaminants in excess of limits below which a concern would not exist for cracking and loss of material in stainless steel. Because of the higher starting level of contaminants, the environment in the Drinking Water System is more likely to lead to cracking or loss of material if it is occurring and bounds the environments of the Containment Valve Injection Water and Solid Radwaste Systems. In addition to the volumetric examination, a visual examination of the interior of a valve will be conducted to determine the presence of pitting corrosion. Therefore, the inspection results will serve as a leading indicator and can be applied to the Containment Valve Injection Water and Solid Radwaste Systems.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

No actions are taken as part of this activity to trend inspection results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

Acceptance Criteria – The acceptance criterion for the *Treated Water Systems Stainless Steel Inspection* is no unacceptable loss of material or cracking that could result in the loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. Specific corrective actions will be implemented in accordance with the corrective action program.

Administrative Controls – The *Treated Water Systems Stainless Steel Inspection* will be implemented in accordance with controlled plant procedures.

Operating Experience – The *Treated Water Systems Stainless Steel Inspection* is a one-time inspection activity for which there is no operating experience. However, a similar inspection activity was reviewed and deemed acceptable by the NRC Staff for Oconee, as stated in the conclusions below.

Conclusion

The *Treated Water Systems Stainless Steel Inspection* described above is similar to the corresponding activity described and evaluated in NUREG-1723, Section 3.2.11 [Reference B - 5]. Based on the above review, implementation of the *Treated Water Systems Stainless Steel Inspection* will adequately verify that no need exists to manage the aging effects on the component or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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B.3.35 UNDERWATER INSPECTION OF NUCLEAR SERVICE WATER STRUCTURES

Note: The UNDERWATER INSPECTION OF NUCLEAR SERVICE WATER STRUCTURES is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

Loss of material due to corrosion of steel components in fluid environments and loss of material and cracking of concrete components have been identified as aging effects requiring management for Nuclear Service Water Structures and the Low Pressure Service Water Intake Structure at Catawba. The structures at McGuire which are exposed to pond water are the Standby Nuclear Service Water Intake and Discharge structures. The structures at Catawba which are exposed to lake or pond water are the Standby Nuclear Service Water Intake Structure, the Standby Nuclear Service Water Discharge Structures, the Standby Nuclear Service Water Pond Outlet, the Nuclear Service Water Pump Structure, the Nuclear Service Water Intake Structure, and the Low Pressure Service Water Intake Structure. The *Underwater Inspection of Nuclear Service Water Structures* is credited with managing loss of material of steel and loss of material and cracking for concrete for the period of extended operation. The *Underwater Inspection of Nuclear Service Water Structures* is a condition monitoring program.

Scope – The scope of the *Underwater Inspection of Nuclear Service Water Structures* includes the following structures:

<u>McGuire Nuclear Station</u>

- Standby Nuclear Service Water Discharge Structures
- Standby Nuclear Service Water Intake Structure

<u>Catawba Nuclear Station</u>

- Low Pressure Service Water Intake Structure
- Nuclear Service Water Intake Structure
- Nuclear Service Water Pump Structure
- Standby Nuclear Service Water Discharge Structures
- Standby Nuclear Service Water Intake Structure
- Standby Nuclear Service Water Pond Outlet

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The *Underwater Inspection of Nuclear Service Water Structures* requires examination of the structure for the following parameters: loss of material of steel components and loss of material and cracking of concrete components.

Detection of Aging Effects – In accordance with information provided in **Monitoring & Trending**, the *Underwater Inspection of Nuclear Service Water Structures* will detect loss of material of steel components and loss of material and cracking of concrete components prior to loss of structure or component intended functions.

Monitoring & Trending – The *Underwater Inspection of Nuclear Service Water Structures* detects aging effects through visual examination. The inspection is performed every five years at McGuire and every Unit 1 refueling outage for Catawba Nuclear Service Water and Standby Nuclear Service Water Intake structures and five years for other Catawba structures. No actions are taken as part of this program to trend inspection or test results.

The underwater inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. The diver report includes the results of the current inspection or findings. The accountable engineer is responsible for reviewing the findings and determining whether or not the results are acceptable.

Acceptance Criteria – The acceptance criteria are no unacceptable visual indication of (1) loss of material for steel components and (2) loss of material and cracking for concrete components, as determined by the accountable engineer.

Corrective Action & Confirmation Process – Structures and components which do not meet the acceptance criteria are evaluated by the accountable engineer for continued service and repair, as required. Structures and components which are deemed unacceptable are documented under the corrective action program. Specific corrective actions and confirmatory actions, as needed, are implemented in accordance with the corrective action program. All prior inspection reports are reviewed to ensure implementation of recommended corrective actions.

Administrative Controls – The *Underwater Inspection of Nuclear Service Water Structures* is implemented by plant work management system using model work orders.

Operating Experience –

McGuire Operating Experience
The *Underwater Inspection of Nuclear Service Water Structures* has been performed for the Standby Nuclear Service Water (SNSW) Intake and Discharge structures since 1989.

In 1992 the SNSW Intake trash racks were replaced. An inspection of the SNSW Intake Structure at that time had revealed deterioration of the trash racks and fasteners. The old trash racks and fasteners were made of galvanized steel. The new trash racks are made of stainless steel.

Review of previous *Underwater Inspection of Nuclear Service Water Structures* reports indicates the SNSW Intake and Discharge structures are in good working condition. Observations included good concrete condition, acceptable silt level, minor biofouling, and intact trash rack and fasteners.

Catawba Operating Experience

Previous *Underwater Inspection of Nuclear Service Water Structures* have revealed only minor degradation. No deterioration that could cause loss of intended function has been identified from the previous inspections.

<u>Conclusion</u>

The *Underwater Inspection of Nuclear Service Water Structures* has been demonstrated to be capable of detecting and managing loss of material for steel components and loss of material and cracking for concrete components. The *Underwater Inspection of Nuclear Service Water Structures* described above is equivalent to the Duke Power 5-Year Underwater Inspection of Hydroelectric Dams and Appurtenances described and evaluated in NUREG-1723, Section 3.2.4 [Reference B - 5]. Based on the above review, the continued implementation of the *Underwater Inspection of Nuclear Service Water Structures* provides reasonable assurance that the aging effects will be managed such that the intended functions of the structures and components will continue to be maintained consistent with the current licensing basis for the period of extended operation (i.e., 20-years from the end of the initial operating license).

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B.3.36 WASTE GAS SYSTEM INSPECTION

Note: The WASTE GAS SYSTEM INSPECTION is generically applicable to both McGuire Nuclear Station and Catawba Nuclear Station, except as otherwise noted.

The purpose of the *Waste Gas System Inspection* is to characterize loss of material and cracking in Waste Gas System components resulting from exposure to unmonitored treated water and gas environments. Unmonitored treated water is condensation of the water vapor contained in the waste gas stream and effluent from the recombiners and separators. The gas environment is a combination of nitrogen, hydrogen, oxygen, and fission product gases. Uncertainty exists as to whether exposure to these environments could cause loss of material or cracking of the Waste Gas System components such that they may lose their pressure boundary function in the period of extended operation. This activity will inspect Waste Gas System components to detect the presence and extent of loss of material and cracking from exposure to unmonitored treated water and gas environments. The *Waste Gas System Inspection* is a one-time inspection.

Scope – The scope of the *Waste Gas System Inspection* is carbon steel, stainless steel, and brass materials that are exposed to unmonitored treated water environments and carbon steel materials that are exposed to gas environments within the license renewal boundaries of the McGuire and Catawba Waste Gas Systems.

Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

Parameters Monitored or Inspected – The parameters monitored or inspected by the *Waste Gas System Inspection* are wall thickness, as a measure of loss of material, and evidence of cracking.

Detection of Aging Effects – The *Waste Gas System Inspection* is a one-time inspection that will detect the presence and extent of any loss of material due to general, crevice, or pitting corrosion or cracking due to stress corrosion in brass, carbon steel, and stainless steel materials subject to an unmonitored treated water environment. The *Waste Gas System Inspection* will also detect the presence and extent of any loss of material due to general corrosion in carbon steel materials subject to a gas environment.

Monitoring & Trending – The *Waste Gas System Inspection* will use a volumetric technique to inspect four sets of material/environment combinations. As an alternative, visual examination will be used should access to internal surfaces become available. The Waste Gas System is primarily a gas environment with unmonitored treated water environments from condensation of entrained water vapor and effluent from the recombiners and separators.

Specific component/environment inspection combinations will include brass, carbon steel, and stainless steel components exposed to an unmonitored treated water environment. Also, carbon steel components exposed to a gas environment will be inspected. Selection of the specific areas for inspection for the above material/environment combinations will be the responsibility of the system engineer.

(1) For the brass seal water control valves on the waste gas compressors at Catawba exposed to unmonitored treated water, an inspection will be performed on one of the two seal water control valves. If no parameters are known that would distinguish the susceptible locations, one of the two available at will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the other brass seal water control valve.

(2) For carbon steel components exposed to unmonitored treated water environments at each site, inspections will be performed on the lower portions of decay tanks and associated drain lines where condensate is likely to accumulate. One of eight possible locations at each site will be examined. If no parameters are known that would distinguish the susceptible locations at each site, one of the eight available at each site will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the remainder of the Waste Gas System carbon steel components within the scope of license renewal exposed to unmonitored treated water environment.

(3) For stainless steel components exposed to unmonitored treated water environments at each site, inspections will be performed on the seal water path of the waste gas compressor. One of two possible locations at each site will be examined. If no parameters are known that would distinguish the susceptible locations at each site, one of the two available at each site will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the remainder of the Waste Gas System stainless steel components within the scope of license renewal exposed to unmonitored treated water environment.

(4) For the carbon steel components exposed to a gas environment at each site, an inspection will be performed on components within the scope of license renewal located between the volume control tanks and the waste gas compressor phase separators. If no parameters are known that would distinguish the most susceptible locations at each site, one location at each site will be examined based on accessibility and radiological concerns. The results of this inspection will be applied to the remainder of the Waste Gas System carbon steel components within the scope of license renewal exposed to gas environments.

For McGuire, this new inspection will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, this new inspection will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

No actions are taken as part of this activity to trend inspection or test results.

Should industry data or other evaluations indicate that the above inspections can be modified or eliminated, Duke will provide plant-specific justification to demonstrate the basis for the modification or elimination.

The Waste Gas System is primarily a gas environment composed of nitrogen, hydrogen, oxygen, and fission product gases. The section of the Waste Gas System between the volume control tanks and the waste gas compressors phase separators will contain a warm, moist gas that could result in the cooler internal surfaces of the carbon steel components being wet due to condensation. As a result, corrosion of the carbon steel surfaces is more likely due to the presence of moisture and would serve as a leading indicator for the remainder of the carbon steel components within the scope of license renewal exposed to the gas environment in the Waste Gas System. Therefore, the results of the inspection can be applied to the remainder of the carbon steel components exposed to gas environments.

Acceptance Criteria – The acceptance criteria for the *Waste Gas System Inspection* is no unacceptable loss of material or cracking that could result in a loss of the component intended function(s) as determined by engineering evaluation.

Corrective Action & Confirmation Process – If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under any current licensing basis design conditions for the period of extended operation, no further action is required. If the engineering evaluation determines that additional information is required to more fully characterize any or all of the aging effects, then additional inspections will be completed or other actions taken in order to obtain the additional information. If further engineering evaluation determines that continuation of the applicable aging effects could cause a loss of component intended function(s) under current licensing basis design conditions for the period of extended operation, then programmatic oversight is required to be defined by engineering. Specific corrective actions will be implemented in accordance with the Corrective Action Program.

Administrative Controls – The *Waste Gas System Inspection* will be implemented in accordance with controlled plant procedures.

Operating Experience – The *Waste Gas System Inspection* is a one-time inspection activity for which there is no operating experience.

Conclusion

Based on the above review, implementation of the *Waste Gas System Inspection* will adequately verify that no need exists to manage the aging effects on the components or will otherwise take appropriate corrective actions so that the components will continue to perform their intended function(s) for the period of extended operation (i.e., 20-years from the end of the initial operating license).

B.4 References for Appendix B

B - 1. *Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3*, July 1998.

B - 2. *Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule,* NEI 95-10, Revision 2, Nuclear Energy Institute, August 2000.

B - 3. *Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants*, Draft, NRC, August 2000.

B - 4. NEI (Alex Marion) letter dated October 13, 2000 to NRC (Secretary of the Commission), *Comments on the Draft Guidance Documents for License Renewal (65 FR 53047, August 31, 2000).*

B - 5. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station*, Units 1, 2, and 3, March 2000, U. S. Nuclear Regulatory Commission, Docket Nos. 50-269, 50-270, and 50-287.

B - 6. IEEE 450-1980, *IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead storage Batteries for Generating Stations and Substations*, Institute of Electrical and Electronics Engineers, 345 East 47th Street, New York, NY.

B - 7. M. S. Tuckman (Duke) letter to Document Control Desk (NRC) dated October 22, 1996, Response to Generic Letter 96-04, *Boraflex Degradation in Spent Fuel Storage Racks*, McGuire Nuclear Station, Dockets No., 50-369 and 50-370.

B - 8. NRC Generic Letter 96-04, Boraflex Degradation in Spent Fuel Pool Storage Racks, June 26, 1996.

B - 9. M. S. Tuckman (Duke) letter to Document Control Desk (NRC) dated December 22, 1997, Response to Generic Letter 96-04, *Boraflex Degradation in Spent Fuel Storage Racks*, McGuire Nuclear Station, Dockets Nos., 50-369 and 50-370.

B - 10. 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.

B - 11. WCAP-12866, *Bottom Mounted Instrumentation Flux Thimble Wear*, January 1991.

B - 12. NRC Bulletin 88-09: Thimble Tube Thinning In Westinghouse Reactors, July 6, 1988.

B - 13. *PWR Primary Water Chemistry Guidelines: Revision 4*, EPRI TR-105714-V1R4, March 1999.

B - 14. *PWR Secondary Water Chemistry Guidelines – Revision 5*, EPRI TR-102134-R5, 2000.

B - 15. *Closed Cooling Water Chemistry Guideline*, EPRI TR-107396, October 1997.

B - 16. Catawba Nuclear Station Units 1 and 2, Technical Specifications.

B - 17. McGuire Nuclear Station Units 1 and 2, Technical Specifications.

B - 18. Catawba Nuclear Station Units 1 and 2, Updated Final Safety Analysis Report, *Chapter 16, Selected Licensee Commitments*, Sections 16.5-3, 16.7-9, and 16.8-5 dated 1/16/99, 9/20/99, and 6/10/99, respectively.

B - 19. McGuire Nuclear Station Units 1 and 2, Updated Final Safety Analysis Report, *Chapter 16, Selected Licensee Commitments*, Sections 16.5-7, 16.8-3, and 16.9-7 revisions 0, 2, and 1, respectively.

B - 20. 10 CFR Part 50, §50.55a, *Codes and Standards.*

B - 21. H. B. Barron (Duke) letter to Document Control Desk (NRC) dated January 11, 2001, *McGuire Nuclear Station Unit 2 Inservice Inspection Summary Report for Class MC Component Examinations Conducted During Refueling Outage EOC13,* McGuire Nuclear Station, Dockets Nos., 50-369 and 50-370.

B - 22. G. R. Peterson (Duke) letter to Document Control Desk (NRC) dated May 1, 2000, *Catawba Nuclear Station Unit 2 Inservice Inspection Summary*

Report for Class MC Component Examinations Conducted During Refueling Outage EOC10, Catawba Nuclear Station, Docket Nos. 50-413 and 50-414.

B - 23. NRC Regulatory Guide 1.163, *Performance-Based Containment Leak-Test Program*, September 1995.

B - 24. W. T. Russell (NRC) letter dated November 19,1993 to William Rasin, (NUMARC), *Safety Evaluation for Potential Reactor Vessel Head Adapter Tube Cracking*.

B - 25. Generic Letter 97-01, *Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations*, U. S. Nuclear Regulatory Commission, April 1, 1997.

B - 26. M.S. Tuckman (Duke) letter dated July 30, 1997 To Document Control *Desk NRC*, *Response to Generic Letter 97-01: Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, Catawba Nuclear Station, Docket Nos. 50-413 and 50-414, McGuire Nuclear Station, Docket Nos. 50-369 and 50-370, Oconee Nuclear Station, Docket Nos. 50-269, 50-270 and 50-287.

B - 27. ANSI B30.2.0, “Overhead and Gantry Cranes,” American National Standard, Section 2-2, *Safety Standards for Cableways, Cranes, Derricks, Hoists, Hooks, Jacks and Slings*, The American Society of Mechanical Engineers, New York.

B - 28. ANSI B30.16, *Overhead Hoists (Underhung)*, The American Society of Mechanical Engineers, New York.

B - 29. 29 CFR Chapter XVII, 1910.179, *Occupational Safety and Health Administration, Overhead and Gantry Cranes*.

B - 30. G. R. Peterson (Duke) letter to Document Control Desk (NRC) dated June 2, 1998, *Special Report: Fire Protection Program Inoperable Fire Barrier Penetrations*, Catawba Nuclear Station, Docket Nos. 50-413 and 50-414.

B - 31. M. S. Tuckman (Duke) letter to L. A. Reyes (NRC) dated August 4, 1998, *Fire Barrier Penetration Seals*, Oconee Nuclear Station, Docket Nos. 50-269, 50-270, and 50-287; McGuire Nuclear Station, Dockets Nos., 50-369 and 50-370; Catawba Nuclear Station, Docket Nos. 50-413 and 50-414.

B - 32. G. R. Peterson (Duke) letter to Document Control Desk (NRC) dated March 1, 2000, *Special Report Concerning Violation of License Conditions Regarding Fire Protection*, Catawba Nuclear Station, Docket Nos. 50-413 and 50-414.

B - 33. NRC Information Notice IN 87-49, *Deficiencies in Outside Containment Flooding Protection*, October 9, 1987.

B - 34. EPRI NSAC-202L-R1, *Recommendations for an Effective Flow Accelerated Corrosion Program*, Revision 2, April 1999.

B - 35. NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*, March 17, 1988.

B - 36. Nuclear System Directive 104, *Housekeeping, Materiel Condition and Foreign Material Exclusion*, Revision 19.

B - 37. Nuclear System Directive 413, *Fluid Leak Management Program*, Revision 0.

B - 38. *McGuire Nuclear Station Updated Final Safety Analysis Report*, as revised.

B - 39. *Catawba Nuclear Station Updated Final Safety Analysis Report*, as revised.

B-40. IE Bulletin 82-02 dated June 2, 1982, Degradation of Threaded Fasteners in Reactor Coolant Pressure Boundary of PWR Plants.

B - 41. 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.

B - 42. WCAP 14572 Revision 1, NP-A, *Westinghouse Owners Group Application of Risk-Based Methods to Piping Inservice Inspection Topical Report*.

B - 43. *Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One*, Unit 1, April 2001.

B - 44. *Guideline for the Management of Adverse Localized Equipment Environments*, EPRI, Palo Alto, CA: 1999. EPRI TR-109619.

B - 45. Regulatory Guide 1.45, *Reactor Coolant Pressure Boundary Leakage Detection Systems*, May, 1973.

B - 46. WCAP-14040, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*, June 1994.

B - 47. ASME Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components, Subsection ND Class 3 Components, 1971 edition.

B - 48. NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment.*

B - 49. Rinaldi, F. (NRC) to Barron, H. B. (Duke) dated March 24, 1999, *Results of Dam Safety Audit related to the Category I Standby Service Water Pond Dam at the McGuire Nuclear Station,* McGuire Nuclear Station, Dockets Nos., 50-369 and 50-370.

B - 50. Crlenjak, R. V. (NRC) to McMeekin, T. C. (Duke) dated January 6, 1995, NRC Inspection Report Nos. 50-369/94-27 and 50-370/94-27, SNSW Pond Dam Inspection on November 14-16, 1994.

B - 51. Peter S. Tam (NRC) letter dated January 13, 1998 to G. R. Peterson (Duke), Docket Nos. 50-413 and 50-414, *Results of Dam Safety Inspection Related to the Category I Standby Nuclear Service Water Pond Dam*.

B - 52. Chandu P. Patel (NRC) letter dated October 5, 2000 to G. R. Peterson (Duke), Docket Nos. 50-413 and 50-414, *Results of Dam Safety Audit Related to the Category I Service Water Pond Dam*.

B - 53. NEI 97-06, *Steam Generator Program Guidelines.*

B - 54. EPRI TR-107569-V1R5, *PWR Steam Generator Examination Guidelines: Revision 5*, September 1997.

Appendix C

Commodity Groups (Optional)

Appendix C is not being used in the Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2.

Appendix D

Technical Specification Changes

No technical specification changes have been identified as being necessary to support issuance of the renewed operating licenses for McGuire Nuclear Station Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2.

Appendix E

Environmental Information

The Environmental Report for McGuire Nuclear Station is contained in a separate document entitled "Applicant's Environmental Report Operating License Renewal Stage, McGuire Nuclear Station."

The Environmental Report for Catawba Nuclear Station is contained in a separate document entitled "Applicant's Environmental Report Operating License Renewal Stage, Catawba Nuclear Station."