V.C. Summer Nuclear Station

License Renewal Application

Docket Number 50/395
License Number NPF-12

SCE&G
A SCANA COMPANY

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INTRODUCTION

Pursuant to Part 54 of Title 10 of the Code of Federal Regulations (10 CFR 54), this application seeks renewal for an additional 20 year term of the facility operating license for Virgil C. Summer Nuclear Station (VCSNS) (NPF-12). The facility operating license (NPF-12) currently expires at midnight August 6, 2022. The application includes renewal of the source, special nuclear, and by-product materials licenses that are combined in the facility operating license.

The application is organized in accordance with the U.S. Nuclear Regulatory Commission Regulatory Guide 1.188, “Standard Format And Content For Applications To Renew Nuclear Power Plant Operating Licenses,” April 2001, and is consistent with the guidance provided by NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR 54 - License Renewal.”

The License Renewal Application is intended to provide sufficient information for the NRC to complete its technical and environmental reviews. Pursuant to 10 CFR Parts 54 and 51, respectively, the License Renewal Application is designed to allow the NRC to make the findings required by 10 CFR 54.29 in support of the issuance of a renewed facility operating license for VCSNS. The following is the general information required by 10 CFR 54.17 and 54.19.

1.0 ADMINISTRATIVE INFORMATION

1.1 NAMES OF APPLICANTS

South Carolina Electric & Gas Company (2/3 owner, and exclusive operator)
South Carolina Public Service Authority (1/3 owner)

1.2 ADDRESSES OF APPLICANTS

South Carolina Electric & Gas Company

US Postal Service Address:
SCE&G
Columbia, SC 29218

Private Courier Address:
SCE&G

1426 Main Street
Columbia, SC 29201

All communications pertaining this application should be sent to:

Stephen A. Byrne
Senior Vice President - Nuclear Operation
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

Ronald B. Clary
Manager, Plant Life Extension
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

In addition, it is requested that additional copies be sent to the company's General Counsel and Washington Counsel:

H. Thomas Arthur
Senior Vice President - General Counsel and Assistant Secretary
SCANA Corporation
1426 Main Street
Columbia, SC 29201

Kathryn M. Sutton
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

South Carolina Public Service Authority

Maxie C. Chaplin
Senior Vice President, Generation
South Carolina Public Service Authority
P. O. Box 2946101
Moncks Corner, SC 29461-2901

Robin White
Nuclear Coordinator
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

1.3 DESCRIPTION OF BUSINESS

South Carolina Electric & Gas Company:

South Carolina Electric & Gas Company (SCE&G), a wholly-owned subsidiary of SCANA Corporation, is primarily engaged in the generation, transmission and distribution of electricity. The service territory covers the southern two-thirds of the state of South Carolina. SCE&G supplies electric service to more than 500,000 residential, commercial, and industrial customers. To service this area, SCE&G operates 19 electric generating facilities with an installed capacity of over 4500 MW electric, including VCSNS.

SCE&G is 2/3 owner of VCSNS and is authorized to act as agent for the South Carolina Public Service Authority and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.[1]

South Carolina Public Service Authority:

South Carolina Public Service Authority is a state owned electric and water utility that serves approximately 130,000 retail customers in Berkeley, Georgetown and Horry counties. The utility generates the power distributed by 20 electric cooperatives to more than 450,000 customers located in 46 counties, supplies power to 34 large industries, and the municipalities of Bamberg and Georgetown.

South Carolina Public Service Authority is 1/3 owner of VCSNS.

1. SCE&G/South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Facility Operating License NPF-12

1.4 ORGANIZATION AND MANAGEMENT OF APPLICANTS

South Carolina Electric & Gas Company:

William B. Timmerman
President & Chief Executive Officer
SCANA Corporation
1426 Main Street
Columbia, SC 29201

Neville O. Lorick
President and Chief Operating Officer
South Carolina Electric & Gas Company
1426 Main Street
Columbia, SC 29201

Kevin B. Marsh
Chief Financial Officer
SCANA Corporation
1426 Main Street
Columbia, SC 29201

H. Thomas Arthur
Senior Vice President - General Counsel and Assistant Secretary
SCANA Corporation
1426 Main Street
Columbia, SC 29201

Duane C. Harris
Senior Vice President - Human Resources
SCANA Corporation
1426 Main Street
Columbia, SC 29201

Stephen A. Byrne
Senior Vice President - Nuclear Operation
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

J. Patrick Hudson
Corporate Compliance and Internal Auditing Officer
SCANA Corporation
1426 Main Street
Columbia, SC 29201

South Carolina Public Service Authority:

John H. Tiencken
President and Chief Executive Officer
South Carolina Public Service Authority
P. O. Box 2946101
Moncks Corner, SC 29461-2901

Bill McCall Jr.
Executive Vice President and Chief Operations Officer
South Carolina Public Service Authority
P. O. Box 2946101
Moncks Corner, SC 29461-2901

John S. West
Executive Vice President and Chief Legal Officer
South Carolina Public Service Authority
P. O. Box 2946101
Moncks Corner, SC 29461-2901

Lonnie N. Carter
Senior Vice President Corporate Planning & Bulk Power
South Carolina Public Service Authority
P. O. Box 2946101
Moncks Corner, SC 29461-2901

Elaine G. Peterson
Senior Vice President Administration & Finance
South Carolina Public Service Authority
P. O. Box 2946101
Moncks Corner, SC 29461-2901

1.5 CLASS AND PERIOD OF LICENSE SOUGHT

SCE&G requests renewal of the Class 104b facility operating license for VCSNS (facility operating license NPF-12) for a period of 20 years beyond the expiration of the current license term. License renewal would extend the facility operating license from midnight August 6, 2022 until midnight August 6, 2042. This application includes a request for renewal of those NRC source material, special nuclear material, and by-product material licenses that are currently subsumed or combined with the current facility operating license.

1.6 ALTERATION SCHEDULE

SCE&G does not propose to construct or alter any production or utilization facility in connection with this renewal application.

1.7 LISTING OF REGULATORY AGENCIES HAVING JURISDICTION & APPROPRIATE NEWS PUBLICATIONS

The Public Service Commission of South Carolina has jurisdiction over the rates and services provided by the company’s utility operations. It’s address is:

Public Service Commission of South Carolina
Koger Executive Center
Saluda Building
101 Executive Center Drive
Columbia, SC 29210

The news publication that serves the VCSNS surrounding area is The State newspaper. It’s address is:

The State Newspaper
P. O. Box 1333
Columbia, SC 29202

1.8 CONFORMING CHANGES TO THE 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 renewal license.” The current indemnity agreement (No. B-86) for VCSNS states in Article VII that the agreement shall terminate at the time of expiration of the license specified in Item 3 of the Attachment to the agreement, which is the last to expire. Item 3 of the Attachment to the indemnity agreement, as revised by Amendment No. 2 lists two license numbers (SNM-1834 and NPF-12). SCE&G requests that conforming changes be made to Article VII of the indemnity agreement, and Item 3 of the Attachment to that agreement, specifying the extension of agreement until the expiration date of the

renewed VCSNS facility operating license as sought in this application. In addition, should the license numbers be changed upon issuance of the renewal license, SCE&G requests that conforming changes be made to Item 3 of the Attachment, and any other sections of the indemnity agreement as appropriate.

1.9 RESTRICTED DATA AGREEMENT

This application does not contain any Restricted Data or National Security Information, and SCE&G does not expect that any activity under the renewed license for VCSNS will involve such information. However, if such information were to become involved, SCE&G agrees that it will appropriately safeguard such information and not permit any individual to have access to, or any facility to possess, such information until the individual or facility has been approved under the provisions of Parts 10 CFR 25 or 10 CFR 95, respectively.

SECTION 2 - SCOPING AND SCREENING REVIEW

2.0 STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW

Section 2.0 describes the process for the identification of structures and components subject to an aging management review for Virgil C. Summer Nuclear Station (VCSNS). For those systems, structures, and components within the scope of license renewal, NEI 95-10.21(a)(1) [**Reference 2.1-1**] requires a license renewal applicant to identify and list the structures and components subject to an aging management review. Furthermore, 10 CFR 54.21(a)(2) requires that the methods used to identify and list these structures and components be described and justified. The technical information in Section 2.0 serves to satisfy these requirements.

10 CFR 54.21 (a)(1)

Each application must contain the following information:
(a) An integrated plant assessment (IPA). The IPA must --
(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. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and
(ii) That are not subject to replacement based on a qualified life or specified time period.

10 CFR 54.21 (a)(2)

Each application must contain the following information:
(a) An integrated plant assessment (IPA). The IPA must --
(2) Describe and justify the methods used in paragraph (a)(1) of this section.

2.1 SCOPING AND SCREENING METHODOLOGY

Scoping is the application of the criteria contained in 10 CFR 54.4(a) [**Reference 2.1-1**] that determine the systems, structures, and components within the scope of license renewal. **Section 2.1.1** provides a description of the scoping methodology that has been implemented to address the criteria of 10 CFR 54.4(a) for VCSNS. This methodology is consistent with the guidance provided in NEI 95-10 [**Reference 2.1-2**].

Screening is the application of the criteria contained in 10 CFR 54.21(a)(1) that determine the structures and components subject to an aging management review. **Section 2.1.2** provides a description of the screening methodology that has been implemented to address the criteria of 10 CFR 54.21(a)(1) for VCSNS. This methodology is consistent with the guidance provided in NEI 95-10. Electrical equipment within mechanical systems or structures considered within the scope of license renewal are carried forward as electrical commodities and are screened as described in **Section 2.1.2.3**.

2.1.1 PLANT LEVEL SCOPING

Plant level scoping begins by defining the plant in terms of major systems and structures. These systems and structures are then evaluated against scoping criteria in 10 CFR 54.4.

10 CFR 54.4 Scope

(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-basis 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 which could result in potential offsite exposures comparable to those referred to in §50.34(a)(1), §50.67(b)(2), or §100.11 of this chapter, as applicable.

(2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.

(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).

(b) The intended functions that these systems, structures, and components must be shown to fulfill in §54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1) - (3) of this section.

The scoping process to identify systems and structures that satisfy the requirements of 10 CFR 54.4(a)(1), 10 CFR 54.4(a)(2), and 10 CFR 54.4(a)(3) is performed on systems and structures using documents which form the Current Licensing Basis (CLB) and other information sources. The CLB for the VCSNS has been defined in accordance with the definition provided in 10 CFR 54.3. The key information sources that form the CLB include the Final Safety Analysis Report (FSAR), Technical Specifications, and the docketed licensing correspondence. Other important information sources used for scoping are further described in **Subsection 2.1.1.1**. The aspects of the scoping process used to identify systems and structures and components that satisfy the requirements of 10 CFR 54.4(a)(1), 10 CFR

54.4(a)(2), and 10 CFR §54.4(a)(3) are described in **Subsections 2.1.1.2**, **2.1.1.3**, and **2.1.1.4** respectively.

2.1.1.1 Information Sources

In addition to the FSAR, Technical Specifications, and docketed licensing correspondence, three information sources – the design basis documents, the component databases, and flow diagrams – were relied upon to a great extent in performing scoping and screening for the VCSNS. A brief discussion of these sources is provided below.

2.1.1.1.1 Design Basis Documents

Design basis documents are tools to explain the requirements behind the design rather than describing the design itself. Design basis documents are intended to complement other upper tier documents, such as the FSAR and Technical Specifications, and are controlled and updated.

2.1.1.1.2 Component Database

Specific component information for Systems, Structures and Components (SSCs) at the VCSNS can be found in the controlled Component History And Maintenance Planning System (CHAMPS) database and the Equipment Qualification Database (EQDB).

CHAMPS is a controlled database that contains as-built information on a component level. The CHAMPS database consists of multiple data fields for each component, such as design-related information, safety and seismic classifications, safety classification bases, and component tag, type, and description.

The EQDB is a controlled database that consists of multiple data fields for each component/sub-component, such as component identification, vendor, vendor model number, Reg. Guide 1.97 category, mild or harsh environment category, and maintenance requirements.

2.1.1.1.3 Flow Diagrams (P&ID)

The flow diagrams were used to delineate the mechanical systems screening boundaries within which the component types (e.g. pipe, valves), that have license renewal intended functions, are identified.

flow diagrams also delineate the relationship of SSCs on a system basis and the interfaces between systems. Additionally, the flow diagrams reflect ASME Code boundaries and quality grouping classifications.

Regarding vendor supplied skid mounted packages, vendor drawings were used to assist in the scoping and screening reviews. The vendor drawings reflect the SSCs within a skid package and the interfaces with the system it supports as outlined in the flow diagrams discussed above.

2.1.1.1.4 Electrical Boundaries

The boundary for the license renewal evaluation (for non-EQ insulated cables, connectors, splices, penetration assemblies and terminal blocks) is considered to be the point of connection to an electrical end device, at either end of a circuit. Thus, the cable to an in-scope non-EQ instrument (and any splices, terminal blocks, electrical penetration or connector) in the circuit was included for aging management review, while the end device (the instrument and its power supply or other active electrical device) was excluded.

The offsite power grid boundary for the two preferred sources of offsite power are as follows:

- For the 115 KV line the boundary is each of the two circuit switchers associated with the plant's ESF Transformers.
- For the 230 KV substation bus 1, the boundary is the 2000 amp power circuit breaker feeding the two Emergency Auxiliary Transformers.

2.1.1.2 Safety-Related Criteria Pursuant To 10 CFR 54.4(a)(1)

Systems, structures, and components that are relied upon to remain functional during and following design basis events to ensure the functions specified in 10 CFR 54.4(a)(1), [**Reference 2.1-1**] are within the scope of license renewal.

10 CFR 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-basis 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 which could result in potential offsite exposures comparable to those referred to in §50.34(a)(1), §50.67(b)(2), or §100.11 of this chapter, as applicable.

2.1.1.2.1 Safety-Related Mechanical Systems

Guidance contained in ANS N18.2 [**Reference 2.1-5**] and Regulatory Guides 1.29 [**Reference 2.1-3**] and 1.143 [**Reference 2.1-6**] has been used to establish those VCSNS systems, structures, and components that satisfy the scoping criteria in 10 CFR 54.4(a)(1).

Regulatory Guide (RG) 1.29, “Seismic Design Classification” [**Reference 2.1-3**] 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, in conformance with the recommendations of RG 1.29 for balance of plant. Nuclear Steam Supply System fluid system components important to safety are classified in accordance with the August 1970 Draft of ANS N18.2, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants” [**Reference 2.1-5**], except that the components of the accumulator subsystem are classified in accordance with the 1973 version of ANS N18.2. The applicability of the requirements in RG 1.29 is summarized in **FSAR Section 3.2.1** [**Reference 2.1-7**], Seismic Classification. The FSAR describes how the systems, structures, and components meet the guidance contained in RG 1.29. Components of the Liquid and Gaseous Waste Processing Systems and the Boron Recycle System are classified in accordance with Regulatory Guide 1.143 [**Reference 2.1-6**].

Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants" [**Reference 2.1-4**] 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. Safety classes at VCSNS are established in conformance with ANS N18.2. These safety classes differ from the system of Quality Group Classification defined in RG 1.26, but meet the intent of RG 1.26, as described in **FSAR Appendix 3A, Conformance with Regulatory Guides**.

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 1, 2a, 2b, 3, NNS, and QR have been established for the classification of components and are defined in **FSAR Section 3.2.2**, System Quality Group Classifications. These categories are based on ANS Safety Classes, and reflect both nuclear safety-related (NSR) and non-nuclear safety (NNS) classifications.

Section 3.2.2 of the FSAR defines these Safety Classes as follows:

Safety Class 1 (SC-1) applies to Reactor Coolant System components, the failure of which could cause a Condition III or Condition IV loss of reactor coolant as defined by ANS N18.2.

Safety Class 2 (SC-2) applies to those components of safety systems required to fulfill a system function. Safety Class 2 is subdivided into Safety Class 2a and Safety Class 2b. A safety system (in this context) is any system that functions to shutdown the reactor, cool the core, cool another safety system or the containment and contains, controls or reduces radioactivity released in an accident.

Safety Class 2a (SC-2a) applies to containment and to components of those safety systems, or portions thereof, through which reactor coolant water flows directly from the Reactor Coolant System or the Reactor Building recirculation sumps.

Safety Class 2b (SC-2b) applies to all other components of Safety Class 2.

Safety Class 3 (SC-3) applies to components not classified as Safety Class 1 or Safety Class 2 and:

- The failure of which would result in the release to the environment of radioactive gases normally required to be held for decay. (The Liquid and Gaseous Waste Processing Systems and the Boron Recycle System are designated non-nuclear safety consistent with Regulatory Guide 1.143 [**Reference 2.1-6**].)
- That provide or support any safety system function,
- That control airborne radioactivity released outside the reactor containment, or

- That remove decay heat from spent fuel.

Non-Nuclear Safety applies to those structures, components and systems which do not fall into the above safety class categories and are designated as Non-Nuclear Safety (NNS).

Quality-Related (QR) applies to designated parts, components, systems, and associated activities that are not Nuclear Safety-Related, but warrant application of a Quality Plan or Program to satisfy regulatory requirements or management decisions.

Comparison of the ANS N18.2 Safety Class criteria, as implemented at VCSNS, to the criteria of 10 CFR 54.4(a)(1) shows that the VCSNS Safety Classes clearly encompass the systems and equipment that meet the criteria of 10 CFR 54.4(a)(1)(i), (ii), and (iii).

License Renewal scoping of mechanical systems, and mechanical portions of non-mechanical systems, at VCSNS relies primarily on the ANS N18.2 safety classifications. All safety-related mechanical systems, and mechanical portions of non-mechanical systems, are considered to be within the scope of the License Renewal Rule. **FSAR Table 3.2-1**, *Mechanical Equipment Classification*, provides a listing of major mechanical components and identification of the applicable safety class, code class, seismic class, and quality assurance class for each. The VCSNS system flow diagrams contain boundary flags which identify the safety classification, as defined above, of the applicable components.

The listing of mechanical systems and components required for compliance with 10 CFR 54.4(a)(1) and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.2.2 Safety-Related Structures

All structures at VCSNS 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 100 [**Reference 2.1-19**], Seismic and Geological Siting Criteria for Nuclear Power Plants, requires that all nuclear power plants be designed such that, if a Safe Shutdown Earthquake (SSE) 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 shutdown 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 of 10 CFR Part 100.

These three functions are identical to 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 Regulatory Guide (RG) 1.29 [**Reference 2.1-3**] as Seismic Category I. All safety-related structures at VCSNS 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 **FSAR Table 3.2-2** [**Reference 2.1-7**], Classification of Structures. Category I structures are identified through a review of the FSAR.

The listing of structures necessary to meet the requirements of 10 CFR 54(a)(1) [**Reference 2.1-1**] and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.2.3 Safety-Related Electrical Systems

Electrical components at VCSNS are classified as either Class 1E, as defined in IEEE-380 [**Reference 2.1-22**], or as Non-Nuclear Safety (NNS). Class 1E is the safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material to the environment.

These functions are the electrical equivalent to the functions specified in the scoping criteria in 10 CFR 54.4(a)(1) [**Reference 2.1-1**]. All electrical systems that contain equipment classified as Class 1E are considered to be safety-related and are within the scope of license renewal. Class 1E equipment is identified through a review of the VCSNS component database.

The listing of electrical systems and components required for compliance with 10 CFR 54(a)(1) and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.3 Non-Safety-Related Criteria Pursuant To 10 CFR 54.4(a)(2)

All non-safety-related systems, structures, and components, whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii) [**Reference 2.1-1**] are within the scope of license renewal.

10 CFR 54.4 (a)(2)

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

(2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.

Two types of systems and structures must be considered for inclusion within the scope of license renewal per 10 CFR 54.4(a)(2):

1. Non-safety-related systems and structures, and non-safety-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.

2. Non-safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

Provided below are the methodologies for identifying the mechanical systems, the structures, and the electrical systems and components that satisfy the scoping criteria in 10 CFR 54.4(a)(2).

2.1.1.3.1 Non-Safety-Related Mechanical Systems

In a letter from Christopher I. Grimes to Alan Nelson and David Lochbaum dated March 15, 2002, [Guidance On The Identification And Treatment Of Structures, Systems, And Components Which Meet 10 CFR 54.4(a)(2)] the staff identified a position on these interactions. Non-safety/safety interactions will need to be addressed in more detail at VCSNS to address the staff's concerns. Several sources of information are available for the identification of non nuclear safety related SSCs that meet the requirements of 10 CFR 54.4(a)(2). The VCSNS FSAR and various systems Design Basis Documents are the major source of this information. The plant equipment database identifies components that are not directly nuclear safety related, but that could have an impact on the ability of nuclear safety related SSCs to perform their required functions. The Maintenance Rule 10 CFR 50.65, includes scoping criteria for non-safety-related SSCs. The maintenance rule scoping criteria are similar to the license renewal scoping criterion for non-safety-related SSCs. The results of the searches of these sources are summarized below for the mechanical systems.

High Energy Piping - Portions of several mechanical systems at VCSNS have specific analysis performed due to NNS pressure boundary (pipe break) concerns. These are piping and components, which are not classified as ASME Code Class piping but whose failure due to seismic and/or pipe rupture considerations could impair the functioning of a NSR structure, system, or component. These mechanical systems, or portions of systems, and the associated supports, rupture restraints, and jet shields are within the scope of license renewal. Piping, that is not specifically analyzed, was not initially considered within the scope of license renewal. (The associated supports rupture restraints and jet shields are within the scope of license renewal.) The non-analyzed piping will be addressed (i.e. scoping, screening and aging management review) by SCE&G in a timely manner. These results will be provided in a supplementary submittal to the NRC.

Alternate Isolation of Steamlines - Several non-nuclear safety related systems are credited for alternate isolation of a steamline break. **FSAR Section 10.3.2.3** provides a discussion of main steam isolation. The position is consistent with NUREG-0138 issue 1. Should a Main Steam Isolation Valve (MSIV) fail to close, flow from the system downstream of the MSIV is limited. Flow is limited by either normally closed valves or by valves that are automatically closed by interlocks from the NSSS system or from the turbine generator system as tabulated in **FSAR Table 10.3-2**. The applicable portions of these systems are within the scope of license renewal.

Instrument Air - Air is required for certain valves and equipment that require control air to perform or support a license renewal function. Portions of the air systems supplying certain valves and equipment that require control air to perform or support a license renewal function are within the scope of license renewal.

Flooding - Flooding resulting from gross leakage from mechanical system components into adjacent areas may prevent the performance of a safety function. In this case, the detection of the mechanical component failure and its isolation are of concern. Systems that are credited for detection and isolation of leaks, to preclude adverse affects on safety-related equipment and functions, are within the scope of license renewal.

Insulation - Insulation is found on piping system components in many of the mechanical systems. Insulation is controlled for seismic/anti-falldown concerns. The insulation does not have a license renewal intended function as identified in NEI 95-10. Seismic/anti-falldown insulation will be addressed (i.e. scoping, screening and aging management review) by SCE&G in a timely manner. These results will be provided in a supplementary submittal to the NRC.

Seismic, Code Break and Leaks - Structural supports that are considered to meet seismic (anti-falldown) criteria or code break criteria, are within the scope of license renewal. These are not included in the mechanical system scoping and screening, but are treated as a structural commodity. SCE&G is participating in the current efforts of the industry to develop a methodology to address issues with the piping and ductwork. The review of piping and ductwork for VCSNS has not been completed at the time of application submittal. The piping and ductwork will be addressed (i.e. scoping, screening and aging management review) by SCE&G in a timely manner. These results will be provided in a supplementary submittal to the NRC.

The listing of mechanical systems and components required for compliance with 10 CFR 54.4(a)(2) and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.3.2 Non-Safety-Related Structures

Structures at VCSNS are classified as either nuclear safety-related (NSR) or non-nuclear safety (NNS). The safety related structures are designed to withstand the Safe Shutdown Earthquake (SSE) and are classified as Seismic Category I, while the non-safety-related structures are generally not designed to SSE seismic levels and are classified as non-seismic. Systems and components that have been seismically mounted to meet anti-falldown criteria are classified as Seismic Category II. The standard industry term for anti-falldown is Seismic II/I (Seismic two over one).

There are no structures designated as Seismic Category II at VCSNS. Non-safety-related structures whose failure could impair the function of safety-related systems, structures, and components are designated as non-seismic but have been designed to withstand earthquake loads and tornado wind loads to the extent required for prevention of damage to Seismic Category I structures.

The listing of systems and structures required for compliance with 10 CFR 54.4(a)(2) and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.3.3 Non-Safety-Related Electrical Systems

Electrical systems and portions of electrical systems that are non-safety-related but whose failure could prevent the satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), and (iii) [**Reference 2.1-1**] are within the scope of license renewal (as outlined in 10 CFR 54.4(a)(2)). The components of the electrical systems that perform these functions are designated as Quality-Related (QR). The VCSNS equipment database contains the electrical equipment designated as Quality-Related.

The listing of electrical systems and components required for compliance with 10 CFR 54.4(a)(2) and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.4 Regulated Events Pursuant To 10 CFR 54.4 (a)(3)

In addition to those systems, structures, and components relied upon to mitigate design basis events (10 CFR 54.4(a)(1)) [**Reference 2.1-1**], or whose failure could prevent mitigation of design basis events (10 CFR 54.4(a)(2)), the systems that are committed/credited to support compliance with the NRC regulations identified in 10 CFR 54.4(a)(3) must be identified for license renewal.

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 methodology for identifying the systems, structures, and components required to demonstrate compliance with the regulations identified in 10 CFR 54.4(a)(3) is provided in this section. The methodology involves an extensive review of safety evaluation reports (SERs), the Fire Protection Evaluation Report (FPER) [**Reference 2.1-9**], the FSAR, design basis documents, licensee event technical reports, licensing correspondence files, and other design and licensing documentation. The following sections provide discussions related to the methodology applied for each of the specified regulated events.

2.1.1.4.1 Fire Protection (FP)

Systems relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with 10 CFR 50.48, “Fire Protection”, are within the scope of license renewal per 10 CFR 54.4(a)(3) [**Reference 2.1-1**].

10 CFR 50.48 (b)

(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 for nuclear power plants licensed to operate before January 1, 1979.

(1) Except for the requirements of Sections III.G, III.J, and III.O, the provisions of Appendix R to this part do not apply to nuclear power plants licensed to operate before January 1, 1979, to the extent that--

(i) 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 NRC fire protection safety evaluation reports issued before the effective date of February 19, 1981; or

(ii) Fire protection features were accepted by the NRC 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.

(2) With respect to all other fire protection features covered by Appendix R, all nuclear power plants licensed to operate before January 1, 1979, must satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O.

The VCSNS fire protection program is based on an evaluation of potential fire hazards throughout areas containing safe shutdown equipment as well as potential fire hazards in various non-safe shutdown facilities/areas. The evaluation of potential fire hazards assures that the capability exists to safely shutdown the unit following loss of functions in any given fire area due to a fire, in compliance with General Design Criteria (GDC) 3, “Fire Protection” [**Reference 2.1-15**], and Branch Technical Position (BTP) APSCB 9.5-1, Appendix A [**Reference 2.1-16**].

NUREG-0717, including the Supplements, [**Reference 2.1-8**] includes documentation of the NRC review and acceptance of the VCSNS fire protection program. The Fire Protection Evaluation Report (FPER) [**Reference 2.1-9**] contains the essential elements of the program. These elements are the fire hazards analysis, safe plant shutdown description, and a point-by-point comparison of the program with the requirements of BTP 9.5-1, Appendix A. The systems containing the equipment that demonstrate compliance in this comparison are considered to be in license renewal scope. The listing of equipment required for Appendix R

Safe Shutdown is documented in a design calculation. Systems containing the equipment listed in this calculation are considered to be in the license renewal scope.

In order to ensure compliance with 10 CFR 50, Appendix R (Sections III.G, III.J, and III.O), VCSNS performed additional analyses to provide further documentation of the ability of the unit to achieve safe shutdown in the event of a fire. These analyses are documented in the FPER [**Reference 2.1-9**]. To safely shutdown the plant without control from the Control Room, an alternate shutdown system is provided consisting of two independent shutdown panels and utilizing some local operator action. NRC review and acceptance of the VCSNS Appendix R evaluations and conclusions are documented in safety evaluation reports [**Reference 2.1-10**, **2.1-11**, **2.1-12**].

Although fire protection equipment is not considered safety related, the Quality Assurance (QA) program for fire protection is part of the overall QA program, and installation, testing, and subsequent operations for areas containing safety related equipment are processed by procedures similar to those utilized for safety related work.

The listing of systems, structures and components required for compliance with 10 CFR 50.48 and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.4.2 Environmental Qualification (EQ)

Electrical systems and components relied upon in safety analyses or in plant evaluations to perform a function that requires demonstrated compliance with 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants”, are within the scope of the license renewal per 10 CFR 54.4(a)(3) [**Reference 2.1-1**].

10 CFR 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 certifications 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.
(i) This equipment is that relied upon to remain functional during and following design basis events to ensure --
(A) The integrity of the reactor coolant pressure boundary;
(B) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
(C) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in §50.34(a)(1), §50.67(b)(2), or §100.11 of this chapter, as applicable.
(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.
(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1) (i)(A) through (C) of paragraph (b)(1) of this section by the safety-related equipment.
(3) Certain post-accident monitoring equipment.

10 CFR 50.49 requires each operating nuclear power plant to establish a program for qualifying certain electrical equipment. NUREG-0588 [Reference 2.1-13] defines the NRC staff position regarding selected areas of environmental qualification (EQ) of electrical equipment important to safety. NUREG-0588 requires that all plant electrical components that are located in a harsh environment and which must function to mitigate a design basis accident, must be qualified to operate in that environment. The components defined within the EQ program include both safety-related and non-safety-related electrical components required for accident mitigation, post-accident monitoring, and safe shutdown.

Components required to be environmentally qualified in accordance with 10 CFR 50.49 are identified in a controlled database. Systems that contain components required to demonstrate compliance with 10 CFR 50.49 are considered within the scope of license renewal.

The listing of electrical systems and components required for compliance with 10 CFR 50.49 and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.4.3 Pressurized Thermal Shock (PTS)

Systems and structures relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", are within the scope of license renewal per 10 CFR 54.4(a)(3) [**Reference 2.1-1**]

10 CFR 50.61(a)(2)

(a) Definitions. For the purposes of this section:
(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.

10 CFR 50.61 contains requirements for utilities to minimize the effects of pressurized thermal shock (PTS) to the reactor vessel. This concern exists during periods in which cold water may be injected into the reactor vessel at relatively high system pressures (e.g., safety system injection after an accident). In addition to the reactor vessel, any systems and equipment needed to prevent potential failure of the reactor vessel associated with pressurized thermal shock is within the scope of license renewal.

The identification of mechanical systems, other than the reactor vessel, relied on to demonstrate compliance with 10 CFR 50.61 required a review of docketed licensing correspondence and related Westinghouse technical reports (WCAPs).

The listing of systems, structures and components required for compliance with 10 CFR 50.61 and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.4.4 Anticipated Transient Without Scram (ATWS)

Systems and structures relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with 10 CFR 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 per 10 CFR 54.4 (a)(3) [**Reference 2.1-1**].

10 CFR 50.62 (b)

(b) Definition. For purposes of this section, 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.

10 CFR 50.62(c)(1)

(c) Requirements.

(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.

In response to this regulation VCSNS installed the ATWS Mitigation System Actuation Circuitry (AMSAC) control system. NRC review and acceptance of the VCSNS response to 10 CFR 50.62 is documented in an NRC safety evaluation report [**Reference 2.1-14**].

Systems that provide input into this control system or respond to an output of this control system are part of the commitment to the ATWS rule and are therefore within the scope of license renewal.

The listing of systems, structures and components required for compliance with 10 CFR 50.62 and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.4.5 Station Blackout (SBO)

Systems and structures relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with 10 CFR 50.63, "Loss of all Alternating Current Power", are within the scope of license renewal per 10 CFR 54.4 (a)(3) [**Reference 2.1-1**].

> **10 CFR 50.63(a)(1)**
>
> *(a) Requirements.*
>
> *(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 10 CFR 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.*

10 CFR 50.63, commonly known as the Station Blackout Rule, requires that each light-water cooled nuclear power plant be able to withstand and restore power after a station blackout of a specified duration. Station blackout is the complete loss of normal AC power and failure of dedicated emergency AC power. For VCSNS, station blackout is defined as the complete loss of offsite and onsight AC power for four (4) hours. NRC Regulatory Guide 1.155 [**Reference 2.1-17**] and NUMARC 87-00 [**Reference 2.1-18**] provide the guidance for coping duration and requirements. NRC review and acceptance of the VCSNS plan for coping with a station blackout is documented in safety evaluation reports [**Reference 2.1-20**, **2.1-21**].

The listing of systems, structures and components required for compliance with 10 CFR 50.63 and therefore within the scope of license renewal is found in **Section 2.2**.

2.1.1.5 Systems And Structures Intended Functions

Intended functions of systems and structures are defined in accordance with 10 CFR 54.4(b) [**Reference 2.1-1**] for the systems and structures within the scope of license renewal. The intended functions define the plant process, condition, or action that must be accomplished in order to perform or support a satisfactory safety response during or following a design basis event or to perform or support a specific requirement of one of the five regulated events in 10 CFR 54.4(a)(3).

10 CFR 54.4 (b)

The intended functions that these systems, structures, and components must be shown to fulfill in §54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1) - (3) of this section.

For each system and structure within the scope of license renewal, intended functions are established based on the results of the scoping process. The system and structure functions established for the Maintenance Rule program are used as the starting point for establishing the intended functions for license renewal. The Maintenance Rule functions that do not meet the criteria of 10 CFR 54.4 are excluded. Functions in addition to those established by the Maintenance Rule are defined through a review of pertinent plant documentation.

The system and structure intended functions are used during the screening portion of license renewal to establish the structures and components required to support those functions.

2.1.2 SCREENING METHODOLOGY

Screening is the process of identifying the structures and components that are subject to aging management review. The criteria that determine the structures and components subject to aging management review for license renewal are provided in 10 CFR 54.21(a)(1) [**Reference 2.1-1**].

10 CFR 54.21(a)(1)

Each application must contain the following information:
(a) An integrated plant assessment (IPA). The IPA must --
(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. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and
(ii) That are not subject to replacement based on a qualified life or specified time period.

The screening methodology for VCSNS is discussed by engineering disciplines; mechanical, civil/structural, and electrical/instrumentation and control (I&C). The discipline separation for license renewal is that components or component parts which carry current are considered Electrical; components that support, protect or restrain movement are considered Civil/Structural; and everything else is Mechanical. The relevant aspects of the screening process for mechanical system components, structural components, and electrical/I&C system components are described in **Sections 2.1.2.1**, **2.1.2.2**, and **2.1.2.3**, respectively.

2.1.2.1 Screening Methodology Mechanical Systems

For mechanical systems, the screening process is performed on each system identified to be within the scope of license renewal. The process includes the following steps to identify specific components or commodity groups that require an aging management review:

1. Establish system evaluation boundaries.
2. Determine components within the system evaluation boundaries.
3. Identify component intended functions.
4. Determine components subject to aging management review.
5. Commodity grouping (material & environment identification).

2.1.2.1.1 Establish System Evaluation Boundaries

Mechanical system evaluation boundaries are established for each system within the scope of license renewal. Precise physical/functional boundaries are necessary to assure that all components required to support system intended functions, which meet the requirements of 10 CFR 54.4(a)(1,2 and/or 3) [**Reference 2.1-1**], are considered for subjection to an aging management review. These boundaries are determined by mapping the flow paths, including pressure boundary, that are necessary for the accomplishment of identified system intended functions onto the system flow diagrams or other drawings, such as FSAR figures. The mechanical components found within the mapped portions of these boundary drawings comprise the complete set of mechanical components within the scope of license renewal. The flow diagram or other boundary drawings associated with each mechanical system within the scope of license renewal are identified with the mechanical system screening results in **Section 2.3**.

The components included within the license renewal evaluation boundaries of a given system have generally been assigned, at the plant, to that parent system. However, in the cases where a system's intended function is performed/supported by a component assigned to another system, the component is included in the evaluation of the system whose intended function it performs/supports rather than the parent system.

2.1.2.1.2 Determine Components Within The System Evaluation Boundaries

A menu listing all passive, long-lived mechanical components or component groupings was developed, based on the guidance in NEI 95-10, Appendix B [**Reference 2.1-2**], using plant system flow diagrams, design guidelines, and the plant component database for consistency with standard plant usage. The components within the mapped areas of the license renewal evaluation boundary diagrams for each system are compared to the menu as a step in listing the components that are subject to aging management review.

Sub-components of heat exchangers and Class 1 components, including non-Class 1 Steam Generator and Reactor Coolant Pump portions, are identified by individual use and characteristics. Although not normally depicted on the boundary drawings, these sub-components are often made of different materials, may have different design characteristics, and may perform different functions at the sub-component level and as such warrant unique consideration. Furthermore, Class 1 bolting is larger in size than non-class 1 bolting and covered by specific ASME Section XI activities.

Otherwise, sub-components are excluded from the menu and comparison and are considered a piece-part of the main component and so addressed implicitly for license renewal at the component/component group level. This is particularly true of Non-Class 1 closure bolting, except for Steam Generator and Reactor Coolant Pump closures. Non-Class 1 mechanical components within the scope of license renewal contain bolted closures that are necessary for the pressure boundary of the component and are comprised of two mating surfaces, a gasket, and a fastener set. By themselves, the mating set, gasket, and fastener set have no component intended function for license renewal. Together, the bolted closure forms an integral part of the pressure-retaining boundary of the component and is exposed to the same internal and external environments.

2.1.2.1.3 Identify Component Intended Functions

A list of potential mechanical component intended functions is developed for each grouping of the components within the mechanical evaluation boundaries based on the guidance of NEI 95-10 [**Reference 2.1-2**]. In accordance with 10 CFR 54.21(a)(1) [**Reference 2.1-1**], component intended functions are those component-level functions that are performed without moving parts or without a change in configuration or properties in support of identified system intended functions. The result is a list of the potential intended functions for each passive, long-lived component types.

Each mechanical component or component group (commodity) within the license renewal evaluation boundaries is reviewed to determine whether the potential intended functions must be performed by that component to meet the requirements that are necessary to enable/ensure that the identified system intended functions for license renewal are accomplished. The functions that must be performed are the actual component intended functions.

2.1.2.1.4 Determine Components Subject To Aging Management Review

The passive, long-lived mechanical components that perform at least one actual component intended function are subject to an aging management review. As described in **Section 2.1.2.1.1**, the evaluation boundaries include all components in the flow paths, including pressure boundary, required to ensure accomplishment of identified license renewal system intended functions (i.e. in-scope components). This provides assurance that all components in the scope of license renewal are considered.

However, consistent with the screening criterion of 10 CFR 54.21(a)(1)(i) [**Reference 2.1-1**], only in-scope components that perform an intended function without moving parts or without a change in configuration or properties are subject to an aging management review. Of these, components that are not subject to replacement based on a qualified life or specified time period [screening criterion of 10 CFR 54.21(a)(1)(ii)] are identified and documented as subject to aging management review. All other components in the scope of license renewal are screened out.

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-retaining 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. However, similar to pumps and valves where only the active portion of the component can be excluded in accordance with 10 CFR 54.21(a)(1)(i) [**Reference 2.1-1**], ductwork (housings) that surround in-scope ventilation equipment is subject to aging management due to the passive pressure boundary considerations.

Filter mediums such as paper filters, charcoal filters, and resins are within the scope of license renewal, but are replaced on condition and not subject to an aging management review. Filter mediums experience a change of configuration or properties in that either the contaminant is caught in a mesh or the medium is chemically bonded with the contaminant. In either case, the overall effect on the medium is a discernible change in configuration or properties as evidenced by an increase of differential pressures across the filter or the decrease of absorption efficiency trended during periodic testing. The filter mediums are replaced or cleaned as conditions warrant and are, therefore, not subject to an aging management review. However, for completeness filtration components are identified as applicable on the evaluation boundary drawings and the particular component function identified.

Portable equipment such as fire extinguishers, self-contained breathing air packs, fire hoses, and portable ductwork are within the scope of license renewal but exempt from an aging management review, in that such portable equipment is typically replaced on condition. Such equipment is not expected to last either forty or sixty years and is routinely inspected for degradation per 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 maintained per the NFPA standards, therefore, an aging management review is not required.

2.1.2.1.5 Establish Commodity Groupings

To facilitate the aging management review, many components are grouped together so that a single aging management review is performed based on common characteristics such as material or environment.

For each mechanical component and component type (commodity) subject to aging management review, the internal and external operating environments to which the component is subjected are established. Operating environments are established based on a review of plant design documents, the FSAR, plant drawings, and plant environmental data.

The materials of construction for the components and component types subject to aging management review are determined based on a review of plant design documents, the FSAR, vendor drawings, specifications, and VCSNS component database.

Components with similar design, materials of construction, and subjected to similar environments within an individual system are evaluated as a commodity group (e.g. pipe). Commodity groups are not used for components with unique design characteristics, such as heat exchangers, pumps, and tanks or Class 1 sub-components.

2.1.2.1.6 Results

The tables contained in Section 2.3 list the mechanical components and commodity groups that are subject to aging management review along with their intended functions. Mechanical components or commodities within the evaluation boundaries of a system that are not listed in the tables are not subject to aging management review because they do not meet the criteria of 10 CFR 54.21(a) [**Reference 2.1-1**] or are specifically excluded by 10 CFR 54.21(a)(1)(i).

2.1.2.2 Screening Methodology For Structures

The screening process for structures is performed on each structure identified to be within the scope of license renewal in accordance with the methodology of **Section 2.1.1**. The screening process identifies the structural components that are subject to an aging management review for each structure within the scope of license renewal.

2.1.2.2.1 Establish Structure Evaluation Boundaries

Structures support, house, and protect systems and components that are relied upon to operate the plant safely. Structures are comprised of structural items such as beams, columns, floors, walls, and foundations. To facilitate the identification of the structural components subject to an aging management review, the evaluation boundaries between mechanical components, electrical components, and structures and structural components subject to an aging management review are established.

The evaluation boundary of the structures in scope is the entire building, including foundation slabs, external and internal walls, roof and internal concrete, and steel columns and beams. Other long-lived passive items within the building such as structural supports (e.g., hangers, cable trays, miscellaneous supports), equipment supports and base plates, specialty doors, curbs, penetration assemblies, jet shields, and instrument racks and frames are grouped as structural component types and are subject to an aging management review.

2.1.2.2.2 Determine Structural Components Within The Evaluation Boundaries

A generic list of structural component types is developed. Based on the generic list, a comprehensive list of the types of structural components that exist within VCSNS evaluation boundary is developed. The list for VCSNS is developed based on a review of plant documentation, including design drawings, specifications, design basis documents, the FSAR, and the component database.

2.1.2.2.3 Determine Components Subject To Aging Management Review

The structural components subject to aging management review are those components that perform an intended function without moving parts or without a change in configuration or properties and are not subject to replacement based on a qualified life or specified time period.

The structural components within the VCSNS evaluation boundaries are reviewed to determine whether they perform their intended function in an active or passive manner. The active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10 [**Reference 2.1-2**]. Most structural components have no moving parts and do not change configuration or properties.

The passive structural components within the VCSNS evaluation boundaries are reviewed to determine whether they are subject to replacement based on a qualified life or specified time period. Those structural components not subject to replacement based on these criteria are considered long-lived and are subject to an aging management review.

For concrete structures and structural components, VCSNS has used the Part 54 process, NUREG-1801, and industry guidelines to determine those specific aging effects that are applicable and require aging management for the extended period of operation. Recent positions by the NRC Staff have determined that all aging effects for concrete are credible and should be managed under the current licensing basis programs for the extended period of operation.

2.1.2.2.4 Identify Component Intended Functions

A list of potential structural component intended functions is developed using the guidance in NEI 95-10 [**Reference 2.1-2**] for each of the component types within the structural evaluation boundaries based on the intended functions of the structure. For each passive, long-lived structural component, the potential intended functions are reviewed to determine which of the functions could be performed by the structural component type. The result is a list of the potential intended functions for each passive, long-lived component type.

Each structural component of the identified component types is reviewed to determine whether the potential intended functions must be performed by that structural component to meet the requirements that are the basis for including the component within the scope of license renewal. The functions that must be performed are the actual component intended functions.

The passive, long-lived structural components that perform at least one component intended function are subject to an aging management review.

2.1.2.2.5 Establish Commodity Groupings

To facilitate the aging management reviews, the structural components are divided into major groupings based on materials of construction and operating environment.

For each structural component subject to aging management review, the internal and external operating environments to which the component is subjected are established. Operating environments are established based on a review of plant design documents, the FSAR, plant drawings, and plant environmental data.

For each structural component subject to aging management review, the materials of construction are determined based on a review of plant design documents, the FSAR, vendor drawings, specifications, and component databases.

Components with similar design, materials of construction, functions, and subjected to similar environments are evaluated as a commodity group.

2.1.2.2.6 Results

The tables contained in **Section 2.4** list all structural components and commodity groups that are subject to aging management review along with the intended functions of the structural components and commodity groups.

2.1.2.3 Screening Methodology For Electrical/I&C Components

For electrical and instrumentation and control (I&C) components, the methodology used to determine which components are subject to an aging management review is organized differently than for the mechanical and structural evaluations. The primary difference in the electrical and I&C methodology is the order in which the component screening steps are performed. Electrical equipment contained in mechanical systems or structures considered within the scope of license renewal are carried forward as electrical commodities and are screened as described in this section. Component commodity groups are established at the start of the process and the screening criteria are applied to the commodity groups. This method was selected since most electrical and I&C components are active. Thus, this method provides the most efficient means for determining the electrical and I&C components that require an aging management review. The method employed is consistent with the guidance in NEI 95-10 [**Reference 2.1-2**].

2.1.2.3.1 Establish Commodity Groups And Intended Functions

The listing of commodity groups for electrical and instrumentation & control (I&C) components in Appendix B of NEI 95-10 [**Reference 2.1-2**] is used as the starting point for the establishment of electrical commodity groups for VCSNS. The initial listing of electrical commodity groups is compared to plant design information to ensure that all electrical and I&C components at VCSNS are included.

The intended functions for each of the electrical commodity groups are then identified. The electrical components groupings are adjusted, as necessary, based on similar design and function attributes. Using the guidance of NEI 95-10 the component intended functions are defined for each electrical commodity group.

2.1.2.3.2 Determine Commodity Groups Subject To Aging Management Review

The intended functions established for each of the electrical commodity groups are compared with the criteria of 10 CFR 54.21(a)(1)(i) and (ii) [**Reference 2.1-1**].

The electrical and instrumentation and control (I&C) components commodity groups that perform an intended function without moving parts or without a change in configuration or properties are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10 [**Reference 2.1-2**].

The passive electrical commodity groups that are not subject to replacement based on a qualified life or specified time period are identified as requiring an aging management review.

2.1.2.3.3 Results

The tables contained in **Section 2.5** list all electrical commodity groups that are subject to aging management review along with their intended functions.

2.1.3 CONCLUSIONS

The methods described in **Sections 2.1.1** and **2.1.2** were used by VCSNS to identify the systems, structures, and components that are within the scope of license renewal and require an aging management review. The methods are consistent with and satisfy the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.1.4 REFERENCES

2.1-1	10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 60 FR 22461, May 8, 1995.
2.1-2	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.
2.1-3	Regulatory Guide 1.29, Revision 2, “Seismic Design Classification”, February 1976
2.1-4	Regulatory Guide 1.26, Revision 2, “Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants”, January 1976.
2.1-5	ANS N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, August 1970 Draft.
2.1-6	Regulatory Guide (RG) 1.143, Revision 2, 'Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water Cooled Nuclear Power Plants”, November 2001.
2.1-7	VCSNS Final Safety Analysis Report (FSAR), through amendment 02-01.
2.1-8	NUREG-0717, "Safety Evaluation Report (SER) Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1", USNRC, dated February 1981 (includes Supplements).
2.1-9	VCSNS Fire Protection Evaluation Report (FPER), Amendment 02-01.
2.1-10	NRC Safety Evaluation Report (SER) for Appendix R Reanalysis, dated February 22, 1986.
2.1-11	NRC Supplemental Safety Evaluation Report (SER) for Appendix R Reanalysis, dated November 26, 1986.
2.1-12	NRC Supplemental Safety Evaluation Report (SER) - Deviation from 10 CFR Part 50 Appendix R Section III.G Safe Shutdown Capability, dated October 10, 1997.
2.1-13	NUREG-0588, Revision 1, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, November 1980.
2.1-14	VCSNS Safety Evaluation Report (SER) on ATWS Rule (10 CFR 50.62), dated June 28, 1988.
2.1-15	10 CFR 50.48 Appendix A, General Design Criteria 3, Fire Protection.

2.1-16	Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976", August 23, 1976.
2.1-17	Regulatory Guide (RG) 1.155, Station Blackout, August 1988.
2.1-18	NUMARC 87-00, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors.
2.1-19	10 CFR 100, Appendix A, Seismic and Geological Siting Criteria for Nuclear Power Plants.
2.1-20	NRC Safety Evaluation Report (SER) Regarding Station Blackout, dated January 30, 1992
2.1-21	NRC Supplemental Safety Evaluation Report Regarding Station Blackout, dated June 1, 1992
2.1-22	IEEE-38--1975, Definitions of Terms Used in IEEE Standards on Nuclear Power Generating Stations.

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**, the criteria that determine the systems, structures, and components that are within the scope of license renewal are provided in 10 CFR 54.4 [**Reference 2.2-1**]. Guidance contained in ANS N18.2 [**Reference 2.2-2**] and Regulatory Guides 1.29 [**Reference 2.2-3**] and 1.143 [**Reference 2.2-4**] has been used to establish those VCSNS systems, structures, and components that satisfy the scoping criteria in 19 CFR 54.4(a)(1) as described in **Section 2.1.1.1**. All non-safety-related systems, structures, and components, whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 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**. Finally, in addition to those systems, structures, and components relied upon to mitigate design basis events (10 CFR 54.4(a)(1)), or whose failure could prevent mitigation of design basis events (10 CFR 54.4(a)(2)), the systems, structures, and components previously committed to support specific NRC regulations (10 CFR 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**.

The VCSNS mechanical systems that are within the scope of license renewal (10 CFR 54.4) are listed in **Table 2.2-1**. The mechanical systems and components that are subject to an aging management review are described in **Section 2.3**. 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 that are within the scope of license renewal are listed in **Table 2.2-2**. The structural components that are subject to an aging management review are described in **Section 2.4**. The electrical and instrumentation and controls (I&C) systems that are within the scope of license renewal are listed in **Table 2.2-3**. The electrical and I&C commodity groups that are subject to an aging management review are described in **Section 2.5**.

Tables 2.2-1, **2.2-2**, and **2.2-3** contain cross-references to the sections in the application that discuss screening results for in-scope systems and structures.

2.2.2 SYSTEMS AND STRUCTURES NOT WITHIN THE SCOPE OF LICENSE RENEWAL

The systems and structures that are not within the scope of license renewal are identified in **Tables 2.2-1**, **2.2-2**, and **2.2-3**. These listings are provided to assist the staff in its review of the scoping results. The systems and structures identified in these tables as “No” do not meet any of the criteria contained in 10 CFR 54.4 [**Reference 2.2-1**].

2.2.3 REFERENCES

2.2-1	10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 60 FR 22461, May 8, 1995.
2.2-2	ANS N18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, August 1970 Draft.
2.2-3	Regulatory Guide 1.29, Revision 2, “Seismic Design Classification”, February 1976
2.2-4	Regulatory Guide (RG) 1.143, Revision 2, 'Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water Cooled Nuclear Power Plants”, November 2001.
2.2-5	VCSNS Fire Protection Evaluation Report (FPER), Amendment 02-01.

Table 2.2-1:
MECHANICAL SCOPING RESULTS

Listing of Plant Mechanical Systems	Mechanical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
AC - Auxiliary Coolant (Closed Loop) / CRDM Cooling Water	Yes	NSR, EQ	2.3.2.2.1
AH - Air Handling (HVAC)	Yes	NSR, FP, EQ	2.3.3.1
AM - Ammonia	No		
AN - Auxiliary Steam Supply-Nuclear	No		
AR - Condenser Air Removal	No		
AS - Aux. Boiler Steam & Feed-water	Yes	NSR, NNS, EQ	2.3.4.1
AT - Condenser Cleaning	No		
BD - Steam Generator Blowdown	Yes	NSR, FP, EQ, SBO	2.3.4.11
BR - Boron Recycle	Yes	NSR	2.3.3.2
BS - Building Services	Yes	NSR, NNS, FP	2.3.3.3
CC - Component Cooling	Yes	NSR, FP, EQ, SBO	2.3.3.6
CD - Carbon Dioxide	No		
CI - Industrial Cooler	Yes	NSR	2.3.3.11
CL - Chlorine	No		
CN - CO2 Purge System-Nuclear	No		
CO - Condensate	Yes	NSR, FP, SBO	2.3.4.2
CS - Chemical and Volume Con-trol	Yes	NSR, NNS, FP, EQ, SBO	2.3.2.1
CV - Chemical & Volume Control Vents & Drains	Yes	NSR	2.3.2.1

Table 2.2-1:
MECHANICAL SCOPING RESULTS

Listing of Plant Mechanical Systems	Mechanical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
CW - Circulating Water	Yes	NNS, Flooding impact on NSR	2.3.3.5
DF - Diesel Generator Fuel	No		
DG - Diesel Generator Services	Yes	NSR, NNS, FP, SBO	2.3.3.7
DN - Demineralized Water-Nuclear Services	Yes	NSR	2.3.2.2.2
DO - Domestic Water	No		
DW - Demineralized Water	No		
EA - Environmental Services	No		
EF - Emergency Feedwater	Yes	NSR, FP, EQ, ATWS, SBO	2.3.4.3
EH - Turbine Electro-Hydraulic	Yes	NNS, ATWS	2.3.4.12
EO - Emergency Offsite Facility	No		
EQ - Emergency Equipment	No		
EX - Extraction Steam	Yes	NNS	2.3.4.4
FH - Fuel Handling	Yes	NSR	2.3.3.9
FI - Filtered Water	No		
FO - Fuel Handling, Oil	No		
FS - Fire Service	Yes	NSR, FP, EQ, SBO	2.3.3.8
FW - Feedwater	Yes	NSR, NNS, EQ	2.3.4.5
GE - Gas Sampling	No		
GS - Gland Sealing Steam	Yes	NNS	2.3.4.6

Table 2.2-1:
MECHANICAL SCOPING RESULTS

Listing of Plant Mechanical Systems	Mechanical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
GW - Gland Sealing Water	No		
HD - Heater Drains	No		
HM - Hydrogen Vent	No		
HN - Hydrogen, Nuclear Plant Use	No		
HR - Hydrogen Removal, Post Accident	Yes	NSR, EQ	2.3.2.3
HV - Heater Vent	No		
HY - Hydrogen	No		
IA - Instrument Air	Yes	NSR, NNS, EQ	2.3.3.12
IC - Incore Instrumentation (Tubes/Thimbles)	Yes	NSR, EQ	2.3.1.5
IV - Isolation Valve Seal Water	No		
LD - Leak Detection	Yes	NSR, NNS, EQ	2.3.3.13
LO - Lube Oil	No		
LR - Reactor Building Leak Rate Testing	Yes	NSR	2.3.2.2.4
LW - Liquid Effluents From Nuclear Plant to Pen Stock	No		
MB - Main Steam Dump	Yes	NNS, EQ	2.3.4.8
MD - Non-Nuclear Plant Drains	Yes	NNS, FP	2.3.3.15
ME - Miscellaneous Equipment	No		
MH - Material Handling	No		

Table 2.2-1:
MECHANICAL SCOPING RESULTS

Listing of Plant Mechanical Systems	Mechanical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
MS - Main Steam	Yes	NSR, NNS, FP, EQ, ATWS, SBO	2.3.4.7
MU - Reactor Makeup Water Supply	Yes	NSR	2.3.3.18
MV - Non-Nuclear Plant Vents	No		
NB - Nuclear Blowdown Processing	No		
ND - Nuclear Plant Drains	Yes	NSR, FP, EQ	2.3.3.15
NG - Nitrogen Blanketing	Yes	NSR	2.3.2.2.3
NN - Nitrogen, Nuclear Plant Use	No		
NP - Nitrogen Processing	No		
ON - Oxygen, Nuclear Plant Use	No		
OX - Oxygen	No		
PC - Penetration Cooling (Liquid)	No		
PP - Penetration Pressurization	No		
PR - Condenser Priming	No		
RC - Reactor Coolant	Yes	NSR, NNS, FP, EQ, PTS, SBO	2.3.1
RD - Roof Drains	Yes	NSR	2.3.3.19
RH - Residual Heat Removal	Yes	NSR, NNS, FP, EQ	2.3.2.6
RM - Radiation Monitoring	Yes	NSR, EQ	2.3.3.17
RS - Reheat Steam	No		
RW - Refueling Water	Yes	NSR, FP	2.3.2.5

Table 2.2-1:
MECHANICAL SCOPING RESULTS

Listing of Plant Mechanical Systems	Mechanical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
SA - Station Service Air	Yes	NSR, NNS	2.3.3.20
SC - Generator Stator Cooling Water	No		
SD - Sump Drain	No		
SE - Sewer	No		
SF - Spent Fuel Cooling	Yes	NSR	2.3.3.22
SI - Safety Injection	Yes	NSR, FP, EQ	2.3.2.7
SM - Site Maintenance	No		
SO - Seal Oil	No		
SP - Reactor Building Spray	Yes	NSR, FP, EQ	2.3.2.4
SR - Spent Resin	No		
SS - Nuclear Sampling	Yes	NSR, NNS, FP, EQ	2.3.3.16
SW - Service Water	Yes	NSR, FP, EQ	2.3.3.21
TA - Turbine Accessories	Yes	NNS	2.3.4.9
TB - Main Turbine	Yes	NNS	2.3.4.9
TC - Turbine Building Closed Cycle Cooling Water	No		
TD - Turbine Drains	No		
TR - Thermal Regeneration	Yes	NSR	2.3.3.23
VL - Local Ventilation & Cooling	Yes	NSR, FP, EQ	2.3.3.1
VU - Chilled Water	Yes	NSR, FP	2.3.3.4
WA - Turbine Cycle Sampling	Yes	NNS	2.3.4.10

Table 2.2-1:
MECHANICAL SCOPING RESULTS

Listing of Plant Mechanical Systems	Mechanical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
WC - Turbine Cycle Chemical Feed	No		
WD - Radwaste Solidification & Solids Handling	No		
WF - Cycle Make-Up Water Treatment	No		
WG - Gaseous Waste Processing	Yes	NSR	2.3.3.10
WI - Condensate Demineralization	No		
WL - Liquid Waste Processing	Yes	NSR, EQ	2.3.3.14
WM - Chemical Cleaning	No		
WN - Circulating Water Clarification	No		
WP - Waste Processing (Industrial)	No		
WT - Raw Water Treatment	No		
WX - Excess Liquid Waste	No		

Table 2.2-1 Notation:

NSR Systems that are designated Nuclear Safety-Related and/or contain ANS Safety Class 1, 2a, 2b, and/or 3 piping and components; intended functions are those that meet the requirements of 10 CFR 54.4(a)(1).

NNS	Other non-nuclear safety-related systems, or non-safety-related portions of nuclear safety-related systems, which may or may not contain anti-falldown piping, conduits, or components, but which provide support functions for nuclear safety-related systems in the mitigation of design basis events and/or maintain pressure boundary to prevent flooding/spray impacts on nuclear-safety related equipment. Intended functions are those that meet the requirements of 10 CFR 54.4(a)(2).
FP	Systems and functions required to assure a safe shutdown following a fire, either Appendix A or Appendix R, in compliance with 10 CFR 50.48. The Fire Protection Evaluation Report (FPER) [**Reference 2.2-5**] identifies the functions required for fire protection.
EQ	Systems containing mechanical equipment associated with electrical equipment required to be environmentally qualified to comply with 10 CFR 50.49.
PTS	System relied upon to prevent potential failure of the reactor vessel from pressurized thermal shock, as required for compliance with 10 CFR 50.61. As described in **Section 2.1.1.4.3**, the Reactor Coolant System is the only system within license renewal scope for PTS.
ATWS	Systems that interface with the ATWS Mitigation System Actuation Circuitry and perform functions required to comply with 10 CFR 50.62.
SBO	Systems and functions required to cope with and restore power after a Station Blackout, in compliance with 10 CFR 50.63.

**Table 2.2-2:
STRUCTURAL SCOPING RESULTS**

Listing of Plant Structures	Structures in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
Auxiliary Building [includes Refueling Water Storage Tank & Reactor Makeup Water Storage Tank foundations & West Penetration Access Area (WPAA), all of which are part of the Auxiliary Building structure]	Yes	NSR, FP, SBO	2.4.2.1
Condensate Storage Tank (CST) Foundation - Yard structure	Yes	NSR, FP, SBO	2.4.2.8.1
Control Building	Yes	NSR, FP, SBO	2.4.2.2
Diesel Generator Building	Yes	NSR, FP, SBO	2.4.2.3
Electrical ManHole MH-2 - Yard structure	Yes	NSR, FP	2.4.2.8.3
Electrical Substation (230 KV substation 2000 amp bus 1 power circuit breaker)	Yes	SBO	2.4.2.8.5
Fire Service Pumphouse - Yard structure	Yes	FP	2.4.2.8.2
Fuel Handling Building	Yes	NSR, FP	2.4.2.4
Hot Machine Shop	Yes	NNS	2.4.2.1
Intermediate Building [includes East Penetration Access Area (EPAA), which is part of the Intermediate Building structure]	Yes	NSR, FP, SBO	2.4.2.5
North Berm - Yard structure	Yes	NNS	2.4.2.8.4
Reactor Building (includes interior structures)	Yes	NSR, FP, SBO	2.4.1

**Table 2.2-2:
STRUCTURAL SCOPING RESULTS**

Listing of Plant Structures	Structures in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
Service Water Pond Dams (North Dam, South Dam, East Dam) and West Embankment - Yard structures	Yes	NSR	2.4.2.8.4
Service Water Discharge Structure	Yes	NSR	2.4.2.7.2
Service Water Intake Structure	Yes	NSR	2.4.2.7.2
Service Water Pumphouse	Yes	NSR, FP	2.4.2.7.1
Transformer Areas (South of Turbine Building) Foundation	Yes	SBO	2.4.2.8.5
Transmission Towers & Foundation (Emergency Auxiliary Transformers to 230 KV substation 2000 amp bus 1 power circuit breaker)	Yes	SBO	2.4.2.8.5
Turbine Building	Yes	NNS, FP	2.4.2.6
Auxiliary Fire Pump House (Adjacent to Filtered Water Storage Tank)	No		
Auxiliary Boiler House	No		
Auxiliary Boiler Fuel Oil Storage Tank and Berm	No		
Auxiliary Service Building	No		
Circulating Water Discharge Structure	No		
Circulating Water Discharge Canal	No		

Table 2.2-2:
STRUCTURAL SCOPING RESULTS

Listing of Plant Structures	Structures in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
Circulating Water Intake Structure	No		
Civil / Welding Shop	No		
Civil Warehouse (South of Civil Shop)	No		
Closed Cycle Cooling Tower and Pumphouse	No		
Containment Access Building (CAB)	No		
Contaminated Tool Warehouse	No		
Craft Training Center (CTC)	No		
CRDM Industrial Cooler Skid	No		
Compressed Gas Tanks Storage Racks (Southeast Section of Protected Area)	No		
Containment Access Runway (CAR)	No		
Demineralized Water Tank Foundation	No		
Demineralized Water Pumphouse	No		
Electrical Sheds (1,2) - Inside Protected Area	No		
Electrical Sheds (3-7) - Outside Protected Area	No		
Electrical Shop	No		

Table 2.2-2:
STRUCTURAL SCOPING RESULTS

Listing of Plant Structures	Structures in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
Facility Maintenance Shop	No		
Facility Storage Building (East of Electrical Shop)	No		
Filtered Water Storage Tank Foundation	No		
Fitness & Wellness Center	No		
Garage & Maintenance Facility	No		
Hydrogen Tanks Storage Rack (North of Radiological Maintenance Building)	No		
Hydrogen / Carbon Dioxide Tanks Storage Rack (South of Turbine Building)	No		
Industrial Waste Facility (Biological Treatment Ponds)	No		
Jetty (Monticello Reservoir)	No		
Lighting Masts (Plant)	No		
Maintenance Shed	No		
Mechanical Maintenance Shop	No		
Metal Shop	No		
Meteorological Relay Shed	No		
Meteorological Tower	No		
Monticello Reservoir Dams (A, B, C, D)	No		
NDE Radiography Lab	No		

Table 2.2-2:
STRUCTURAL SCOPING RESULTS

Listing of Plant Structures	Structures in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
Nitrogen Tank Storage Area (North of Radiological Maintenance Building)	No		
Nuclear Operations Building (NOB)	No		
Oxygen Tanks Storage Rack (North of Fuel Handling Building)	No		
Paint Shop	No		
Paint Storage Building	No		
Pipe Shop	No		
Potable Water Pumphouse	No		
Radiological Maintenance Building	No		
Railroad Spur Lines	No		
Respiratory Protection Facility	No		
Sand Blast Sheds (2)	No		
Sanitary Waste Facility (Biological Treatment Ponds)	No		
Screen Wash Pumphouse	No		
Security Building / Access Portal	No		
Service Building	No		
Site Collection Facility (Fitness-For-Duty)	No		
Sodium Hydroxide Storage Tank Foundation (North of Water Treatment Building)	No		

Table 2.2-2:
STRUCTURAL SCOPING RESULTS

Listing of Plant Structures	Structures in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
Steam Generator (Old) Recycle Facility (South of Plant Site)	No		
Storm Drainage / Catch Basins	No		
Telephone Switching Station Shed	No		
Time Clock Alley Shed	No		
Turbine Lube Oil Storage Tanks (Buried)	No		
Vehicle Access Portal (South Gate)	No		
Vehicle Barrier & Protection Fences	No		
Warehouses (A, B, E, F) - Outside Protected Area	No		
Warehouses (C, D) - Inside Protected Area	No		
Waste Neutralization Basin (North of Water Treatment Building)	No		
Waste Oil Incinerator Facility (Southwest of Plant Site)	No		
Water Treatment Building	No		

Table 2.2-2 Notation

NSR	Safety-related structures and structural components which are those relied upon to remain functional during and following design-basis events to ensure the intended functional requirements of 10 CFR 54.4(a)(1)(i), (ii), or (iii) are met.
NNS	All nonsafety-related structures and structural components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs 10 CFR 54.4(a)(1)(i), (ii), or (iii).
FP	Structures and Structural Components supporting systems and functions required to assure a safe shutdown following a fire, either Appendix A or Appendix R, in compliance with 10 CFR 50.48. The Fire Protection Evaluation Report (FPER) [**Reference 2.2-5**] identifies the functions required for fire protection.
EQ	Structures and Structural Components supporting systems containing mechanical equipment associated with electrical equipment required to be environmentally qualified to comply with 10 CFR 50.49.
PTS	Structures and Structural Components supporting systems relied upon to prevent potential failure of the reactor vessel from pressurized thermal shock, as required for compliance with 10 CFR 50.61. As described in **Section 2.1.1.4.3**, the Reactor Coolant System is the only system within license renewal scope for PTS.
ATWS	Structures and Structural Components supporting systems that interface with the ATWS Mitigation System Actuation Circuitry and perform functions required to comply with 10 CFR 50.62.
SBO	Structures and structural components relied upon, in safety analysis or plant evaluations, to perform a function that demonstrates compliance with the Commission’s regulation for Station Blackout (10 CFR 50.63).

Table 2.2-3:
ELECTRICAL SCOPING RESULTS

Listing of Plant Electrical Systems	Electrical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
AA - Annunciators	No		
BP - Balance of Plant I&C	Yes	NSR, FP, EQ, SBO	2.5
CE - Control Room Evacuation Panel	Yes	NSR, FP, SBO	2.5
CP - Computer	No		
CR - Rod Control & Position System	Yes	NSR	2.5
DG - Diesel Generator Services	Yes	NSR, NNS, FP, SBO	2.5
EC - Grounding & Cathodic Protection	No		
ED - DC Distribution	Yes	NSR, NNS, FP, EQ, SBO	2.5
EE - Communications	No		
EG - Generator & Main Transformer	No		
EH - Turbine Electro-Hydraulic Control	Yes	ATWS	2.5
EI - Earthquake Instrumentation	No		
EM - Miscellaneous AC Distribution	Yes	NSR, FP, EQ, ATWS, SBO	2.5
EP - Emergency Power	No		
ES - Electrical System	Yes	NSR, NNS, FP, EQ, SBO	2.5
ET - Heat Tracing	Yes	NSR, NNS	2.5

Table 2.2-3:
ELECTRICAL SCOPING RESULTS

Listing of Plant Electrical Systems	Electrical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
EV - AC Vital Buses (120 Volt Distribution)	Yes	NSR, NNS, FP, EQ, SBO	2.5
EW - Warehouse Electrical Equipment	Yes	FP	2.5
IC - Incore Instrumentation	Yes	NSR, EQ	2.5
IE - Inhouse Electrical Maintenance	No		
IN - Integrated Control	No		
LF - Low Frequency Grounding	No		
MC - Main Control Board	Yes	NSR, FP, SBO	2.5
MI - Misc. Instrumentation (Pressure Flow Monitor System only)	Yes	NSR, FP, EQ	2.5
MI - Misc. Instrumentation (Incore Neutron Flux Monitoring System	No		
MI - Misc. Instrumentation (Incore Temperature Monitoring System	Yes	NSR, EQ	2.5
MI - Misc. Instrumentation (Core Cooling Monitoring System)	No		
MI - Misc. Instrumentation (Reactor Vessel Level Indication System)	Yes	NSR, EQ	2.5
MI - Misc. Instrumentation (Digital Metal Impact Monitoring System)	No		

Table 2.2-3:
ELECTRICAL SCOPING RESULTS

Listing of Plant Electrical Systems	Electrical Systems in License Renewal Scope	Reasons for In License Renewal Scope	Screening Results Application Section
NI - Nuclear Instrumentation	Yes	NSR, FP, EQ, SBO	2.5
OW - Off-Site Warning System	No		
PS - Plant Surveillance	Yes	FP, SBO	2.5
RM - Radiation Monitoring	Yes	NSR, NNS, EQ	2.5
RP - Reactor Protection Control System	Yes	NSR, FP, EQ, SBO	2.5
SG - Engineered Safety Features	Yes	NSR, FP, ATWS, SBO	2.5
TS - Substation	Yes	SBO	2.5
TX - Technical Support Center	Yes	NSR, FP	2.5
WE - Welding Receptacles	No		
XE - Excitation System	No		
XI - Process Instrumentation/ Control	Yes	NSR, FP, SBO	2.5

Table 2.2-3 Notation:

NSR Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the intended functional requirements of 10 CFR 54.4(a)(1)(i), (ii), or (iii) are met.

NNS All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs 10 CFR 54.4(a)(1)(i), (ii), or (iii).

FP Systems and component functions required to assure a safe shutdown following a fire, either Appendix A or Appendix R, in compliance with 10 CFR 50.48. The Fire Protection Evaluation Report (FPER) [**Reference 2.2-5**] identifies the functions required for fire protection.

EQ Systems containing components associated with electrical equipment required to be environmentally qualified to comply with 10 CFR 50.49.

PTS System relied upon to prevent potential failure of the reactor vessel from pressurized thermal shock, as required for compliance with 10 CFR 50.61. As described in **Section 2.1.1.4.3**, the Reactor Coolant System is the only system within license renewal scope for PTS.

ATWS Systems and components that interface with the ATWS Mitigation System Actuation Circuitry and perform functions required to comply with 10 CFR 50.62.

SBO Systems and components relied upon, in safety analysis or plant evaluations, to perform a function that demonstrates compliance with the Commission's regulation for Station Blackout (10 CFR 50.63).

2.3 SYSTEM SCOPING AND SCREENING RESULTS: MECHANICAL

The determination of the mechanical systems and components within the scope of license renewal is made by initially identifying VCSNS mechanical systems and then reviewing them to determine which systems and components satisfy one or more of the criteria in 10 CFR 54.4 [**Reference 2.3-1**]. The scoping and screening process is described in **Section 2.1** and the results of the mechanical systems scoping review are contained in **Section 2.2**.

The screening results are provided below in four sections:

- Reactor Coolant System (**2.3.1**)
- Engineered Safety Features (**2.3.2**)
- Auxiliary Systems (**2.3.3**)
- Steam and Power Conversion Systems (**2.3.4**)

2.3.1 REACTOR VESSEL, INTERNALS AND REACTOR COOLANT SYSTEM

This section of the Application applies to the Reactor Coolant System and ancillary piping from adjoining systems that form the Reactor Coolant pressure boundary.

The Reactor Coolant System non-Class 1 components subject to aging management review include the following:

- Instrumentation tubing downstream of flow restrictors
- Piping connecting to the ECCS check valve testing system downstream of Class 1 boundary valves
- Test connection piping associated with the Safety Injection System piping downstream of Class 1 boundary valves
- Charging System and Letdown piping outside of the Class 1 boundary valves
- Vent and drain piping connecting to the Charging System Reactor Coolant Pump seal cooling piping
- Discharge piping from relief valves and pressure control valves
- Valve leakoff piping
- Sampling system piping downstream of Class 1 boundary valves
- Reactor Coolant Pump oil collection system (evaluated as an Auxiliary System)

The Reactor Coolant System Class 1 components subject to aging management review include the following:

- Reactor Coolant Piping, Valves and Pumps
- Reactor Vessel

- Reactor Vessel Internals
- Incore Instrumentation System
- Pressurizer
- Steam Generators

2.3.1.1 Reactor Coolant System Description

The Reactor Coolant System (RCS) consists of three 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, pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control. All of the above components are located in the Reactor Building.

During operation, the RCS transfers 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 RCS at a flowrate and temperature consistent with achieving the required reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber (boron) used in chemical shim control.

The RCS 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 plant.

The license renewal evaluation boundaries for the Reactor Coolant System are depicted on the following drawings:

E-302-601	**Reactor Coolant System**
E-302-602	**Reactor Coolant System**
D-302-606	**Reactor Coolant Pumps Oil Collection System**
E-302-641	**Residual Heat Removal**
E-302-671	**Chemical and Volume Control**
E-302-672	**Chemical and Volume Control**
E-302-673	**Chemical and Volume Control**
E-302-691	**Safety Injection**
E-302-692	**Safety Injection**

E-302-693 **Safety Injection**

D-302-812 **ECCS Check Valve Testing**

FSAR Section 5.0, Reactor Coolant System, provides additional information concerning the Reactor Coolant System. The non-class 1 mechanical component types and component intended functions for the Reactor Coolant System are listed in Table 2.3-1
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Table 2.3-1:
REACTOR COOLANT SYSTEM NON-CLASS 1 COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Drip Pan, RCP Oil Collection	Fire Protection	Table 3.3-2 Item 1, Table 3.3-1 Item 6
Flame Arrester, RCP Oil Collection	Fire Protection	Table 3.3-2 Item 1, Table 3.3-1 Item 6
Flexible Hose, RCP Oil Collection	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-1 Item 6
Heat Exchanger, RCP Oil Cooler Tubes	Pressure Boundary	Table 3.2-2 Item 4, Table 3.3-2 Item 28
Heat Exchanger, RCP Oil Cooler Tubesheet	Pressure Boundary	Table 3.2-2 Item 4, Table 3.3-2 Item 28
Orifices	Pressure Boundary Throttling	Table 3.1-1 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6
Pipe	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6, Table 3.1-2 Item 14, Table 3.1-2 Item15
Pipe, RCP Oil Collection	Pressure Boundary Fire Protection	Table 3.3-1 Item 6, Table 3.3-2 Item 1

**Table 2.3-1:
REACTOR COOLANT SYSTEM NON-CLASS 1
COMPONENT TYPES SUBJECT TO AGING
MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Tanks, RCP Oil Collection Drain Tank	Fire Protection	Table 3.3-1 Item 6, Table 3.3-2 Item 1
Tanks, RCP Oil Cooler Enclosure	Fire Protection	Table 3.3-1 Item 6, Table 3.3-2 Item 1
Tanks, RCP Oil Lift Enclosure	Fire Protection	Table 3.3-1 Item 6, Table 3.3-2 Item 1
Tanks, RCP Oil Fill & Drain Enclosure	Fire Protection	Table 3.3-1 Item 6, Table 3.3-2 Item 1
Tanks, RCP Upper Oil Alarm & Gauge Enclosure	Fire Protection	Table 3.3-1 Item 6, Table 3.3-2 Item 1
Flange - Thermal Barrier Piping	Pressure Boundary	Table 3.2-1 Item 1, Table 3.3-1 Item 14
Tube and Tube Fittings	Pressure Boundary	Table 3.1-2 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6, Table 3.1-2 Item 12
Valves (Body Only)	Pressure Boundary	Table 3.1-2 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6, Table 3.1-2 Item 14, Table 3.1-2 Item 15
Valves, RCP Oil Collection (Body Only)	Pressure Boundary	Table 3.3-1 Item 6, Table 3.3-2 Item 1

2.3.1.2 Piping, Valves And Pumps

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

- Primary loop piping interconnecting the reactor vessel, steam generators and reactor coolant pumps.

- The piping (including fittings, branch connections, safe ends, thermal sleeves, flow restrictors and thermowells) and valves leading to connecting auxiliary or support systems, up to and including the second isolation valve (from the high pressure side) on each line.
- Pressure boundary portion of Class 1 valves (body, bonnet and bolting).
- Pressure boundary portion of the reactor coolant pumps (casing, main closure flange, thermal barrier heat exchanger and bolting).

The primary loop piping consists of three closed reactor coolant loops interconnecting the reactor vessel, steam generators and reactor coolant pumps.

Class 1 branch piping consists of piping connected to the Class 1 primary loop piping out to and including the outermost containment isolation valve in piping which penetrates primary containment, or 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 lines are equipped with 3/8 inch inside diameter flow restrictors. These flow restrictors limit the maximum flow from a break downstream of the flow restrictor to below the makeup capability of the charging system.

The pressure retaining portion of the Class 1 valves includes the body, bonnet and bolting. The valves are welded into the piping, except for the pressurizer relief and pressurizer code safety valves, which have flanged connections.

The portions of the reactor coolant pumps that perform a pressure boundary function are the pump casing, main closure flange, thermal barrier heat exchanger and bolting. The reactor coolant pumps are vertical, single stage, centrifugal pumps, equipped with controlled leakage shaft seals. The shaft seals are excluded from aging management review because they are periodically replaced. Preventive maintenance for the reactor coolant pump seals is currently scheduled for one pump each outage, with the seals for each pump being maintained every third outage.

The Class 1 portion of the Reactor Coolant System includes portions of the Chemical and Volume Control System, Emergency Core Cooling System, Residual Heat Removal System and Safety Injection System.

FSAR Section 5.5, Component and Subsystem Design, provides additional information concerning Class 1 piping and associated pressure boundary components. The mechanical component types and component intended functions for the RCS Class 1 piping and associated pressure boundary components are listed in Table 2.3-2.

Table 2.3-2:
REACTOR COOLANT SYSTEM CLASS 1 PIPING, VALVE, & PUMP COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Bolting Materials (<2" diameter)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 22, Table 3.1-1 Item 26
Elbows and Nozzles - Reactor Coolant Loop	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 10, Table 3.1-2 Item 1, Table 3.1-2 Item 5
Orifices - Piping	Pressure Boundary Throttling	Table 3.1-1 Item 6, Table 3.1-2 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6
Piping and Fittings - less than NPS 4"	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 6, Table 3.1-2 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6
Piping and Fittings - Reactor Coolant System	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 5
Valves - body and/or bonnet (Cast)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 19, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 5
Valves - body and/or bonnet (Forged)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 5

Table 2.3-2:
REACTOR COOLANT SYSTEM CLASS 1 PIPING, VALVE, & PUMP COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Bolting Materials - RCP Main Flange	Pressure Boundary	Table 3.1-1 Item 22, Table 3.1-1 Item 26
Flange - RCP Thermal Barrier	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-2 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6
Main Closure Flange - RCP	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 19, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 7
Piping/Tubing - RCP Thermal Barrier (less than NPS 4 inch)	Pressure Boundary	Table 3.1-1 Item 6, Table 3.1-2 Item 1, Table 3.1-2 Item 5, Table 3.1-2 Item 6, Table 3.3-1 Item 14
Pump Casing - RCP	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 19, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 7

2.3.1.3 Reactor Vessel

The reactor vessel, fabricated by Chicago Bridge and Iron, Inc., is cylindrical, with a welded hemispherical bottom head and a removable, bolted, flanged, and gasketed, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains adapters for connecting the control rod drive mechanisms and instrumentation. Inlet and outlet nozzles are located symmetrically around the vessel. The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore instrumentation.

FSAR Section 5.4, Reactor Vessel and Appurtenances, provides additional information concerning the reactor vessel. The mechanical component types and component intended functions for the reactor vessel are listed in Table 2.3-3.

**Table 2.3-3:
REACTOR VESSEL
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
CRDM Latch Housing	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 7
CRDM Rod Travel Housing	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 7
CRDM Top Cap/Vent Plug	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 7
RV Bottom Head Dome	Pressure Boundary	Table 3.1-1 Item 26
RV Bottom Head Dome Cladding	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 7
RV Bottom Head Penetration Tubes	Pressure Boundary	Table 3.1-1 Item 9, Table 3.1-2 Item 2, Table 3.1-2 Item 7
RV Closure Head & Vessel Flanges	Pressure Boundary	Table 3.1-1 Item 26, Table 3.1-1 Item 28
RV Closure Head & Vessel Flanges Cladding	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 7
RV Closure Head Dome Cladding	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 7

**Table 2.3-3:
REACTOR VESSEL
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
RV Closure Head Dome & Lifting Lugs	Pressure Boundary	Table 3.1-1 Item 26
RV Closure Head Penetration Safe Ends	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 7
RV Closure Head Penetration Tubes	Pressure Boundary	Table 3.1-1 Item 23, Table 3.1-2 Item 2, Table 3.1-2 Item 7, Table 3.1-2 Item 11
RV Closure Stud Assembly (Including Nuts & Washers)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 18, Table 3.1-1 Item 26, Table 3.1-1 Item 28, Table 3.1-1 Item 34
RV Core Support Pads (Clevises/ Keyways)	Pressure Boundary	Table 3.1-1 Item 9, Table 3.1-2 Item 7
RV Inlet & Outlet Nozzles (Including Cladding)	Pressure Boundary	Table 3.1-1 Item1, Table 3.1-1 Item 3, Table 3.1-1 Item 4, Table 3.1-2 Item 7
RV Inlet & Outlet Nozzles (Including Nozzle Support Pads)	Pressure Boundary	Table 3.1-1 Item 26
RV Inlet & Outlet Nozzle Safe Ends	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 2, Table 3.1-2 Item 7
RV Shell, Core Region	Pressure Boundary	Table 3.1-1 Item 26

Table 2.3-3:
REACTOR VESSEL
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
RV Shell, Core Region (Including Cladding)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 3, Table 3.1-1 Item 4, Table 3.1-1 Item 7, Table 3.1-2 Item 7
RV Shell, Nozzle Course (Including Cladding)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 3, Table 3.1-1 Item 4, Table 3.1-2 Item 7
RV Shell, Nozzle Course & Refueling Seal Ledge	Pressure Boundary	Table 3.1-1 Item 26
RV Ventilation Shroud Support Ring	Pressure Boundary	Table 3.1-1 Item 26

2.3.1.4 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 incore instrumentation support structure. The reactor 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, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the incore instrumentation.

The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and into a plenum at the bottom of the vessel. The coolant then reverses direction and flows up through the core support and lower core plates. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

FSAR Section 4.2.2, Reactor Vessel Internals, provides additional information concerning the reactor vessel internals. The mechanical component types and component intended functions for the reactor vessel internals are listed in Table 2.3-4.

Table 2.3-4:
REACTOR VESSEL INTERNALS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Baffle and Former Assembly	Core Support & Orientation, Flow Distribution, Reactor Vessel Shielding	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Baffle and Former Assembly Bolts	Core Support & Orientation, Flow Distribution, Reactor Vessel Shielding	Table 3.1-1 Item 1, Table 3.1-1 Item 5, Table 3.1-1 Item 8, Table 3.1-1 Item 12, Table 3.1-1 Item 13, Table 3.1-2 Item 7
Bottom Mounted Instrumentation (BMI) Columns	In-Core Guidance & Protection	Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Clevis Inserts	Core Support & Orientation	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 28, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Clevis Inserts Bolts	Core Support & Orientation	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 30, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Core Barrel and Flange	Core Support & Orientation, Flow Distribution, Reactor Vessel Shielding	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7

Table 2.3-4:
REACTOR VESSEL INTERNALS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Core Barrel Outlet Nozzle	Flow Distribution	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Guide Tube	Control Rod Guidance & Protection	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Guide Tube Bolts and Support Pins (Split Pins)	Control Rod Guidance & Protection	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Head and Vessel Alignment Pins	Control Rod Guidance & Protection	Table 3.1-1 Item 8, Table 3.1-1 Item 28, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Holdown Spring	Core Support & Orientation	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 30, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Lower Core Plate	Core Support & Orientation, Flow Distribution, In-Core Guidance & Protection, Secondary Core Support	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Lower Core Plate Fuel Alignment Pins	Core Support & Orientation, Flow Distribution, In-Core Guidance & Protection, Secondary Core Support	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7

Table 2.3-4:
REACTOR VESSEL INTERNALS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Lower Support Columns	Core Support & Orientation, In-Core Guidance & Protection, Secondary Core Support	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Lower Support Columns Bolts	Core Support & Orientation, In-Core Guidance & Protection, Secondary Core Support	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-1 Item 35, Table 3.1-2 Item 7
Lower Support Plate	Core Support & Orientation, Flow Distribution, In-Core Guidance & Protection, Secondary Core Support	Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Neutron Panels	Reactor Vessel Shielding	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 31, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Radial Keys	Core Support & Orientation	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 28, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Secondary Core Support	Core Support & Orientation, Flow Distribution, In-Core Guidance & Protection, Secondary Core Support	Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7

Table 2.3-4:
REACTOR VESSEL INTERNALS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Spray Nozzle	Flow Distribution	Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Upper Core Plate	Core Support & Orientation, Flow Distribution	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Upper Core Plate Alignment Pin	Core Support & Orientation, Flow Distribution	Table 3.1-1 Item 8, Table 3.1-1 Item 28, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Upper Core Plate Fuel Alignment Pins	Core Support & Orientation, Flow Distribution	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Upper Instrumentation Conduit and Supports	In-Core Guidance & Protection	Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Upper Support Column	Control Rod Guidance & Protection, In-Core Guidance & Protection	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7
Upper Support Column Bolts	Control Rod Guidance & Protection, In-Core Guidance & Protection	Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7, Table 3.1-1 Item 35
Upper Support Plate Assembly	Control Rod Guidance & Protection	Table 3.1-1 Item 1, Table 3.1-1 Item 8, Table 3.1-1 Item 33, Table 3.1-2 Item 7

2.3.1.5 Incore Instrumentation System

The Incore Instrumentation System is comprised of thermocouples, positioned to measure fuel assembly coolant outlet temperatures at pre-selected positions, and fission chamber detectors positioned in guide thimbles, which run the length of selected fuel assemblies, to measure neutron flux distribution.

FSAR Section 4.4.5.1, Incore Instrumentation, provides additional information concerning the Incore Instrumentation System.

The license renewal evaluation boundaries for the Incore Instrumentation System are depicted on the following drawing:

1MS-44-014	**3-Loop Plant Bottom Mounted Inst. Standard Layout**

A complete list of Incore Instrumentation System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-5.

Table 2.3-5:
INCORE INSTRUMENTATION SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Incore Neutron Detector (Flux) Thimbles	In-Core Guidance & Protection, Pressure Boundary	**Table 3.1-1 Item 28, Table 3.1-2 Item 1, Table 3.1-2 Item 7, Table 3.1-2 Item 11**
Incore Thermocouples (Pressure Retaining Only)	In-Core Guidance & Protection, Pressure Boundary	**Table 3.1-2 Item 4, Table 3.1-2 Item 5, Table 3.1-2 Item 6**
Tube and Tube Fittings, Incore Neutron Detector Conduits	Pressure Boundary Support	**Table 3.1-2 Item 1, Table 3.1-2 Item 7, Table 3.1-2 Item 11**

2.3.1.6 Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. A pressurizer surge line connects the pressurizer to one of the hot legs in the Reactor Coolant System (RCS). The line enables continuous coolant volume pressure adjustments between the RCS and the pressurizer. The surge line nozzle and removable electric heaters are installed in the bottom head of the vessel, while spray line nozzles and relief and safety valve connections are located in the top head.

FSAR Section 5.5.10, Pressurizer, provides additional information concerning the pressurizer. The mechanical component types and component intended functions for the pressurizer are listed in Table 2.3-6.

Table 2.3-6:
PRESSURIZER
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Tank, PZR Head (Lower)	Pressure Boundary	**Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-1 Item 26, Table 3.1-1 Item 29, Table 3.1-2 Item 7**
Tank, PZR Head (Upper)	Pressure Boundary	**Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-1 Item 26, Table 3.1-2 Item 7**
Tank, PZR Immersion Heater Well Assemblies	Pressure Boundary	**Table 3.1-1 Item 1, Table 3.1-2 Item 1, Table 3.1-2 Item 7, Table 3.1-2 Item 11**
Tank, PZR Manway Cover	Pressure Boundary	**Table 3.1-2 Item 7, Table 3.1-2 Item 11**
Tank, PZR Manway Cover (& Bolts)	Pressure Boundary	**Table 3.1-1 Item 26**
Tank, PZR Manway Cover Bolts/ Studs	Pressure Boundary	**Table 3.1-1 Item 22**

Table 2.3-6:
PRESSURIZER
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Tank, PZR Nozzle & Manway Forgings	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-1 Item 26, Table 3.1-2 Item 7
Tank, PZR Nozzle Safe Ends	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 1, Table 3.1-2 Item 7
Tank, PZR Nozzle Thermal Sleeves	Heat Transfer	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-2 Item 7
Tank, PZR Nozzle Safe End Weld Metal	Pressure Boundary	Table 3.1-1 Item 11, Table 3.1-2 Item 2, Table 3.1-2 Item 7
Tank, PZR Shell Barrel	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 24, Table 3.1-1 Item 26, Table 3.1-2 Item 7
Tank, PZR Tube Couplings (Instrument and Sample Lines)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-2 Item 1, Table 3.1-2 Item 7, Table 3.1-2 Item 11
Tank, PZR Tubing (Instrument and Sample Lines)	Pressure Boundary	Table 3.1-2 Item 1, Table 3.1-2 Item 7, Table 3.1-2 Item 11

2.3.1.7 Steam Generators

The steam generators installed at VCSNS are Westinghouse Delta-75, feedring type steam generators. The steam generators are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The feedwater enters at approximately two-thirds (2/3) of the steam generator height and is distributed equally around the circumference of the steam generator shell. Feedwater enters the tube bundle by flowing downward between the steam generator external shell and inner wrapper barrel. An open area at the bottom of the wrapper barrel permits the feedwater to enter the tube bundle. Steam is generated and flows upward through the moisture separators and flow restrictor outlet nozzle at the top of the steam drum. High efficiency centrifugal steam separators remove most of the entrained water. Dryers are employed to increase the steam quality to a minimum of 99.90% (0.10% moisture).

FSAR Section 5.5.2, Steam Generators, provides additional information concerning the steam generators. The mechanical component types and component intended functions for the steam generators are listed in Table 2.3-7.

**Table 2.3-7:
STEAM GENERATORS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Channel Head and Inlet/Outlet Nozzles	Pressure Boundary	**Table 3.1-1 Item 1, Table 3.1-1 Item 26, Table 3.1-1 Item 32, Table 3.1-2 Item 7**
Channel Head Divider Plate	Pressure Boundary	**Table 3.1-1 Item 32, Table 3.1-2 Item 7**
Closure Bolting (Primary and Secondary Side)	Pressure Boundary	**Table 3.1-1 Item 22, Table 3.1-1 Item 26**
External Support Attachments, Lifting Lugs, Support Pads, Barrel Trunnions	Pressure Boundary	**Table 3.1-1 Item 26**
Feedwater Distribution Ring	Pressure Boundary Throttling	**Table 3.1-2 Item 10**
Manway Covers (Primary Side)	Pressure Boundary	**Table 3.1-1 Item 26**
Manway Covers Insert Plates (Primary Side)	Pressure Boundary	**Table 3.1-2 Item 7, Table 3.1-2 Item 11**

**Table 2.3-7:
STEAM GENERATORS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Manway, Handhole, Sludge Collector Opening & Inspection Port Covers (Secondary Side)	Pressure Boundary	Table 3.1-1 Item 26, Table 3.1-2 Item 3
Manways, Handholes, Sludge Collector Opening & Inspection Ports (Secondary Side)	Pressure Boundary	Table 3.1-1 Item 26, Table 3.1-2 Item 3
Nozzle Safe Ends (Primary Side)	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 32, Table 3.1-2 Item 2, Table 3.1-2 Item 7
Nozzle Thermal Sleeve, Secondary Side Feedwater	Heat Transfer	Table 3.1-2 Item 8, Table 3.1-2 Item 9
Nozzle Thermal Sleeve, Secondary Side Emergency Feedwater	Heat Transfer	Table 3.1-2 Item 8, Table 3.1-2 Item 9
Nozzle, Secondary Side Feedwater	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-2 Item 3
Nozzle, Secondary Side Emergency Feedwater	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-2 Item 3
Nozzles, Secondary Side Emergency Feedwater/Feedwater	Pressure Boundary	Table 3.1-1 Item 26
Nozzles, Inlet/Outlet, Closure Ring & Weld Metal	Pressure Boundary	Table 3.1-1 Item 32, Table 3.1-2 Item 7
Shell, Upper and Lower Barrel, Transition Cone, Elliptical Head	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 2, Table 3.1-1 Item 26
Steam Outlet Nozzle	Pressure Boundary	Table 3.1-1 Item 1, Table 3.1-1 Item 26
Steam Outlet Nozzle Flow Limiter	Pressure Boundary Throttling	Table 3.1-2 Item 8, Table 3.1-2 Item 9

**Table 2.3-7:
STEAM GENERATORS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Taps, Blowdown, Drain, Level, Sample, Wet Layup (Plugged)	Pressure Boundary	Table 3.1-1 Item 26, Table 3.1-2 Item 3
Tube Bundle Wrapper, Wrapper Support and Downcomer (Secondary Side)	Support	Table 3.1-2 Item 3, Table 3.1-2 Item 13
Tube Support Plates, Plates, AVBs, Flow Distribution Baffle (Secondary Side)	Support	Table 3.1-2 Item 13
Tubes (Secondary Side)	Pressure Boundary Heat Transfer	Table 3.1-1 Item 15
Tubes/Plugs (Primary Side)	Pressure Boundary Heat Transfer (Tubes only)	Table 3.1-1 Item 1, Table 3.1-1 Item 15
Tubeplate (Primary Side)	Pressure Boundary	Table 3.1-1 Item 32, Table 3.1-2 Item 7
Tubeplate (Secondary Side)	Pressure Boundary	Table 3.1-2 Item 3
Tubeplate	Pressure Boundary	Table 3.1-1 Item 26

2.3.2 ENGINEERED SAFETY FEATURES

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

Engineered Safety Features, as described in **FSAR Section 6.0**, are those systems and components designed to function under accident conditions to:

- Preclude the release of fission products from the fuel,
- Retain fission products in the Reactor Coolant System,
- Retain fission products within the containment for operational and accidental releases beyond the Reactor Coolant System boundary, and
- Limit or optimize fission product dispersal to minimize population exposure for an accidental release beyond the containment.

The following mechanical systems that make up the Engineered Safety Features are described in the section indicated:

- Chemical and Volume Control System (**Section 2.3.2.1**)
- Containment Isolation System (**Section 2.3.2.2**)
- Hydrogen Removal System (**Section 2.3.2.3**)
- Reactor Building Spray System (**Section 2.3.2.4**)
- Refueling Water System (**Section 2.3.2.5**)
- Residual Heat Removal System (**Section 2.3.2.6**)
- Safety Injection System (**Section 2.3.2.7**)

2.3.2.1 Chemical And Volume Control System

The Chemical and Volume Control System (CVCS) is designed to provide the following services to the Reactor Coolant System (RCS): (1) maintain required water inventory in the RCS, (2) maintain seal-water injection flow to the reactor coolant pumps, (3) control reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration and makeup, and (4) provide a means for filling, draining and pressure testing the RCS.

The Chemical and Volume Control System also operates in conjunction with the Refueling Water, Residual Heat Removal and Safety Injection Systems to deliver borated emergency core cooling water to the Reactor Coolant System following a loss-of-coolant accident (LOCA). During the injection phase, the centrifugal charging pumps in the CVCS, along with the residual heat removal pumps, draw suction from the Refueling Water Storage Tank and inject borated water directly into the Reactor Coolant System.

The Chemical and Volume Control Vents and Drains System is designed to maintain the safety-related pressure boundaries of the Chemical and Volume Control System.

FSAR Section 6.3, Emergency Core Cooling System, and **FSAR Section 9.3.4**, Chemical and Volume Control System, provide additional information concerning the Chemical and Volume Control System.

The license renewal evaluation boundaries for the Chemical and Volume Control System are depicted on the following drawings:

E-302-602	**Reactor Coolant System**
E-302-671	**Chemical and Volume Control**
E-302-672	**Chemical and Volume Control**
E-302-673	**Chemical and Volume Control**
E-302-674	**Chemical and Volume Control**
E-302-675	**Chemical and Volume Control**
E-302-677	**Chemical and Volume Control**
E-302-751	**Boron Recycle**
1MS-12-004 Sh 2	**Charging/Safety - Injection Pumps**

A complete list of Chemical and Volume Control System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-8

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Table 2.3-8:
CHEMICAL AND VOLUME CONTROL SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Agitators and Mixers	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Demineralizers	Pressure Boundary Filtration	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Filters	Pressure Boundary Filtration	Table 3.2-1 Item 10, Table 3.2-2 Item 1, Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 5, Table 3.3-2 Item 6
Flexible Couplings	Pressure Boundary	Table 3.3-1 Item 13, Table 3.3-2 Item 5
Gearboxes	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 5
Heat Exchangers, Channel Head	Pressure Boundary	Table 3.3-1 Item 8, Table 3.3-2 Item 1
Heat Exchangers, Shell	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 8, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 5
Heat Exchangers, Tubes	Pressure Boundary Heat Transfer	Table 3.3-1 Item 8, Table 3.3-2 Item 5, Table 3.3-2 Item 17, Table 3.3-2 Item 28

Table 2.3-8:
CHEMICAL AND VOLUME CONTROL SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 8, Table 3.3-2 Item 5, Table 3.3-2 Item 17
Instrumentation (Pressure Retaining Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6, Table 3.3-2 Item 23
Oil Reservoir	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 5
Orifices	Pressure Boundary Throttling	Table 3.2-1 Item 5, Table 3.2-1 Item 10, Table 3.2-2 Item 1, Table 3.3-2 Item 1, Table 3.3-2 Item 6
Pipe	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 10, Table 3.2-1 Item 11, Table 3.2-2 Item 1, Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 5, Table 3.3-2 Item 6, Table 3.3-2 Item 17
Pumps, Bearing Housings	Pressure Boundary	Table 3.2-1 Item 11, Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 5

Table 2.3-8:
CHEMICAL AND VOLUME CONTROL SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pumps (Casing Only)	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 10, Table 3.2-1 Item 11, Table 3.2-2 Item 1, Table 3.3-1 Item 4, Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 5
Tanks	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Tube and Tube Fitting	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1, Table 3.2-2 Item 6, Table 3.3-2 Item 1, Table 3.3-2 Item 5, Table 3.3-2 Item 6
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 10, Table 3.2-1 Item 11, Table 3.2-2 Item 1, Table 3.3-1 Item 13, Table 3.3-1 Item 5, Table 3.3-2 Item 1, Table 3.3-2 Item 5, Table 3.3-2 Item 6, Table 3.3-2 Item 17

2.3.2.2 Containment Isolation System

The objective of the containment isolation system is to allow the passage of fluids through the containment boundary under normal and accident conditions, while preserving the integrity of the containment boundary, when required to prevent or limit the escape of fission products as a result of a postulated LOCA.

The containment isolation system is not an independent system. Rather, the system is comprised of specific features included in each piping system that penetrates the Reactor Building. Actuation of containment isolation is accomplished through the Engineered Safety Features Actuation System. **FSAR Section 6.2.4**, Containment Isolation System, provides additional information concerning the containment isolation features of these systems.

Four systems at VCSNS have been identified whose only mechanical license renewal function is to provide containment isolation. These systems are: Auxiliary Coolant (Closed Loop) / CRDM Cooling Water (AC), Demineralized Water - Nuclear Service (DN), RB Leak Rate Testing (LR) and Nitrogen Blanketing (NG).

2.3.2.2.1 Auxiliary Coolant / Control Rod Drive Mechanism (CRDM) Cooling Water System

The Auxiliary Coolant / CRDM Cooling Water System is designed to remove heat from the containment air used to cool the Control Rod Drive Mechanism (CRDM) and dissipate this heat to the atmosphere via the Industrial Cooler.

The license renewal evaluation boundaries for the Auxiliary Coolant / CRDM Cooling Water System are depicted on the following drawing:

D-302-852 **CRDM Cooling Water**

A complete list of Auxiliary Coolant / CRDM Cooling Water System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-9.

**Table 2.3-9:
AUXILIARY COOLANT / CRDM COOLING WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.2-1 Item 2 Table 3.2-1 Item 3 Table 3.2-1 Item 11
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 2 Table 3.2-1 Item 3 Table 3.2-1 Item 11

2.3.2.2.2 Demineralized Water - Nuclear Services

The Demineralized Water - Nuclear Services System is designed to clarify, filter and demineralize raw water from Monticello Reservoir for distribution to the Nuclear Steam Supply System (NSSS), secondary (turbine) cycle and other miscellaneous plant systems.

The license renewal evaluation boundaries for the Demineralized Water - Nuclear Services System are depicted on the following drawing:

D-302-715 **Demineralized Water Nuclear Services**

A complete list of Demineralized Water - Nuclear Services System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-10.

Table 2.3-10:
DEMINERALIZED WATER - NUCLEAR SERVICES SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.2-1 Item 3**
Valves (Body Only)	Pressure Boundary	**Table 3.2-1 Item 3**

2.3.2.2.3 Nitrogen Blanketing System

The Nitrogen Blanketing System is designed to provide pressurized nitrogen to hose connections located inside containment.

The license renewal evaluation boundaries for the Nitrogen Blanketing System are depicted on the following drawing:

D-302-311 **Nitrogen Blanketing**

A complete list of Nitrogen Blanketing System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-11.

Table 2.3-11:
NITROGEN BLANKETING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.2-1 Item 2, Table 3.2-1 Item 3, Table 3.2-1 Item 11**

Table 2.3-11:
NITROGEN BLANKETING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 2, Table 3.2-1 Item 3, Table 3.2-1 Item 11

2.3.2.2.4 Reactor Building Leak Rate Testing System

The Reactor Building Leak Rate Testing System is designed to permit containment leakage testing in accordance with 10 CFR 50, Appendix J.

The license renewal evaluation boundaries for the Reactor Building Leak Rate Testing System are depicted on the following drawing:

D-302-811	Reactor Building Leak Rate Testing System

A complete list of Reactor Building Leak Rate Testing System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-12.

Table 2.3-12:
REACTOR BUILDING LEAK RATE TESTING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.2-1 Item 2, Table 3.2-1 Item 3, Table 3.2-1 Item 11
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 2, Table 3.2-1 Item 3, Table 3.2-1 Item 11

2.3.2.3 Hydrogen Removal - Post Accident System

The Hydrogen Removal System is designed for control of combustible hydrogen concentrations in the Reactor Building following a loss-of-coolant accident (LOCA). The system uses electric hydrogen recombiners as a primary means of reducing hydrogen concentrations, while a purge system is provided as a backup to the recombiners.

FSAR Section 6.2.5, Combustible Gas Control in Reactor Building, provides additional information concerning the Hydrogen Removal System.

The license renewal evaluation boundaries for the Hydrogen Removal System are depicted on the following drawing:

D-302-861	**Post Accident Hydrogen Removal & Alternate Purge System**

A complete list of Hydrogen Removal System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-13.

**Table 2.3-13:
HYDROGEN REMOVAL SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Hydrogen Recombiners, Electric	Heat Transfer	**Table 3.2-2 Item 1, Table 3.2-2 Item 3, Table 3.2-1 Item 6, Table 3.2-1 Item 11**
Pipe	Pressure Boundary	**Table 3.2-1 Item 2, Table 3.2-1 Item 3, Table 3.2-1 Item 6, Table 3.2-1 Item 11, Table 3.2-2 Item 1**
Tubing	Pressure Boundary	**Table 3.2-1 Item 3, Table 3.2-2 Item 1**

**Table 2.3-13:
HYDROGEN REMOVAL SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 2, Table 3.2-1 Item 3, Table 3.2-1 Item 6, Table 3.2-1 Item 11, Table 3.2-2 Item 1

2.3.2.4 Reactor Building Spray System

The basic functions of the Reactor Building Spray System are to: (1) remove the thermal energy released to containment by a loss-of-coolant accident (LOCA) at a rate sufficient to limit the resulting over-pressurization to a level below the design limit, thereby maintaining containment structural integrity, and (2) to subsequently reduce the over-pressure to a level that minimizes the pressure differential which induces leakage out of containment. An alternate function of the Reactor Building Spray System is to reduce the concentration of air-borne radioactive iodine in the containment atmosphere.

These functions are accomplished by spraying water containing sodium hydroxide into the containment atmosphere to absorb heat, condense steam and remove air-borne radioactive iodine from the steam-air atmosphere.

During normal plant operation, the Reactor Building Spray System is in a standby condition. Operation of the system is automatically initiated following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) by signals from the Engineered Safety Features (ESF) Actuation System, when the Reactor Building pressure increases to the actuation set point.

FSAR Section 6.2.2, Reactor Building Heat Removal Systems, provides additional information concerning the Reactor Building Spray System.

The license renewal evaluation boundaries for the Reactor Building Spray System are depicted on the following drawing:

D-302-661 **Reactor Bldg. Spray System**

A complete list of Reactor Building Spray System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-14.

Table 2.3-14:
REACTOR BUILDING SPRAY SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Orifices	Pressure Boundary Throttling	Table 3.2-1 Item 10, Table 3.2-2 Item 1, Table 3.2-2 Item 5
Pipe	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 10, Table 3.2-1 Item 11, Table 3.2-2 Item 1, Table 3.2-2 Item 5
Pumps (Casing Only)	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1
Spray Nozzles	Pressure Boundary Spray Flow	Table 3.2-2 Item 1
Tank	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-2 Item 2, Table 3.2-2 Item 5
Tube and Tube Fitting	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1, Table 3.2-2 Item 5
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 10, Table 3.2-1 Item 11, Table 3.2-2 Item 1, Table 3.2-2 Item 5

2.3.2.5 Refueling Water System

The primary function of the Refueling Water System is to support refueling operations, refueling water cleanup, spent fuel pool makeup, and other borated water needs associated with plant operations. The Refueling Water System also operates in conjunction with the Chemical and Volume Control, Residual Heat Removal and Safety Injection Systems to deliver borated emergency core cooling water to the Reactor Coolant System following a loss-of-coolant accident (LOCA). During the injection phase, the Refueling Water Storage Tank provides an adequate supply of borated water for the residual heat removal and centrifugal charging pumps for injection directly into the Reactor Coolant System.

FSAR Section 6.3, Emergency Core Cooling System, and **FSAR Section 9.1.3**, Spent Fuel Cooling System, provide additional information concerning the Refueling Water System.

The license renewal evaluation boundaries for the Refueling Water System are depicted on the following drawing:

D-302-651	**Spent Fuel Cooling**

A complete list of Refueling Water System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-15.

Table 2.3-15:
REFUELING WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.2-1 Item 10, Table 3.2-2 Item 1**
Tanks	Pressure Boundary	**Table 3.2-2 Item 1, Table 3.2-2 Item 7**

2.3.2.6 Residual Heat Removal System

The primary function of the Residual Heat Removal System is to remove radioactive decay heat energy from the core, and sensible and pump heat from the Reactor Coolant System during plant cooldown and refueling operations. The Residual Heat Removal System also operates in conjunction with the Chemical and Volume Control, Refueling Water and Safety Injection Systems to deliver borated emergency core cooling water to the Reactor Coolant System following a loss-of-coolant accident (LOCA). The system operation is categorized in two phases, injection and recirculation.

During the injection phase, the residual heat removal pumps, along with the centrifugal charging pumps in the CVCS, draw suction from the Refueling Water Storage Tank and inject borated water directly into the Reactor Coolant System. During the recirculation phase, the residual heat removal pumps draw suction from the containment sump, remove decay heat via the Residual Heat Removal System heat exchangers, and then deliver flow to the charging pumps suction and to the Reactor Coolant System. As during the injection phase, the charging pumps then inject borated water directly into the Reactor Coolant System.

A secondary function of the Residual Heat Removal System is to transfer refueling water between the Refueling Water Storage Tank and the refueling cavity at the beginning and end of refueling operations.

FSAR Section 6.3, Emergency Core Cooling System, provides additional information concerning the Residual Heat Removal System.

The license renewal evaluation boundaries for the Residual Heat Removal System are depicted on the following drawing:

E-302-641 **Residual Heat Removal**

A complete list of Residual Heat Removal System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-16.

Table 2.3-16:
RESIDUAL HEAT REMOVAL SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Channel Head	Pressure Boundary	**Table 3.2-1 Item 9, Table 3.2-2 Item 1**

Table 2.3-16:
RESIDUAL HEAT REMOVAL SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Shell	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 9, Table 3.2-1 Item 11
Heat Exchangers, Tubes	Pressure Boundary Heat Transfer	Table 3.2-1 Item 9
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.2-1 Item 9
Orifices	Pressure Boundary Throttling	Table 3.2-1 Item 10, Table 3.2-2 Item 1
Pipe	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1
Pumps (Casing Only)	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1
Tube and Tube Fitting	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1

2.3.2.7 Safety Injection System

The Safety Injection System operates in conjunction with the Chemical and Volume Control, Refueling Water and Residual Heat Removal Systems to deliver borated emergency core cooling water to the Reactor Coolant System following a loss-of-coolant accident (LOCA). The principal components of the Safety Injection System that provide emergency core cooling immediately following a LOCA are the accumulators (one for each loop), three CVCS charging pumps and two Residual Heat Removal System pumps. The system operation is categorized in two phases, injection and recirculation.

During the injection phase, the residual heat removal pumps, along with the centrifugal charging pumps in the CVCS, draw suction from the Refueling Water Storage Tank and inject borated water directly into the Reactor Coolant System. In addition, the gas pressurized accumulators discharge their borated water contents directly into the Reactor Coolant

System. During the recirculation phase, the residual heat removal pumps draw suction from the containment sump, remove decay heat via the Residual Heat Removal System heat exchangers, and then deliver flow to the charging pumps suction and the Reactor Coolant System. As during the injection phase, the charging pumps then inject borated water directly into the Reactor Coolant System.

FSAR Section 6.3, Emergency Core Cooling System, provides additional information concerning the Safety Injection System.

The license renewal evaluation boundaries for the Safety Injection System are depicted on the following drawings:

E-302-691	**Safety Injection**
E-302-692	**Safety Injection**
E-302-693	**Safety Injection**
D-302-861	**Post Accident Hydrogen Removal & Alternate Purge System**
D-302-812	**ECCS Check Valve Testing**

A complete list of Safety Injection System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-17.

Table 2.3-17:
SAFETY INJECTION SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Orifices	Pressure Boundary Throttling	**Table 3.2-1 Item 10, Table 3.2-2 Item 1**
Pipe	Pressure Boundary	**Table 3.2-1 Item 10, Table 3.2-2 Item 1, Table 3.2-2 Item 2**

Table 2.3-17:
SAFETY INJECTION SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Tanks	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 10, Table 3.2-1 Item 11, Table 3.2-2 Item 1
Tube and Tube Fittings	Pressure Boundary	Table 3.2-1 Item 10, Table 3.2-2 Item 1, Table 3.2-2 Item 6
Valves (Body Only)	Pressure Boundary	Table 3.2-1 Item 6, Table 3.2-1 Item 10, Table 3.2-1 Item 11, Table 3.2-2 Item 1, Table 3.2-2 Item 2, Table 3.2-2 Item 6

2.3.3 AUXILIARY SYSTEMS

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.

The Auxiliary Systems are those systems used to support normal and emergency plant operations. The systems provide cooling, ventilation, sampling and other required functions. The following systems are included in this section:

- Air Handling and Local Ventilation and Cooling Systems (**Section 2.3.3.1**)
- Boron Recycle System (**Section 2.3.3.2**)
- Building Services (**Section 2.3.3.3**)
- Chilled Water System (**Section 2.3.3.4**)
- Circulating Water System (**Section 2.3.3.5**)
- Component Cooling Water System (**Section 2.3.3.6**)
- Diesel Generator Services Systems (**Section 2.3.3.7**)
- Fire Service System (**Section 2.3.3.8**)
- Fuel Handling System (**Section 2.3.3.9**)
- Gaseous Waste Processing System (**Section 2.3.3.10**)
- Industrial Cooler System (**Section 2.3.3.11**)
- Instrument Air Supply System (**Section 2.3.3.12**)
- Leak Detection System (**Section 2.3.3.13**)
- Liquid Waste Processing System (**Section 2.3.3.14**)
- Nuclear & Non-Nuclear Plant Drains (**Section 2.3.3.15**)
- Nuclear Sampling System (**Section 2.3.3.16**)
- Radiation Monitoring System (**Section 2.3.3.17**)
- Reactor Makeup Water Supply System (**Section 2.3.3.18**)
- Roof Drains System (**Section 2.3.3.19**)
- Station Service Air System (**Section 2.3.3.20**)
- Service Water System (**Section 2.3.3.21**)
- Spent Fuel Cooling System (**Section 2.3.3.22**)
- Thermal Regeneration System (**Section 2.3.3.23**)

2.3.3.1 Air Handling And Local Ventilation And Cooling Systems

Since similar materials and environments exist in heating, ventilation, and air conditioning (HVAC) systems, the Air Handling and Local Ventilation and Cooling Systems were evaluated as single groupings of commodities, instead of as localized systems.

FSAR Section 9.4, Air Conditioning, Heating, Cooling, and Ventilation Systems, provides additional information concerning the Air Handling and Local Ventilation and Cooling Systems.

The license renewal evaluation boundaries for the Air Handling System are depicted on the following drawings:

D-806-001	**Radiation Monitoring System Diagram Atmospheric**
D-806-002	**Radiation Monitoring System Diagram Atmospheric**
D-912-102	**RB Cooling System**
D-912-103	**RB Purge Supply & Purge Exhaust Systems**
D-912-105	**RB Refueling Water Surface System**
D-912-120	**Building Service System Flow Diagram Aux Bldg. HEPA EXH SYS Flow Diagram**
D-912-131	**Building Service System Flow Diagram Fuel Handling Charcoal Exhaust System & Air Supply Distribution**
D-912-132	**Building Service System Flow Diagram Auxiliary Building Pump Room Cooling System**
D-912-134	**Building Service System Flow Diagram Diesel Generator Areas**
D-912-136	**Building Service System Flow Diagram Relay & Computer Room Cooling System**
D-912-138	**Building Service System Flow Diagram Battery Rm & Charging Rm., BOP Charger Area Ventilation System**
D-912-139	**Building Service System Flow Diagram CRDM Switchgear Room Cooling System & Water Chiller Area Ventilation System**
D-912-140	**Building Service System Flow Diagram Control Room Normal & Emergency Air Handling System**

D-912-141	**Building Service System Flow Diagram Technical Support Center & Main Control Board Ventilation**
D-912-154	**Building Service System Flow Diagram Computer Rooms & SAS Room Cooling Unit**
D-912-155	**Building Service System Flow Diagram Service Water Intake Screen & Pump House Bldg. Ventilation System**
D-912-158	**Intermediate Building General Ventilation & Pump Area Cooling Systems**

The license renewal evaluation boundaries for the Local Ventilation and Cooling System are depicted on the following drawings:

D-912-132	**Building Service System Flow Diagram Auxiliary Building Pump Room Cooling System**
D-912-157	**Building Service System Flow Diagram Intermediate Building ESF SWGR Rooms & Speed Switch Rooms Cooling Systems**
D-912-158	**Intermediate Building General Ventilation & Pump Area Cooling Systems**

A complete list of Air Handling and Local Ventilation and Cooling Systems mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-18.

Table 2.3-18:
AIR HANDLING AND LOCAL VENTILATION AND COOLING SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Air Handling Units	Pressure Boundary	Table 3.3-1 Item 5 Table 3.3-1 Item 13
Air Plenum	Pressure Boundary	Table 3.3-1 Item 5 Table 3.3-1 Item 13
Cooling Coils, Fins	Heat Transfer	Table 3.3-1 Item 5
Cooling Coils, Headers	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-1 Item 16, Table 3.3-1 Item 24
Cooling Coils, Tubes	Pressure Boundary Heat Transfer	Table 3.3-1 Item 5, Table 3.3-1 Item 16, Table 3.3-1 Item 24, Table 3.3-2 Item 28
Cooling Coils, Tubesheet	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13
Ductwork	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 2, Table 3.3-2 Item 3
Ductwork, Fan and Plenum Housings	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13
Ductwork, Flexible Connections	Pressure Boundary	Table 3.3-1 Item 2
Ductwork, Exhaust Air Relief Heads	Pressure Boundary	Table 3.3-2 Item 3
Expansion Joints, Mechanical	Pressure Boundary	Table 3.3-1 Item 2
Expansion Joints, Mechanical, Expansion Boot	Pressure Boundary	Table 3.3-1 Item 2

Table 2.3-18:
AIR HANDLING AND LOCAL VENTILATION AND COOLING SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Expansion Joints, Mechanical, Retaining Rings	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13
Heating Coils	Pressure Boundary	Table 3.3-2 Item 3, Table 3.3-1 Item 5, Table 3.3-1 Item 13
Pipe	Pressure Boundary	Table 3.3-2 Item 2, Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16
Tube and Tube Fitting	Pressure Boundary	Table 3.3-2 Item 2, Table 3.3-2 Item 9
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 2, Table 3.3-2 Item 3

2.3.3.2 Boron Recycle System

The Boron Recycle System collects 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. The mechanical license renewal function of the Boron Recycle System is to maintain its system boundary with the Component Cooling (CC) and Chemical and Volume Control (CS) systems.

FSAR Section 9.3.6, Boron Recycle System, provides additional information concerning the Boron Recycle System.

The license renewal evaluation boundaries for the Boron Recycle System are depicted on the following drawings:

E-302-751	Boron Recycle
1MS-09-269	Flow Diagram - Recycle Evaporating Package

A complete list of Boron Recycle System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-19.

**Table 2.3-19:
BORON RECYCLE SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Condensers, Channel Head	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Condensers, Tubes	Pressure Boundary	Table 3.3-1 Item 14
Condensers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14
Heat Exchangers, Shell	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Heat Exchangers, Shell (nozzles)	Pressure Boundary	Table 3.3-1 Item 14
Heat Exchangers, Tubes	Pressure Boundary	Table 3.3-2 Item 6, Table 3.3-1 Item 14
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14
Heat Exchangers, Manifolds	Pressure Boundary	Table 3.3-2 Item 6, Table 3.3-1 Item 14
Valves (Body Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6

2.3.3.3 Building Services

The Building Services System is designed to provide for the structural integrity of various buildings on site. However, some of the components of the Building Services System are mechanical components that maintain a pressure boundary for containment integrity, including valves and piping in the emergency air supply for the Reactor Building personnel airlocks (see **Section 2.4.1.3**). The mechanical license renewal function of the Building Services System is to provide containment isolation.

The license renewal evaluation boundaries for the Building Services System are depicted on the following drawing:

D-302-242	**Station Air Supply To Personnel, Emergency Personnel and Equipment Hatches**

A complete list of Building Services System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-20.

**Table 2.3-20:
BUILDING SERVICES SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.3-2 Item 1, Table 3.3-2 Item 11**
Tubing	Pressure Boundary	**Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 11, Table 3.3-2 Item 14**
Valves (Body Only)	Pressure Boundary	**Table 3.3-2 Item 1, Table 3.3-2 Item 11**

2.3.3.4 Chilled Water System

The Chilled Water System maintains a continuous flow of chilled water through various chilled water coils in different areas of the plant. The Chilled Water System is a closed system with redundant supply and return mains. The HVAC mechanical water chillers reject the total heat load from the refrigeration system to the Service Water System. This system maintains control room habitability under normal and accident conditions.

FSAR Section 9.4.7, Miscellaneous Building Ventilation and Cooling Systems, provides additional information concerning the Chilled Water System.

The license renewal evaluation boundaries for the Chilled Water System are depicted on the following drawings:

D-302-222	Service Water Cooling
D-302-841	Chilled Water Pump & Chiller Area
D-302-842	Chilled Water to Cooling Coils
D-302-843	Chilled Water to Cooling Coils
1MS-54-064-2	VU Mechanical Chillers

A complete list of Chilled Water System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-21.

**Table 2.3-21:
CHILLED WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTION**

Component Type	Component Function	AMR Results
Compressors (Housing Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 4
Condensers, Fins	Heat Transfer	Table 3.3-2 Item 4
Condensers, Shells	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 4

Table 2.3-21:
CHILLED WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTION

Component Type	Component Function	AMR Results
Condensers, Tubes	Pressure Boundary Heat Transfer	Table 3.3-1 Item 16, Table 3.3-2 Item 4
Condensers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 16, Table 3.3-2 Item 4
Condensers, Waterboxes	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16
Eductors, Lubrication System	Pressure Boundary Gas Removal	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 5
Eductors, Jet Pump	Pressure Boundary Gas Removal	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 4
Evaporators, Fins	Heat Transfer	Table 3.3-2 Item 4
Evaporators, Shells	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 4
Evaporators, Tubes	Pressure Boundary Heat Transfer	Table 3.3-2 Item 4, Table 3.3-2 Item 28
Evaporators, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 4
Evaporators, Waterboxes	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Filters	Pressure Boundary Filtration	Table 3.3-2 Item 1, Table 3.3-2 Item 4, Table 3.3-2 Item 5

Table 2.3-21:
CHILLED WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTION

Component Type	Component Function	AMR Results
Flow Control Chambers	Pressure Boundary Throttling	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 4
Orifices	Pressure Boundary Throttling	Table 3.3-1 Item 14, Table 3.3-2 Item 1
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 4, Table 3.3-2 Item 5
Pumps (Casing Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 5
Purge Units	Pressure Boundary Gas Removal	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 4
Sight Glass	Pressure Boundary	Table 3.3-2 Item 23
Tanks	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-2 Item 19
Tube and Tube Fittings	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 4, Table 3.3-2 Item 5

Table 2.3-21:
CHILLED WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTION

Component Type	Component Function	AMR Results
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 4, Table 3.3-2 Item 5

2.3.3.5 Circulating Water System

The Circulating Water System removes thermal energy from the main and auxiliary condensers and dissipates this energy to the Monticello Reservoir. This system is not required to function under plant emergency or faulted conditions.

FSAR Section 10.4.5, Circulating Water System, provides additional information concerning the Circulating Water System.

The only license renewal function of the Circulating Water System (CW) is to provide level instrumentation that will trip the Circulating Water pumps on high level to protect the Intermediate and Control Buildings from flooding. There are no mechanical components/component types required for the Circulating Water System to perform its system intended function; therefore, no aging management review is required. Since instruments are not highlighted as within license renewal evaluation boundaries on flow diagrams, no flow diagrams have been marked for the Circulating Water System.

2.3.3.6 Component Cooling Water System

The Component Cooling Water System serves as an intermediate, closed-loop cooling system to transfer heat from systems and components important to safety, including those which may contain radioactive (or potentially radioactive) fluids, to the Service Water System. The Component Cooling Water System is also utilized during normal plant operation to transfer heat from various systems and components that are not important to safety, but could result in the release of radioactivity to the ultimate heat sink if direct, open loop cooling were used. This latter type of service is referred to as “nonessential.”

FSAR Section 9.2.2, Component Cooling Water System, provides additional information concerning the Component Cooling Water System.

The license renewal evaluation boundaries for the Component Cooling Water System are depicted on the following drawings:

D-302-611	Component Cooling System
D-302-612	Component Cooling System
D-302-613	Component Cooling System Non.Ess.
D-302-614	Component Cooling Sys. To NSSS Pumps
D-806-005	Radiation Monitoring System Diagram Liquid
1MS-09-238 Sh. 16	Flow Diagram - Waste Evaporated Package
1MS-09-269	Flow Diagram - Recycle Evaporating Package

A complete list of Component Cooling Water System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-22.

Table 2.3-22:
COMPONENT COOLING WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Flexible Hose	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-1 Item 14
Heat Exchangers, Channel Head	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16
Heat Exchangers, Shell	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14

Table 2.3-22:
COMPONENT COOLING WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Tubes	Pressure Boundary Heat Transfer	Table 3.3-1 Item 14, Table 3.3-1 Item 16
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-1 Item 16
Coolers, Motor Bearings, Fins	Heat Transfer	Table 3.3-2 Item 24
Coolers, Motor Bearings, Tubes	Pressure Boundary Heat Transfer	Table 3.3-1 Item 5, Table 3.3-2 Item 28
Coolers, Motor Bearings, Tubesheet	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 14
Coolers, Motor Bearings, Water-boxes	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Orifices	Pressure Boundary Throttling	Table 3.3-1 Item 14, Table 3.3-2 Item 1
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 19
Thermowells, Piping	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 1
Pumps (Casing Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 1
Tanks, Surge	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 14, Table 3.3-2 Item 19

**Table 2.3-22:
COMPONENT COOLING WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Tube and Tube Fittings	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 1
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 19

2.3.3.7 Diesel Generator Services Systems

The two Diesel Generators (DG) provide the emergency on-site power supply in the event of an off-site power interruption. Each Diesel Generator is capable of supplying 100 percent of the Engineered Safety Features loads plus other vital, non-nuclear, safety-related loads. Other than testing or maintenance, the emergency Diesel Generator sets operate only when an emergency occurs and provide sufficient power to maintain the reactor in a safe condition.

The mechanical support functions of the Diesel Generators include: (1) Diesel Engine Lube Oil, (2) Diesel Engine Fuel Oil, (3) Diesel Engine Cooling Water, (4) Diesel Engine Air Intake & Exhaust, (5) Diesel Engine Air Starting, and (6) Diesel Engine Crankcase Vacuum.

FSAR Sections 9.5.4 through **9.5.8** provide additional information concerning the Diesel Generator Services Systems.

The license renewal evaluation boundaries for the Diesel Generator Services Systems are depicted on the following drawings:

D-302-222	**Service Water Cooling**
D-302-351	**Diesel Generator-Fuel Oil**
D-302-353	**Diesel Generator Miscellaneous Services**

1MS-32-005 Sh. 2	**Fuel Oil System**
1MS-32-005 Sh. 3	**Lube Oil System**
1MS-32-005 Sh. 4	**Jacket Water System**
1MS-32-005 Sh. 5	**Intercooler & Injector Cooling System**
1MS-32-005 Sh. 6	**Starting & Control Air System - Colt Ind.**
1MS-32-005 Sh. 7	**Crank Case Vac Air Intake and Exhaust System**

A complete list of Diesel Generator Services Systems mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-23.

Table 2.3-23:
DIESEL GENERATOR SERVICES SYSTEMS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Expansion Joint, Engine Exhaust	Pressure Boundary	**Table 3.3-2 Item 1,** **Table 3.3-2 Item 10,** **Table 3.3-1 Item 5**
Filters (body only)	Pressure Boundary Filtration	**Table 3.3-1 Item 5,** **Table 3.3-1 Item 7,** **Table 3.3-2 Item 5**
Flexible Coupling	Pressure Boundary	**Table 3.3-1 Item 2,** **Table 3.3-2 Item 26**
Flexible Hose	Pressure Boundary	**Table 3.3-1 Item 2,** **Table 3.3-2 Item 26**
Heat Exchangers, Channel Head	Pressure Boundary	**Table 3.3-1 Item 5,** **Table 3.3-1 Item 14,** **Table 3.3-1 Item 16,** **Table 3.3-2 Item 1**
Heat Exchangers, Shell	Pressure Boundary	**Table 3.3-1 Item 5,** **Table 3.3-1 Item 14,** **Table 3.3-2 Item 5**

Table 2.3-23:
DIESEL GENERATOR SERVICES SYSTEMS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Tubes	Pressure Boundary Heat Transfer	Table 3.3-1 Item 16, Table 3.3-1 Item 14, Table 3.3-2 Item 5, Table 3.3-2 Item 12, Table 3.3-2 Item 28
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-1 Item 16, Table 3.3-2 Item 5, Table 3.3-2 Item 12, Table 3.3-2 Item 28
Heaters (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 14, Table 3.3-2 Item 5
Mufflers	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-2 Item 10
Orifices	Pressure Boundary Throttling	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 14, Table 3.3-2 Item 1
Pipe and Fittings	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 14, Table 3.3-1 Item 16, Table 3.3-1 Item 17, Table 3.3-2 Item 5, Table 3.3-2 Item 10, Table 3.3-2 Item 25, Table 3.3-2 Item 27, Table 3.3-2 Item 28

Table 2.3-23:
DIESEL GENERATOR SERVICES SYSTEMS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pumps (Casing Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 14, Table 3.3-2 Item 5
Reservoir, Air	Pressure Boundary	Table 3.3-1 Item 5
Reservoir, Rocker	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-2 Item 5
Sight Glass (Body Only)	Pressure Boundary	Table 3.3-2 Item 23
Silencers	Pressure Boundary Filtration	Table 3.3-1 Item 5
Strainers (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-2 Item 5
Tanks	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 17
Tube and Tube Fittings	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 5, Table 3.3-2 Item 12, Table 3.3-2 Item 25, Table 3.3-2 Item 27, Table 3.3-2 Item 28
Turbocharger (Casing Only)	Pressure Boundary	Table 3.3-1 Item 5

Table 2.3-23:
DIESEL GENERATOR SERVICES SYSTEMS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 5, Table 3.3-2 Item 12, Table 3.3-2 Item 25

2.3.3.8 Fire Service System

The Fire Service System is designed to ensure adequate fire protection for each fire hazard. The total fire protection system provides fire detection, audible and visual alarms and extinguishment. Fire Service System functions include: (1) fire protection water supply and distribution, (2) fire detection and alarm, (3) fire extinguishing, (4) cooling of equipment and buildings exposed to fire, (5) control of fire spread, (6) inerting of hazardous atmospheres, and (7) efficient and effective use of proper fire extinguishing agent.

FSAR Section 9.5.1, Fire Protection System, provides additional information concerning the Fire Service System.

The license renewal evaluation boundaries for the Fire Service System are depicted on the following drawings:

D-302-231 Sh. 1	**Fire Service Pumps**
D-302-231 Sh. 2	**Fire Service Hydrants & Loop**
D-302-231 Sh. 3	**Fire Service Reactor Building, Auxiliary Building, Intermediate Building, Diesel Generator Building, Fuel Handling Building, Control Building**
D-302-231 Sh. 4	**Fire Service Turbine Building & Water Treatment Building**

D-302-231 Sh. 5	Fire Service Valve Manifolds
D-302-232	Fire Service-Halon & Low Pressure CO2
1MS-55-040 Sh. 1	Low Pressure CO2 Fire Extinquishing System Schematic Arrangement
1MS-55-040 Sh. 2	Low Pressure CO2 Fire Extinquishing System Schematic Arrangement
1MS-55-040 Sh. 3	Low Pressure CO2 Fire Extinquishing System Schematic Arrangement
1MS-55-040 Sh. 10	Low Pressure CO2 Fire Extinquishing System Schematic Arrangement
1MS-55-059 Sh. 1	Deluge Water Spray System - Deluge Valve Station - Turbine Bldg.
1MS-55-059 Sh. 9	Deluge Water Spray System - Turbine Bldg. Scaler Feed Pump Protection
1MS-55-085 Sh. 10	Valves, Sections, & Details for Control Complex Cable Spreading Rooms & Control Bldg. El. 412
1MS-55-085 Sh. 11	Cable Spreading Room El 425' 0" - Pre-Action Sprinkler System
1MS-55-085 Sh. 12	Cable Spreading Room El 448' 0" - Pre-Action Sprinkler System
1MS-55-085 Sh. 13	Cable Spreading Room El 463' 0" - Pre-Action Sprinkler System
1MS-55-085 Sh. 26	Diesel Generator Bldg. - Diesel Fire Pump Room
1MS-55-085 Sh. 27	Charcoal Filter Plenum Systems - Auxiliary Bldg., Control Bldg., Reactor Bldg.
1MS-55-085 Sh. 28	Valve Details - Charcoal Filter Plenums
1MS-55-137 Sh. 1	Service Water Pump House Pre-Action System
1MS-55-137 Sh. 4	Auxiliary Bldg. El. 463'-0' Pre-Action Sprinkler System
1MS-55-137 Sh. 5	Intermediate Bldg. El. 412'

1MS-55-137 Sh. 6 **Intermediate Bldg. El. 412'**

1MS-55-137 Sh. 6A **Intermediate Building El. 412'-0"**

A complete list of Fire Service System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-24.

Table 2.3-24:
FIRE SERVICE SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Fire Hydrants	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 17, Table 3.3-1 Item 20, Table 3.3-2 Item 18
Flexible Hose	Pressure Boundary	Table 3.3-2 Item 5, Table 3.3-2 Item 25
Mufflers	Noise Reduction	Table 3.3-1 Item 5, Table 3.3-2 Item 10
Nozzles	Throttling	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 20, Table 3.3-2 Item 1, Table 3.3-2 Item 4
Orifices	Pressure Boundary Throttling	Table 3.3-1 Item 20, Table 3.3-2 Item 1
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 17, Table 3.3-1 Item 20, Table 3.3-2 Item 1, Table 3.3-2 Item 4, Table 3.3-2 Item 10, Table 3.3-2 Item 16, Table 3.3-2 Item 18

**Table 2.3-24:
FIRE SERVICE SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Pumps (Casing Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 20
Strainers	Pressure Boundary Filtration	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 20, Table 3.3-2 Item 1
Tanks	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 22, Table 3.3-2 Item 4
Tubing	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-2 Item 5, Table 3.3-2 Item 25
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 7, Table 3.3-1 Item 13, Table 3.3-1 Item 17, Table 3.3-1 Item 20, Table 3.3-2 Item 1, Table 3.3-2 Item 4, Table 3.3-2 Item 5

2.3.3.9 Fuel Handling System

The Fuel Handling System consists of the equipment needed for transporting and handling fuel. The associated fuel handling structures may be generally divided into the following: (1) refueling cavity, (2) refueling canal and fuel transfer canal, which are flooded during plant shutdown for refueling, (3) spent fuel pool, which is kept full of water and is accessible to operating personnel, and (4) new fuel storage area. A fuel transfer tube connects the refueling canal and the fuel transfer canal. This tube is fitted with a blind flange on the refueling canal end and a gate valve on the fuel transfer canal end. This blind flange is always in place, except during refueling, to ensure containment integrity. The fuel transfer tube is required to maintain pressure boundary integrity.

FSAR Section 9.1.4, Fuel Handling System, provides additional information concerning the Fuel Handling System.

The license renewal evaluation boundaries for the Fuel Handling System are depicted on the following drawing:

E-302-651 **Spent Fuel Cooling**

The Fuel Transfer Tube, along with its component intended functions, is provided in Table 2.3-25. Other associated mechanical components are discussed in **Section 2.3.3.22**. Structural components are discussed in **Sections 2.4.1** and **2.4.2.4**.

Table 2.3-25:
FUEL HANDLING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Fuel Transfer Tube	Pressure Boundary	**Table 3.3-2 Item 1, Table 3.3-2 Item 8**

2.3.3.10 Gaseous Waste Processing System

The Gaseous Waste Processing System is designed to remove fission product gases from the reactor coolant in the volume control tank. The system is also designed to collect gases from the boron recycle and waste evaporators, reactor coolant drain tank, recycle holdup tanks and reactor vessel. Fission gases can be contained indefinitely by the Gaseous Waste Processing System.

FSAR Section 11.3, Gaseous Waste System, provides additional information concerning the Gaseous Waste Processing System. The mechanical license renewal functions of the Gaseous Waste Processing System (WG) are to maintain containment isolation for containment integrity and to maintain WG system boundary with the Component Cooling Water and the Chemical and Volume Control systems.

The license renewal evaluation boundaries for the Gaseous Waste Processing System are depicted on the following drawings:

E-302-741	Waste Processing
E-302-742	Waste Processing
E-302-744	Catalytic Hydrogen Recombiner A
E-302-745	Catalytic Hydrogen Recombiner B
1MS-09-012	Assembly & Unit Flow Diagram - AL-623C Waste Gas Compressor

A complete list of Gaseous Waste Processing System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-26.

Table 2.3-26:
GASEOUS WASTE PROCESSING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Channel Head	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Heat Exchangers, Shell	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13
Heat Exchangers, Tubes	Pressure Boundary	Table 3.3-1 Item 14
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14
Heat Exchangers, Helical, Shell	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Heat Exchangers, Helical, Spiral Baffle	Throttling	Table 3.3-1 Item 14

Table 2.3-26:
GASEOUS WASTE PROCESSING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Helical, Tube Coils	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 15
Heat Exchangers, Helical, Tube Manifolds	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 15
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-2 Item 1, Table 3.3-2 Item 15
Tube and Tube Fittings	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 15

2.3.3.11 Industrial Cooler System

The Industrial Cooler System is a closed cooling system, and during normal plant operation, this system supplies water to the cooling coils of the Reactor Building cooling units. During post-accident conditions or following a loss of off-site power, the Reactor Building cooling units are cooled by water from the Service Water System. The activation of an engineered safety features actuation system (ESFAS) signal automatically transfers the source of cooling water for the Reactor Building cooling units.

FSAR Section 9.4.7.2.5, Industrial Cooling Water System, provides additional information concerning the Industrial Cooler System.

The only license renewal function of the Industrial Cooler System is to maintain Reactor Building temperature monitoring capability during accident conditions. There are no mechanical components/component types required for the Industrial Cooler System to perform its system intended function; therefore, no aging management review is required.

2.3.3.12 Instrument Air Supply System

The Instrument Air System, including the Reactor Building Air System, provides clean, dry air for instruments and controls. This system is non-nuclear safety-related (NNS) except for the containment isolation valves and the piping between them. These components are nuclear safety-related (NSR) and in scope for license renewal because they form part of containment. Also in scope for license renewal are the air accumulators and associated air components for various valves required to perform a specified manipulation for event mitigation.

FSAR Section 9.3.1, Compressed Air Systems, provides additional information concerning the Instrument Air Supply System.

The license renewal evaluation boundaries for the Instrument Air Supply System are depicted on the following drawings:

D-302-273	Reactor Building Instrument Air Services
D-302-274	Instrument Air Backup
B-817-113	Control Air Signal Tubing Diagram
B-817-026	Control Air Signal Tubing Diagram
B-817-056 Sh. 1	Control Air Signal Tubing Diagram
B-817-042	Control Air Signal Tubing Diagram
B-817-047	Control Air Signal Tubing Diagram
B-816-061	Control Air Signal Tubing Diagram
B-817-048	Control Air Signal Tubing Diagram
B-816-062	Control Air Signal Tubing Diagram
B-817-062	Control Air Signal Tubing Diagram
B-817-130	PZR PORV Control Air Signal Tubing Diagram
1MS-25-898	Actuator Cylinder Assembly

A complete list of Instrument Air Supply System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-27.

Table 2.3-27:
INSTRUMENT AIR SUPPLY SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 9, Table 3.3-2 Item 13
Tanks	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 13
Tube and Tube Fittings	Pressure Boundary	Table 3.3-1 Item 13, Table 3.3-2 Item 9, Table 3.3-2 Item 13
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 13

2.3.3.13 Leak Detection System

The Leak Detection System provides inputs to the plant annunciator system when leakage of liquids or steam from system piping and/or components is detected. Automatic leak isolation functions such as pump tripping and valve closing are provided in certain critical systems through instrumentation and control functions of the Leak Detection System.

FSAR Section 7.6.5, Leakage Detection System, provides additional information concerning the Leak Detection System.

The portion of the Leak Detection System that is in scope for license renewal is comprised entirely of level switches, level transmitters, temperature elements, and flow switches. These are all active, non-pressure boundary components; therefore, no aging management review is required.

2.3.3.14 Liquid Waste Processing System

The function of the Liquid Waste Processing System is to collect, segregate and process reactor-grade and non-reactor-grade liquid wastes produced during plant operation, refueling and maintenance activities. The processed reactor-grade stream is recycled for plant use, while all non-reactor-grade liquids are processed and disposed of in accordance with applicable NRC regulations.

The Liquid Waste Processing System is designed to control and minimize releases of radioactivity to the environment. The system does not perform any safety-related functions with respect to reactor cooling, shutdown or accident mitigation. However, two of the lines in the system penetrate containment and therefore, portions of the system are safety-related. The system also maintains a pressure boundary with the Component Cooling System and the Spent Fuel Cooling System.

FSAR Section 11.2, Liquid Waste Systems, provides additional information concerning the Liquid Waste Processing System.

The license renewal evaluation boundaries for the Liquid Waste Processing System are depicted on the following drawing:
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E-302-735 **Waste Processing**

A complete list of Liquid Waste Processing System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-28.

Table 2.3-28:
LIQUID WASTE PROCESSING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Condensers, Channel Head	Pressure Boundary	**Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14**
Condensers, Tubes	Pressure Boundary	**Table 3.3-1 Item 14**
Condensers, Tubesheet	Pressure Boundary	**Table 3.3-1 Item 14**

Table 2.3-28:
LIQUID WASTE PROCESSING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Shell	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Heat Exchangers, Tubes	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 21
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 21
Heat Exchanger, Manifolds	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 21
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 9, Table 3.3-2 Item 21
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 21

2.3.3.15 Nuclear & Non-nuclear Plant Drains

Nuclear & Non-Nuclear Plant Drains provide drainage paths for potentially radioactive and non-radioactive liquid wastes through separate systems. Both systems provide drainage and holdup of expected fire fighting water flow (this is accomplished with floor drains and sumps, and does not credit sump pumps and associated discharge piping). In addition, the Non-Nuclear Plant Drains system provides the Circulating Water Pump trip function to prevent flooding in the Control and Intermediate Buildings which is an active function. Therefore the Non-Nuclear Plant Drain System does not require aging management. The Nuclear Plant Drains System provides the additional functions of Reactor Cavity drainage and containment isolation for containment isolation.

The license renewal evaluation boundaries for the Nuclear Plant Drains are depicted on the following drawing:

D-302-821 **Reactor & Auxiliary Building Sump Pumps**

The license renewal evaluation boundaries for the Non-Nuclear Plant Drains are depicted on the following drawing:

D-302-352 **Non-Nuclear Plant Drains**

A complete list of Nuclear Plant Drains mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-29.

Table 2.3-29:
NUCLEAR PLANT DRAINS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 8, Table 3.3-2 Item 21
Valves (Body Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 21

2.3.3.16 Nuclear Sampling System

The Nuclear Sampling System provides for centralized sampling of primary system fluids and permits continuous steam generator blowdown flow to the Secondary Cycle Sampling System for analysis. Samples requiring cooling and depressurization and which are, or could be radioactive are piped to the Nuclear Sampling Room. This system includes sample vessels used at various locations throughout the plant. It also monitors primary letdown water for failed fuel detection.

The mechanical license renewal functions of the system are: providing the sampling capability for the Reactor Coolant System and containment atmosphere following an accident, maintaining containment isolation for containment integrity, and maintaining system boundary with the Component Cooling System.

FSAR Section 9.3.2, Process Sampling System, provides additional information concerning the Nuclear Sampling System.

The license renewal evaluation boundaries for the Nuclear Sampling System are depicted on the following drawings:

D-302-771	**Nuclear Sampling**
D-302-772	**Normal & Post-Accident Sampling**
D-806-001	**Radiation Monitoring System Diagram Atmospheric**
D-912-102	**Reactor Building Cooling System**

A complete list of Nuclear Sampling System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-30.

Table 2.3-30:
NUCLEAR SAMPLING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchanger, Shell	Pressure Boundary	**Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14**
Heat Exchanger, Tubes	Pressure Boundary Heat Transfer	**Table 3.3-1 Item 14, Table 3.3-2 Item 6, Table 3.3-2 Item 7**

Table 2.3-30:
NUCLEAR SAMPLING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 6, Table 3.3-2 Item 9, Table 3.3-2 Item 17
Pumps (Casing Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 17
Tanks	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 20
Tube and Tube Fittings	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6, Table 3.3-2 Item 9, Table 3.3-2 Item 17
Valves (Body Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6, Table 3.3-2 Item 17

2.3.3.17 Radiation Monitoring System

Radiation monitors and analysis of samples are used to monitor process and effluent streams in order to record and control releases of radioactive materials generated as a result of normal operations, including anticipated operational occurrences and postulated accidents.

The mechanical license renewal functions of the system are providing post accident monitoring capability for containment activity and maintaining system boundaries with the Component Cooling, Spent Fuel Cooling, and Chemical and Volume Control systems.

FSAR Section 11.4, Process and Effluent Radiological Monitoring Systems, provides additional information concerning the Radiation Monitoring System.

The license renewal evaluation boundaries for the Radiation Monitoring System are depicted on the following drawings:

D-302-611	**Component Cooling**
D-302-651	**Spent Fuel Cooling**
D-302-771	**Nuclear Sampling**
D-806-010	**Radiation Monitoring System Diagram Area Gamma**
D-806-011	**Radiation Monitoring System Diagram Area Gamma**

A complete list of Radiation Monitoring System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-31.

Table 2.3-31:
RADIATION MONITORING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Instrumentation (Pressure Retaining Only)	Pressure Boundary	**Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 6**
Pipe	Pressure Boundary	**Table 3.3-1 Item 14, Table 3.3-2 Item 6, Table 3.3-2 Item 9**
Tanks	Pressure Boundary	**Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 6**
Tube and Tube Fittings	Pressure Boundary	**Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 6**

Table 2.3-31:
RADIATION MONITORING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Valves (Body Only)	Pressure Boundary	**Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 6**

2.3.3.18 Reactor Makeup Water Supply System

The Reactor Makeup Water Supply System provides for the storage of recycled primary coolant grade water. This system is designed to: (1) supply water to the Chemical and Volume Control System (CVCS), (2) supply makeup water to the spent fuel pool, (3) provide a backup water supply for spray cooling in the pressurizer relief tank, (4) provide a water supply for makeup to and flushing of reactor auxiliary systems, and (5) provide storage capacity equal to or greater than the total capacity of the recycle holdup tanks for recycle primary coolant grade water produced in the boron recovery system and Liquid Waste Processing System. The portion of the Reactor Makeup Water Supply System between the reactor makeup water storage tank and the CVCS and Spent Fuel Pool Cooling System is safety-related; the remaining portion of the system is non-nuclear, safety-related.

FSAR Section 9.2.7, Reactor Makeup Water Supply System, provides additional information concerning the Reactor Makeup Water Supply System.

The license renewal evaluation boundaries for the Reactor Makeup Water Supply System are depicted on the following drawings:

D-302-651	**Spent Fuel Cooling**
E-302-675	**Chemical and Volume Control**
D-302-791	**Reactor Make-Up**

A complete list of Reactor Makeup Water Supply System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-32.

**Table 2.3-32:
REACTOR MAKEUP WATER SUPPLY SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Orifices	Pressure Boundary Throttling	Table 3.3-2 Item 1, Table 3.3-2 Item 17
Pipe	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 9, Table 3.3-2 Item 17
Pumps (Casing Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 17
Tank	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 20
Tube and Tube Fittings	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 17
Valves (Body Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 17

2.3.3.19 Roof Drains System

The Roof Drains System provides for drainage of various plant structures. The Roof Drains System fluid is water draining from the demister banks and plenums of the Reactor Building Cooling Units (RBCUs). The RBCUs are capable of operation during emergency conditions, with potential exposure to Reactor Building spray solution. The mechanical license renewal function of the system is to maintain RBCU drain flow piping integrity.

The Roof Drains System is not described in the FSAR.

The license renewal evaluation boundaries for the Roof Drains System are depicted on the following drawing:

D-302-824 **Reactor Building Cooling Unit Drains**

A complete list of Roof Drains System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-33.

Table 2.3-33:
ROOF DRAINS SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 22

2.3.3.20 Station Service Air System

The Station Service Air System provides air for general plant use and is distributed via quick disconnect hose connections throughout the plant. The operation of this system is not required to mitigate the consequences of a loss-of-coolant accident or achieve a safe shutdown condition. The mechanical license renewal functions of the system are to provide containment isolation for containment integrity and to supply air to the Personnel, Emergency Personnel, and Equipment Hatches.

FSAR Section 9.3.1, Compressed Air Systems, provides additional information concerning the Station Service Air System.

The license renewal evaluation boundaries for the Station Service Air System are depicted on the following drawings:

D-302-241	**Station Service Air**
D-302-242	**Station Air Supply To Personnel, Emergency Personnel and Equipment Hatches**

A complete list of Station Service Air System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-34.

Table 2.3-34:
STATION SERVICE AIR SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 14
Tube and Tube Fittings	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-2 Item 1
Valves (Body Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 1, Table 3.3-2 Item 14

2.3.3.21 Service Water System

The Service Water System (SWS) provides water from the Service Water Pond for cooling of the emergency diesel generators, component cooling heat exchangers, heating ventilating and air conditioning (HVAC) mechanical water chiller condensers, and Service Water Pumphouse cooling coils. During post-accident conditions, loss of offsite power or testing, the Service Water System cools the reactor building cooling units (RBCUs). In addition, this system is the backup water source for the Emergency Feedwater and Component Cooling Water Systems. The system consists of two independent full capacity loops with the capability of valving a third swing service water pump into either loop. The Service Water System is safety-related, and is designed such that a single failure does not cause loss of cooling to more than one of the redundant loops.

FSAR Section 9.2.1, Service Water System, provides additional information concerning the Service Water System.

The license renewal evaluation boundaries for the Service Water System are depicted on the following drawings:

D-302-221	**Service Water Cooling**

D-302-222 Service Water Cooling

A complete list of Service Water System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-35.

Table 2.3-35:
SERVICE WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Couplings	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16, Table 3.3-1 Item 17
Coolers, Motor Bearings	Pressure Boundary Heat Transfer	Table 3.3-1 Item 5, Table 3.3-1 Item 16, Table 3.3-1 Item 24
Expansion Joints, Mechanical, Piping	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16
Expansion Joints, Mechanical, Bellows	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-1 Item 16
Orifices	Pressure Boundary Throttling	Table 3.3-2 Item 1, Table 3.3-1 Item 16
Pipe	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16, Table 3.3-1 Item 17
Thermowells, Piping	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-1 Item 16
Pipe and Fittings	Pressure Boundary	Table 3.3-2 Item 8
Pumps, (Casing Only)	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16

Table 2.3-35:
SERVICE WATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Trash Racks	Filtration	Table 3.3-1 Item 16
Traveling Screens, Cloth Screen	Filtration	Table 3.3-1 Item 16
Traveling Screens, Screen Frame	Filtration	Table 3.3-1 Item 16
Tube and Tube Fittings	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 9, Table 3.3-1 Item 16
Valves (Body Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 16

2.3.3.22 Spent Fuel Cooling System

The Spent Fuel Cooling System cools spent fuel pool water to remove decay heat from the spent fuel elements. This system also: (1) transfers water between the refueling water storage tank (RWST) and refueling cavity, (2) maintains purity and clarity of water in spent fuel pool and/or refueling cavity, (3) provides means for adding boric acid to spent fuel pool, (4) provides means for adding demineralized water to spent fuel pool, (5) monitors spent fuel coolant for excessive radioactivity due to defective fuel elements, (6) provides for filtering and/or demineralization to clean the water in the RWST, and (7) maintains a water shield above spent fuel elements to limit radiation levels in the area of the pool.

FSAR Section 9.1.3, Spent Fuel Cooling System, provides additional information concerning the Spent Fuel Cooling System.

The license renewal evaluation boundaries for the Spent Fuel Cooling System are depicted on the following drawing:

D-302-651 **Spent Fuel Cooling**

A complete list of Spent Fuel Cooling System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-36.

Table 2.3-36:
SPENT FUEL COOLING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Channel Head	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-2 Item 6
Heat Exchangers, Shell	Pressure Boundary	Table 3.3-1 Item 5, Table 3.3-1 Item 13, Table 3.3-1 Item 14
Heat Exchangers, Tubes	Pressure Boundary Heat Transfer	Table 3.3-1 Item 14, Table 3.3-2 Item 6
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 6
Orifices	Pressure Boundary Throttling	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Pipe	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6, Table 3.3-2 Item 9
Pumps (Casing Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Tube and Tube Fittings	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Valves (Body Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6

2.3.3.23 Thermal Regeneration System

The load following capabilities of the (Boron) Thermal Regeneration System (BTRS), which were part of the original design, were removed by plant modification MRF 21511. The system continues to be used as deborating demineralizers to reduce reactor coolant boron concentration towards the end of core life. The Thermal Regeneration System is also used to cool the letdown flow for enhanced Reactor Coolant Pump seal performance, and to clean up the Reactor Coolant System before shutting down the reactor. The mechanical license renewal function of this system is to maintain a pressure boundary with the Chemical and Volume Control System.

FSAR Section 9.3.4, Chemical and Volume Control System, provides additional information concerning the Thermal Regeneration System.

The license renewal evaluation boundaries for the Thermal Regeneration System are depicted on the following drawing:

E-302-676 **Chemical and Volume Control**

A complete list of Thermal Regeneration System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-37.

Table 2.3-37:
THERMAL REGENERATION SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Demineralizers	Pressure Boundary	**Table 3.3-2 Item 1, Table 3.3-2 Item 6**
Heat Exchangers, Channel Head	Pressure Boundary	**Table 3.3-2 Item 1, Table 3.3-2 Item 6**
Heat Exchangers, Shell	Pressure Boundary	**Table 3.3-2 Item 1, Table 3.3-2 Item 6**

Table 2.3-37:
THERMAL REGENERATION SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Heat Exchangers, Tubes	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 6
Heat Exchangers, Tubesheet	Pressure Boundary	Table 3.3-1 Item 14, Table 3.3-2 Item 1, Table 3.3-2 Item 6
Orifices	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Pipe	Pressure Boundary	Table 3.3-2 Item 6, Table 3.3-2 Item 9
Tube and Tube Fittings	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6
Valves (Body Only)	Pressure Boundary	Table 3.3-2 Item 1, Table 3.3-2 Item 6

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** 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.

The following systems are included in this section:

- Auxiliary Boiler Steam and Feedwater System (**Section 2.3.4.1**)
- Condensate System (**Section 2.3.4.2**)
- Emergency Feedwater System (**Section 2.3.4.3**)
- Extraction Steam System (**Section 2.3.4.4**)
- Feedwater System (**Section 2.3.4.5**)
- Gland Sealing Steam System (**Section 2.3.4.6**)
- Main Steam System (**Section 2.3.4.7**)
- Main Steam Dump System (**Section 2.3.4.8**)
- Main Turbine & Turbine Accessories Systems (**Section 2.3.4.9**)
- Turbine Cycle Sampling System (**Section 2.3.4.10**)
- Steam Generator Blowdown System (**Section 2.3.4.11**)
- Turbine Electro-Hydraulic System (**Section 2.3.4.12**)

2.3.4.1 Auxiliary Boiler Steam And Feedwater System

The Auxiliary Boiler Steam and Feedwater (AS) System provides steam to various plant equipment as required during all modes of plant operation. This system is non-nuclear safety-related. The mechanical license renewal function of this system is to isolate the section of AS piping supplying the Auxiliary Building in order to prevent a high energy fluid piping rupture from affecting nuclear safety-related equipment in the Auxiliary Building.

The license renewal evaluation boundaries for the Auxiliary Boiler Steam and Feedwater System are depicted on the following drawing:

D-302-051	**Auxiliary Steam**

A complete list of Auxiliary Boiler Steam and Feedwater System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-38.

Table 2.3-38:
AUXILIARY BOILER STEAM AND FEEDWATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	Table 3.4-1 Item 5, Table 3.4-1 Item 7, Table 3.4-1 Item 13
Valves (Body Only)	Pressure Boundary	Table 3.4-1 Item 5, Table 3.4-1 Item 7, Table 3.4-1 Item 13, Table 3.4-2 Item 1, Table 3.4-2 Item 5

2.3.4.2 Condensate System

The Condensate System is designed to pump condensed turbine exhaust steam from the main condenser hotwell through the low pressure feedwater heaters to maintain deaerator storage tank level for anticipated operating conditions. It also serves as a source of cooling water for the steam packing condenser and steam generator blowdown heat exchanger, and provides sealing water for various vacuum valves and feedwater pump seals.

Except for the condensate storage tank (CST), the Condensate System is non-nuclear, safety-related. The CST is safety-related since it is the primary inventory source for the Emergency Feedwater System. Makeup water to the CST is demineralized water, admitted through the condenser and condenser storage subsystem.

FSAR Section 10.4.7.1, Condensate System, provides additional information concerning the Condensate System.

The license renewal evaluation boundaries for the Condensate System are depicted on the following drawing:

D-302-101	**Condensate**

A complete list of Condensate System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-39.

Table 2.3-39:
CONDENSATE SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Tank	Pressure Boundary	**Table 3.4-1 Item 2, Table 3.4-1 Item 11**

2.3.4.3 Emergency Feedwater System

The Emergency Feedwater System is designed to deliver sufficient feedwater to the steam generators for cooldown subsequent to a loss of normal feedwater supply (i.e., when the Main Feedwater System is not available), and during an Anticipated Transient Without Scram (ATWS) event. The Emergency Feedwater System operates in conjunction with the Main Steam Dump System, if available, or the main steam power relief valves and safety valves, to remove thermal energy from the steam generators. The Emergency Feedwater System is also used to supply feedwater to the steam generators during testing, startup, shutdown and layup operations. During normal plant operation, the system is in a standby condition, with the system controls set for automatic operation.

FSAR Section 10.4.9, Emergency Feedwater System, provides additional information concerning the Emergency Feedwater System.

The license renewal evaluation boundaries for the Emergency Feedwater System are depicted on the following drawing:

D-302-085	**Emergency Feedwater (Nuclear)**
1MS-17-125	**Terry Turbine Diagram - Oil Piping**

A complete list of Emergency Feedwater System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-40.

Table 2.3-40:
EMERGENCY FEEDWATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Filter	Pressure Boundary Filtration	Table 3.4-1 Item 4, Table 3.4-1 Item 5, Table 3.4-1 Item 13, Table 3.4-2 Item 2, Table 3.4-2 Item 3
Heat Exchanger, Shell	Pressure Boundary	Table 3.4-1 Item 4, Table 3.4-1 Item 5, Table 3.4-1 Item 13
Heat Exchanger, Tubes	Pressure Boundary Heat Transfer	Table 3.4-2 Item 3, Table 3.4-2 Item 6
Orifices	Pressure Boundary Throttling	Table 3.4-1 Item 2, Table 3.4-1 Item 4, Table 3.4-1 Item 5, Table 3.4-1 Item 12, Table 3.4-1 Item 13, Table 3.4-2 Item 1
Pipe	Pressure Boundary	Table 3.4-1 Item 2, Table 3.4-1 Item 4, Table 3.4-1 Item 5, Table 3.4-1 Item 11, Table 3.4-1 Item 13, Table 3.4-2 Item 1
Pump, (Casing Only)	Pressure Boundary	Table 3.4-1 Item 2, Table 3.4-1 Item 4, Table 3.4-1 Item 5, Table 3.4-1 Item 13
Strainers	Pressure Boundary	Table 3.4-1 Item 2, Table 3.4-2 Item 1

Table 2.3-40:
EMERGENCY FEEDWATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Tank (Reservoir)	Pressure Boundary	Table 3.4-1 Item 4, Table 3.4-1 Item 5, Table 3.4-1 Item 13
Thermowells, Piping	Pressure Boundary	Table 3.4-1 Item 4, Table 3.4-2 Item 1
Tube and Tube Fittings	Pressure Boundary	Table 3.4-1 Item 2, Table 3.4-2 Item 1
Valves (Body Only)	Pressure Boundary	Table 3.4-1 Item 2, Table 3.4-1 Item 4, Table 3.4-1 Item 5, Table 3.4-1 Item 11, Table 3.4-1 Item 13, Table 3.4-2 Item 1, Table 3.4-2 Item 2, Table 3.4-2 Item 3

2.3.4.4 Extraction Steam System

The Extraction Steam System supplies steam for heating the condensate and feedwater, and also, for maintaining the auxiliary boilers in a hot stand-by condition. The mechanical license renewal function of this system is to provide a means of Main Steam isolation (when used in conjunction with components from various other systems) for a steamline break coincident with failure of a Main Steam Isolation Valve.

The Extraction Steam System is not described in the FSAR.

The license renewal evaluation boundaries for the Extraction Steam System are depicted on the following drawing:

D-302-041	Extraction Steam

A complete list of Extraction Steam System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-41.

Table 2.3-41:
EXTRACTION STEAM SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.4-1 Item 2, Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 13**
Valves (Body Only)	Pressure Boundary	**Table 3.4-1 Item 2, Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 13**

2.3.4.5 Feedwater System

The Feedwater System is designed to pump feedwater from the deaerator storage tank through two stages of high pressure heaters to the steam generators. The operation of this system ensures that the required amount of heated and deaerated water is available to maintain an adequate steam generator water level during normal plant operation and transients. The nuclear portion of the Feedwater System conveys feedwater from the non-nuclear portion of the Feedwater System (located within the Turbine Building) to the steam generators, and includes the containment isolation valves.

FSAR Section 10.4.7.2, Feedwater System, provides additional information concerning the Feedwater System.

The license renewal evaluation boundaries for the Feedwater System are depicted on the following drawing:

D-302-083	**Feedwater (Nuclear)**

A complete list of Feedwater System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-42.

Table 2.3-42:
FEEDWATER SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Flow Venturi	Pressure Boundary Throttling	Table 3.4-1 Item 2, Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 13
Pipe	Pressure Boundary	Table 3.4-1 Item 2, Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 13
Tube and Tube Fittings	Pressure Boundary	Table 3.4-2 Item 1, Table 3.4-1 Item 2
Valves (Body Only)	Pressure Boundary	Table 3.4-2 Item 1, Table 3.4-1 Item 2, Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 13

2.3.4.6 Gland Sealing Steam System

The Gland Sealing Steam System is designed to provide steam to the main turbine and feedwater pump turbine shaft seals in order to prevent air leakage into and/or steam leakage out of the turbine casings. Sealing steam is normally supplied to the Gland Sealing Steam System from the Main Steam System under all load conditions, but may be provided by the auxiliary boiler through the Auxiliary Steam System. The mechanical license renewal function of this system is to provide a means of Main Steam isolation (when used in conjunction with components from various other systems) for a steamline break coincident with failure of a Main Steam Isolation Valve.

FSAR Section 10.4.3, Turbine Gland Sealing System and **FSAR Section 10.3.2.3**, Main Steam Isolation Valves, provide additional information concerning the Gland Sealing Steam System.

The Gland Sealing Steam System was included within the scope of license renewal for non-nuclear, safety-related components that could potentially impact the performance of safety functions.

The license renewal evaluation boundaries for the Gland Sealing Steam System are depicted on the following drawing:

D-302-141 **Turbine Gland Steam**

A complete list of Gland Sealing Steam System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-43.

Table 2.3-43
GLAND SEALING STEAM SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 7**
Valves (Body Only)	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 7**

2.3.4.7 Main Steam System

The Main Steam System conveys saturated steam from the three steam generators to the turbine-generator. Main steam is also supplied, through branch lines, to the following: (a) feedwater pump drive turbines, (b) emergency feedwater pump drive turbine, (c) moisture separator reheaters, (d) auxiliary steam system, (e) deaerating feedwater heater, and (f) steam dumps to the condenser and atmosphere.

FSAR Section 10.3, Main Steam Supply System, provides additional information concerning the Main Steam System.

The license renewal evaluation boundaries for the Main Steam System are depicted on the following drawings:

D-302-011	**Main Stream (Nuclear)**
D-302-012	**Main Stream (Non-Nuclear)**
D-302-014	**Main & Reheat Steam (Non-Nuclear)**
D-302-121	**Steam Drains**
D-302-122	**Feed Pump Start-Up, Extraction & Misc. Steam Drains**
D-302-181	**Turbine Cycle Sampling**
1MS-17-125-5	**Diagram - Terry Turbine Oil Piping**

A complete list of Main Steam System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-44.

Table 2.3-44:
MAIN STEAM SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pump Turbine (Casing Only)	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 13, Table 3.4-2 Item 4**
Pipe	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 7, Table 3.4-1 Item 13**

Table 2.3-44:
MAIN STEAM SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Valves (Body Only)	Pressure Boundary	Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 7, Table 3.4-1 Item 13, Table 3.4-2 Item 1, Table 3.4-2 Item 4, Table 3.4-2 Item 5
Steam Traps	Pressure Boundary	Table 3.4-1 Item 5, Table 3.4-1 Item 7, Table 3.4-1 Item 13

2.3.4.8 Main Steam Dump System

The Main Steam System is capable of following a large turbine-generator load reduction without reactor trip, through actuation of the Main Steam Dump System. This system bypasses main steam to the main condenser and/or to the atmosphere. Steam dump valves permit unit operation at turbine loads lower than the minimum power setting (15% reactor power) of the Nuclear Steam Supply System (NSSS) automatic control. In addition, the steam dump valves permit reduction of turbine-generator load at a rate greater than the 5% per minute maximum rate of load reduction for the NSSS.

The mechanical license renewal function of this system is to provide a means of Main Steam isolation (when used in conjunction with components from various other systems) for a steamline break coincident with failure of a Main Steam Isolation Valve.

FSAR Section 10.4.4, Turbine Bypass System, provides additional information concerning the Main Steam Dump System.

The license renewal evaluation boundaries for the Main Steam Dump System are depicted on the following drawing:

D-302-031	**Main Stream Dump System**

A complete list of Main Steam Dump System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-45.

Table 2.3-45:
MAIN STEAM DUMP SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 7, Table 3.4-1 Item 13**
Tube and Tube Fittings	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 7, Table 3.4-1 Item 13**
Valves (Body Only)	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 7, Table 3.4-1 Item 13**

2.3.4.9 Main Turbine and Turbine Accessories Systems

The Main Turbine System receives main steam from the steam generators and converts steam energy into mechanical energy for the main generator. The Turbine Accessories system supplies high pressure bearing lift oil to the turbine and generator bearings to lift the shaft slightly and reduce the torque requirements on the turning gear. These two systems provide turbine trip signals that have a license renewal function of providing a means of Main Steam isolation (when used in conjunction with components from various other systems) for a steamline break coincident with failure of a Main Steam Isolation Valve.

FSAR Section 10.2, Turbine Generator, provides additional information concerning the Main Turbine.

There are no mechanical components/component types required for the Main Turbine or Turbine Accessories systems to perform their system intended functions; therefore, no aging management review is required and no drawings are needed to depict license renewal evaluation boundaries.

2.3.4.10 Turbine Cycle Sampling System

The Turbine Cycle Sampling System provides sampling of secondary system fluids from locations such as the Main Condenser Hotwell, Deaerator, Feedwater Booster Pumps, High Pressure Heater Drains, Emergency Feedwater Pumps, and Main Steam System. The mechanical license renewal function of this system is to provide a means of Main Steam isolation (when used in conjunction with components from various other systems) for a steamline break coincident with failure of a Main Steam Isolation Valve.

FSAR Section 10.3.5, Water Chemistry, provides additional information concerning the Turbine Cycle Sampling System.

The license renewal boundaries for the Turbine Cycle Sampling System are depicted on the following drawings:

D-302-012	**Main Steam System**
D-302-181	**Turbine Cycle Sampling**

A complete list of Turbine Sampling System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-46.

**Table 2.3-46:
TURBINE CYCLE SAMPLING SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 7**
Valves (Body Only)	Pressure Boundary	**Table 3.4-1 Item 5, Table 3.4-1 Item 7**

2.3.4.11 Steam Generator Blowdown System

The Steam Generator Blowdown System continuously purges the steam generators of concentrated impurities, thereby maintaining secondary side steam generator water chemistry. This system is non-nuclear, safety-related except for the portion inside the Reactor Building, up to and including the containment isolation valves.

FSAR Section 10.4.8, Steam Generator Blowdown System, provides additional information concerning the Steam Generator Blowdown System.

The license renewal evaluation boundaries for the Steam Generator Blowdown System are depicted on the following drawings:

D-302-771	**Nuclear Sampling**
D-302-781	**Steam Generator Blowdown**

A complete list of Steam Generator Blowdown System mechanical component types subject to aging management review, along with their component intended functions, is provided in Table 2.3-47.

Table 2.3-47:
STEAM GENERATOR BLOWDOWN SYSTEM
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Component Function	AMR Results
Pipe	Pressure Boundary	**Table 3.4-1 Item 2, Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 13**
Valves (Body Only)	Pressure Boundary	**Table 3.4-1 Item 2, Table 3.4-1 Item 5, Table 3.4-1 Item 6, Table 3.4-1 Item 13**

2.3.4.12 Turbine Electro-Hydraulic System

An Electro-Hydraulic Control System actuates and controls the steam valves. This system is completely separated from the bearing oil supply.

During normal plant operation, reactor power is controlled to match turbine load as measured by turbine first stage pressure. The Turbine Electro-Hydraulic Control System establishes the desired turbine steady-state load. This system provides turbine trip signals that have license renewal functions of ATWS mitigation and Main Steam isolation (when used in conjunction with components from various other systems) for a steamline break coincident with failure of a Main Steam Isolation Valve.

FSAR Section 10.2.2.2, Turbine Generator Control, provides additional information concerning the Turbine Electro-Hydraulic System.

There are no mechanical components/component types required for the Turbine Electro-Hydraulic System to perform its system intended function; therefore, no aging management review is required.

2.3.5 REFERENCES

2.3-1	10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 60 FR 22461, May 8, 1995.

2.4 STRUCTURES AND STRUCTURAL COMPONENTS SCOPING AND SCREENING RESULTS

The determination of the structures and structural components within the scope of license renewal is made by initially identifying all VCSNS structures and then reviewing them to determine which structures and structural components satisfy one or more of the criteria in 10 CFR 54.4 [**Reference 2.4-1**]. The results of the structural scoping were cross-discipline reviewed to ensure that all structures containing in-scope electrical and mechanical components were addressed. The scoping and screening process is described in **Section 2.1** and the results of the structures scoping review are presented in **Section 2.2**.

The structures and structural component types scoping and screening results are provided below in two subsections:

- Reactor Building and Internal Structures (see **Site Facilities Drawing**)
- Other Structures (see **Site Facilities Drawing**)

A table is provided for each of the structures within the scope of license renewal that lists the structural component type, the associated intended functions and a link to Section 3 AMR results. The tables contain an abbreviated description of the intended functions. A complete description of each intended function is presented in Table 2.4-1.

Table 2.4-1:
INTENDED FUNCTIONS FOR STRUCTURAL COMPONENT TYPES

Abbreviated Description	Complete Function Description
Rated Fire Barrier	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
Shelter/protection to NSR	Provide shelter/protection to safety-related components.
Structural Support to NSR	Provide structural and/or functional support to safety-related equipment.
Flood Protection Barrier	Provide flood protection barrier (internal and external flooding event).
Pressure Boundary or Leak Barrier	Provide pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events.

Table 2.4-1:
INTENDED FUNCTIONS FOR STRUCTURAL COMPONENT TYPES

Abbreviated Description	Complete Function Description
Spray Shield or Curb	Provide spray shield or curbs for directing flow.
Radiation Shielding	Provide shielding against radiation.
Missile Barrier	Provide missile barrier (internally or externally generated).
HELB Shielding	Provide shielding against high-energy line breaks.
Structural Support to NNS	Provide structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions. Note: This function is also used when component type provides structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events.
Pipe Whip Restraint	Provide pipe whip restraint.
Release Path	Provide path for release of filtered and unfiltered gaseous discharge.
Source of Cooling Water	Provide source of cooling water for plant shutdown.
Heat Sink	Provide heat sink during SBO or design basis accidents.
Impound Water	Impound water for ultimate heat sink during loss of Monticello Reservoir.

2.4.1 REACTOR BUILDING

The Reactor Building (for plant location see **Site Facilities Drawing**) is described in **FSAR Section 3.8.1** [**Reference 2.4-2**], Concrete Reactor Building. The Reactor Building is a post tensioned, reinforced concrete structure with an integral steel liner. The Reactor Building consists of a cylindrical wall, a shallow dome roof and a foundation mat with a depressed incore instrumentation pit under the reactor vessel. The foundation mat bears on fill concrete that extends to competent rock. At the underside of the Reactor Building foundation mat a tendon access gallery is formed into the top of the fill concrete. A retaining wall, extending approximately one quarter (1/4) of the way around the Reactor Building, protects the below grade portions of the Reactor Building wall from the subgrade and groundwater. Adjacent buildings surround the remaining three-quarters (3/4) of the Reactor Building.

2.4.1.1 Structure And Foundation

The Reactor Building structure, as described in **FSAR Section 3.8.1** [**Reference 2.4-2**], consists of a circular cylindrical shell with a shallow dome roof set upon a circular structural mat. The shell consists of a reinforced concrete wall and a shallow dome roof, joined at a ring girder. The foundation consists of a reinforced concrete structural mat, which is supported by fill concrete that extends down to competent rock.

The Reactor Building shell is post tensioned by ungrouted tendons. The cylindrical wall employs a three-buttress, 240-degree hoop tendon concept, with 115 vertical tendons and 150 hoop tendons. The dome contains a total of 99 tendons arranged in a three-way system with 33 tendons per band.

2.4.1.2 Containment Liner

The Reactor Building is lined on the inside face with a carbon steel plate liner that forms an essentially leak-tight membrane sealing the entire Reactor Building for any postulated conditions which may be encountered throughout the operating life of the plant. At its base, in the haunch area, a truncated conical transition section tapers inward to accommodate the thickened concrete of the cylindrical shell.

A dome closes the top of the cylindrical portion of the liner. The bottom of the liner consists of flat floor liner plates welded to anchors that are embedded in the mat concrete. The liner plate extends downward into the foundation mat to line the incore instrumentation pit, the Reactor Building sump, the incore instrumentation pit sump, the residual heat removal sumps, and the Reactor Building spray sumps. The incore instrumentation pit walls are lined with carbon steel plates, while the pit bottom and the walls of the incore instrumentation tunnel are lined with stainless steel plates. The residual heat removal sump, Reactor Building sump, and Reactor Building spray sump floors and sidewalls are lined with stainless steel plate.

Small diameter circular overlay plates are welded to the liner plate to support piping, ducts, conduit, and electric cable trays. Studs or angle anchors are provided on the liner behind the attachment plates to transfer loads on the pads into the concrete shell.

2.4.1.3 Penetrations

All Reactor Building penetrations are anchored to the concrete Reactor Building wall or foundation mat so that loads are transferred from the penetrations to the concrete. All penetrations satisfy the requirements of 10 CFR 50, Appendix J [**Reference 2.4-3**], Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.

Piping penetrations consist of a sleeve around the outside of the piping. The piping is joined to the sleeve inside the Reactor Building by an attachment plate. Outside the Reactor Building, piping is attached to the sleeve by an attachment plate or by a bellows assembly.

Electrical penetration sleeves are provided to accommodate electrical and instrumentation cables that pass through the Reactor Building wall. The sleeves are welded to the Reactor Building inner reinforcing plates. The electrical leads are installed in the penetration assemblies that are bolted to the electrical penetration sleeve.

Spare penetrations consist of sleeves passing through the Reactor Building wall with the liner reinforced around the sleeve. Both ends of the sleeve are sealed with butt-welded pipe caps.

A fuel transfer tube penetrates the Reactor Building connecting the refueling canal in the Reactor Building and the fuel transfer canal in the Fuel Handling Building. This penetration consists of a pipe installed inside a sleeve.

Two personnel airlocks are provided for access to the Reactor Building, each with two doors, one on the inside and one on the outside. Each door is sealed with double O-rings, which are tested and replaced when warranted by their condition. The O-rings are not long-lived components and therefore do not require an aging management review.

An equipment hatch, equipped with an inside-mounted hatch cover, is also provided for access to the Reactor Building. A concrete shield located outside the Reactor Building acts as a missile and biological shield. The hatch cover is sealed with double O-rings, which are tested and replaced when warranted by their condition. The O-rings are not long-lived and therefore do not require an aging management review.

2.4.1.4 Internal Structures

The internal structures of the Reactor Building consist of the following:

- Primary shield wall surrounding and supporting the reactor vessel.
- Secondary shield walls, surrounding and laterally supporting each steam generator and the pressurizer.
- Refueling cavity and fuel transfer canal.
- Mezzanine floor and operating floor, both consisting of concrete slabs supported by structural steel framing.
- Polar crane supports.
- Concrete basement slab supported by the structural foundation mat.

The primary shield wall is a reinforced concrete wall surrounding and supporting the reactor vessel. The lower portion of the wall, below the base slab, is a cylindrical section surrounded by structural foundation mat concrete. The Reactor Building foundation mat provides vertical support of the primary shield wall.

The secondary shield wall forms three compartments that are located adjacent to and are connected with the primary shield wall. These compartments form enclosures for Reactor Coolant System equipment. The function of these enclosures is to protect the Reactor Building from the effects of a postulated pipe break, provide biological shielding, and provide lateral support for Reactor Coolant System equipment. The pressurizer is enclosed in a separate compartment.

The refueling cavity/fuel transfer canal is located above and adjacent to the reactor vessel and is a stainless steel lined reinforced concrete structure. The walls of the refueling cavity/ fuel transfer canal form part of the secondary shield wall system.

The operating floor slab and the mezzanine floor slab are supported by a structural steel framing system and by the secondary shield walls. The inner edge of these reinforced concrete slabs are keyed and doweled into the secondary shield walls. The outer edge stops short of the Reactor Building liner to provide separation from the Reactor Building.

The structural steel framing system consists of radially oriented girders supported on the inboard ends by concrete corbels or steel brackets attached to the secondary shield wall. Steel columns support the outboard ends of the girders, with base plates supported on piers cast in the basement floor slab. Perimeter girders between the columns support the outboard portion of the operating and mezzanine floor slabs. Steel beams frame between the girders to support the floor slabs.

Perimeter runway girders around the Reactor Building support the polar crane. The circular crane rail is attached to the top flange of the perimeter runway girders. The runway girders include a weldment to engage the polar crane seismic uplift lugs. The polar crane brackets provide vertical and lateral support for the polar crane steel runway girders. The brackets extend through the Reactor Building liner and are anchored into the concrete wall.

The Reactor Building foundation mat supports a concrete basement slab, which supports and anchors all the internal structures and equipment. The Reactor Building base mat liner is not penetrated for anchorage of any internal structure or equipment.

2.4.1.5 Conclusions

The Reactor Building is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides shelter/protection to safety-related components.
- Provides structural and/or functional support to safety-related equipment.
- Provides flood protection barrier (internal and external flooding event).
- Provides pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events.
- Provides spray shield or curbs for directing flow.
- Provides shielding against radiation.
- Provides missile barrier (internally or externally generated).
- Provides shielding against high-energy line breaks.
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.
- Provides pipe whip restraint.

A complete list of Reactor Building structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-2.

**Table 2.4-2:
REACTOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 25, Table 3.5-1 Item 26, Table 3.5-1 Item 27, Table 3.5-1 Item 28, Table 3.5-1 Item 29
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13, Table 3.5-1 Item 25, Table 3.5-1 Item 26, Table 3.5-1 Item 27, Table 3.5-1 Item 28, Table 3.5-1 Item 29
Bellows (Penetration)	Structural Support to NSR	Table 3.5-1 Item 1, Table 3.5-1 Item 2, Table 3.5-1 Item 3
Cable Tray & Conduit	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13
Checkered Plate	Structural Support to NNS	Table 3.5-1 Item 13
Compressible Joints & Seals	Flood Protection Barrier	Table 3.5-1 Item 6
Control Board (Refuel Cavity Crane)	Shelter/protection to NSR	Table 3.5-1 Item 13
Crane Rails & Girders	Structural Support to NSR	Table 3.5-1 Item 13
Electrical and Instrument Panels & Enclosures	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13

Table 2.4-2:
REACTOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 25
Equipment Component Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13, Table 3.5-1 Item 25, Table 3.5-1 Item 28, Table 3.5-1 Item 29
Equipment Hatch	Rated Fire Barrier Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding HELB Shielding	Table 3.5-1 Item 4, Table 3.5-1 Item 5, Table 3.5-1 Item 13
Equipment Pads	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 7, Table 3.5-1 Item 10, Table 3.5-1 Item 15, Table 3.5-1 Item 25
Escape Air Lock	Rated Fire Barrier Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding HELB Shielding	Table 3.5-1 Item 4, Table 3.5-1 Item 5, Table 3.5-1 Item 13
Expansion Anchors	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 25, Table 3.5-1 Item 26, Table 3.5-1 Item 27, Table 3.5-1 Item 28
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	Table 3.5-1 Item 7, Table 3.5-1 Item 10, Table 3.5-1 Item 15

Table 2.4-2:
REACTOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 13
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 7, Table 3.5-1 Item 10, Table 3.5-1 Item 15
Flood Curbs (Steel)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 13
Flood, Pressure and Specialty Doors	Rated Fire Barrier Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding Missile Barrier HELB Shielding	Table 3.5-1 Item 13
Foundations	Structural Support to NSR Radiation Shielding Structural Support to NNS	Table 3.5-1 Item 7, Table 3.5-1 Item 8, Table 3.5-1 Item 9, Table 3.5-1 Item 10, Table 3.5-1 Item 15
Hatches (Steel)	Rated Fire Barrier Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding Missile Barrier HELB Shielding	Table 3.5-1 Item 4, Table 3.5-1 Item 5
HVAC Duct Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13, Table 3.5-1 Item 25
Instrument Line Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13, Table 3.5-1 Item 25

Table 2.4-2:
REACTOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Instrument Racks & Frames	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13
Jet Barriers (Concrete and Steel)	Structural Support to NSR Spray Shield or Curb HELB Shielding Structural Support to NNS	Table 3.5-1 Item 7, Table 3.5-1 Item 10, Table 3.5-1 Item 13, Table 3.5-1 Item 15
Lead Shielding Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13, Table 3.5-1 Item 25
Liner Plate	Structural Support to NSR Pressure Boundary or Leak Barrier Radiation Shielding Structural Support to NNS	Table 3.5-1 Item 12, Table 3.5-1 Item 13, Table 3.5-2 Item 6
Metal Partition Walls	Shelter/protection to NSR	Table 3.5-1 Item 13
Metal Siding	Shelter/protection to NSR	Table 3.5-1 Item 13
Missile Shields	Shelter/protection to NSR Structural Support to NSR Spray Shield or Curb Missile Barrier HELB Shielding Structural Support to NNS	Table 3.5-1 Item 7, Table 3.5-1 Item 10, Table 3.5-1 Item 15
Penetrations (Mechanical and Electrical)	Shelter/protection to NSR Pressure Boundary or Leak Barrier	Table 3.5-1 Item 1, Table 3.5-1 Item 2, Table 3.5-1 Item 3, Table 3.5-1 Item 13
Personnel Air Lock	Rated Fire Barrier Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding HELB Shielding	Table 3.5-1 Item 4, Table 3.5-1 Item 5, Table 3.5-1 Item 13

Table 2.4-2:
REACTOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Pipe Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13, Table 3.5-1 Item 28, Table 3.5-2 Item 4
Pipe Whip Restraint	Structural Support to NSR HELB Shielding Structural Support to NNS Pipe Whip Restraint	Table 3.5-1 Item 13, Table 3.5-1 Item 28
Post-Tensioning System	Structural Support to NSR	Table 3.5-1 Item 11, Table 3.5-1 Item 14
Refueling Canal Liner Plate	Pressure Boundary or Leak Barrier	Table 3.5-1 Item 12
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Spray Shield or Curb Radiation Shielding Missile Barrier HELB Shielding Structural Support to NNS Pipe Whip Restraint	Table 3.5-1 Item 7, Table 3.5-1 Item 10, Table 3.5-1 Item 15, Table 3.5-1 Item 16
Seismic Joint Filler	Rated Fire Barrier Structural Support to NSR Flood Protection Barrier	Table 3.5-1 Item 6
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 13, Table 3.5-1 Item 25

Table 2.4-2:
REACTOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Structural Steel – beams, columns, plates, trusses	Shelter/protection to NSR Structural Support to NSR Missile Barrier HELB Shielding Structural Support to NNS Pipe Whip Restraint	**Table 3.5-1 Item 13**
Sump Screens	Structural Support to NSR Flood Protection Barrier	**Table 3.5-1 Item 13**
Sumps	Structural Support to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding	**Table 3.5-1 Item 12**
Tube Track	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 13**

2.4.2 OTHER STRUCTURES

The following structures are included in Section 2.4.2 (for plant location see **Site Facilities Drawing**):

- Auxiliary Building (including Refueling Water Storage Tank and Reactor Make-up Water Storage Tank foundations, West Penetration Access Area, and Hot Machine Shop) (**Section 2.4.2.1**)
- Control Building (**Section 2.4.2.2**)
- Diesel Generator Building (**Section 2.4.2.3**)
- Fuel Handling Building (**Section 2.4.2.4**)
- Intermediate Building (including East Penetration Access Area) (**Section 2.4.2.5**)
- Turbine Building (**Section 2.4.2.6**)
- Service Water Pumphouse, Intake and Discharge Structures (**Section 2.4.2.7**)
- Yard Structures (Condensate Storage Tank Foundation, Electrical Manhole MH-2, Fire Service Pumphouse, Service Water Pond Dams and West Embankment, and North Berm) (**Section 2.4.2.8**)

Note that waterstops are used in safety related structures at construction joints located below grade to inhibit the intrusion of groundwater. Waterstops and waterproofing membrane are inaccessible and considered to be subcomponents of the concrete walls and slabs.

The intended functions described for each structure apply to the structure and/or structure components.

2.4.2.1 Auxiliary Building

The foundation system for the Auxiliary Building, described in **FSAR Section 3.8.5.1.2** [**Reference 2.4-2**], consists of a four foot thick structural reinforced concrete mat supported by fill concrete extending down to competent rock. A waterproofing membrane is provided between the fill concrete and the structural mat.

The Auxiliary Building is a Seismic Category I structure described in **FSAR Section 3.8.4.1.2**. The main Auxiliary Building superstructure is a reinforced concrete shear wall (box type) structure whose foundation is comprised of a reinforced concrete structural mat. The exterior walls are reinforced concrete designed to prevent damage to safety related equipment from design basis events such as seismic and tornado generated missiles.

The Auxiliary Building is designed to withstand the various combinations of dead and live loads, design basis event loads, and other generic design criteria loads as defined in the FSAR [**Reference 2.4-2**].

The southwestern portion of the Auxiliary Building supports two large tanks, the Refueling Water Storage Tank and the Reactor Make-up Water Storage Tank. Concrete retaining walls provide compartmentalization protection from tornado generated missiles.

The southeastern portion of the Auxiliary Building is designated the West Penetration Access Area (WPAA), which houses the containment personnel airlock (the emergency airlock connects to the Fuel Handling Building). The WPAA utilizes structural steel framing to support the floor slabs up to the elevation of the roof.

The Hot Machine Shop is a non-Seismic Category I structure located just north of the Auxiliary Building. The Hot Machine Shop is a steel framed building with metal siding designed to withstand earthquake loads and tornado wind loads to the extent required for prevention of damage to Seismic Category I structures. The north wall of the Auxiliary Building is separated from the Hot Machine Shop by a seismic gap. The failure of the Hot Machine Shop will not prevent the satisfactory accomplishment of any required safety-related functions. The Hot Machine Shop is therefore not subject to an aging management review.

The Auxiliary Building is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides shelter/protection to safety-related components.
- Provides structural and/or functional support to safety-related equipment.
- Provides flood protection barrier (internal and external flooding event).
- Provides pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events.
- Provides spray shield or curbs for directing flow.
- Provides shielding against radiation.
- Provides missile barrier (internally or externally generated).
- Provides shielding against high-energy line breaks.
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.
- Provides pipe whip restraint.

A complete list of Auxiliary Building structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-3.

Table 2.4-3:
AUXILIARY BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Bellows (RHR and Reactor Building Spray system isolation valve chambers and guard pipe)	Structural Support to NSR Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-2 Item 6
Blowout or Blow-off Panels	Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16
Cable Tray & Conduit	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Compressible Joints & Seals	Flood Protection Barrier Pressure Boundary or Leak Barrier	Table 3.5-1 Item 6
Crane Rails & Girders	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Duct Banks	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Electrical and Instrument Panels & Enclosures	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16

Table 2.4-3:
AUXILIARY BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Equipment Component Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Equipment Pads	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23, Table 3.5-1 Item 25
Expansion Anchors	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19
Fire Barriers (Walls, Ceilings and Floors	Rated Fire Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 16
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Flood, Pressure and Specialty Doors	Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16, Table 3.5-2 Item 5
Foundations	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23

Table 2.4-3:
AUXILIARY BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Hatches (Concrete)	Rated Fire Barrier Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Missile Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23
HVAC Duct Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Instrument Line Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Instrument Racks & Frames	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Jet Barriers	HELB Shielding	Table 3.5-1 Item 16
Lead Shielding Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Liner Plate	Shelter/protection to NSR Structural Support to NSR Pressure Boundary or Leak Barrier Radiation Shielding	Table 3.5-1 Item 16
Masonry Block, Brick Walls, or Knockdown Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Radiation Shielding Structural Support to NNS	Table 3.5-1 Item 20
Metal Spray Shields	Spray Shield or Curb	Table 3.5-1 Item 16

Table 2.4-3:
AUXILIARY BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Missile Shields	Missile Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 27
Pipe Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 26, Table 3.5-1 Item 27, Table 3.5-1 Item 28
Pipe Whip Restraint	Structural Support to NSR HELB Shielding Structural Support to NNS Pipe Whip Restraint	Table 3.5-1 Item 16, Table 3.5-1 Item 26, Table 3.5-1 Item 28
Reinforced Concrete - Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding Missile Barrier HELB Shielding Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Roof Slabs	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Pressure Boundary or Leak Barrier Radiation Shielding Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Seismic Joint Filler	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 6

Table 2.4-3:
AUXILIARY BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 25,** **Table 3.5-1 Item 27**
Structural Steel – beams, columns, plates, trusses	Shelter/protection to NSR Structural Support to NSR Missile Barrier HELB Shielding Structural Support to NNS Pipe Whip Restraint	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 25,** **Table 3.5-1 Item 27**
Sumps	Structural Support to NSR Flood Protection Barrier Structural Support to NNS	**Table 3.5-1 Item 16**
Tube Track	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16**

2.4.2.2 Control Building

The foundation system for the Control Building, described in **FSAR Section 3.8.5.1.4** [**Reference 2.4-2**], consists of a reinforced concrete mat designed to transfer vertical load from superstructure columns to fill concrete extending down to competent rock. Vertical reinforcing steel extends from exterior shear walls into fill concrete.

The Control Building is a Seismic Category I structure described in **FSAR Section 3.8.4.1.5**. The superstructure is a steel frame structure with concrete exterior shear walls containing four main floor levels and a concrete roof, and is designed to withstand the various combinations of dead and live loads, design basis event loads, and other generic design criteria loads as defined in the FSAR.

The Control Building is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.

- Provides shelter/protection to safety-related components.
- Provides structural and/or functional support to safety-related equipment.
- Provides flood protection barrier (internal and external flooding event).
- Provides pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events.
- Provides curbs for directing flow.
- Provides shielding against radiation.
- Provides missile barrier (internally or externally generated).
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.

A complete list of Control Building structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-4.

Table 2.4-4:
CONTROL BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27**
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27**
Cable Tray & Conduit	Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16**
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27**
Checkered Plate	Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16**

Table 2.4-4:
CONTROL BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Compressible Joints & Seals	Flood Protection Barrier Pressure Boundary or Leak Barrier	Table 3.5-1 Item 6
Control Boards and Panels	Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Control Room Ceiling	Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Crane Rails & Girders	Structural Support to NNS	Table 3.5-1 Item 16
Duct Banks	Rated Fire Barrier Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Electrical and Instrument Panels & Enclosures	Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Equipment Component Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Equipment Pads	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23, Table 3.5-1 Item 25
Expansion Anchors	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19

Table 2.4-4:
CONTROL BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 16
Flood Barriers (Elastomers)	Flood Protection Barrier	Table 3.5-2 Item 3
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Flood, Pressure and Specialty Doors	Shelter/protection for NSR Flood Protection Barrier Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16, Table 3.5-2 Item 5
Foundations	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Hatches (Concrete)	Rated Fire Barrier Shelter/protection for NSR Flood Protection Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23
HVAC Duct Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Instrument Line Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Instrument Racks & Frames	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16

Table 2.4-4:
CONTROL BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Lead Shielding Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Masonry Block, Brick Walls, or Knockdown Walls	Rated Fire Barrier Shelter/protection for NSR Structural Support to NSR Radiation Shielding Structural Support to NNS	Table 3.5-1 Item 20
Metal Partition Walls	Rated Fire Barrier	Table 3.5-1 Item 16
Missile Shields	Missile Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 27
Pipe Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 26, Table 3.5-1 Item 27, Table 3.5-1 Item 28
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection for NSR Structural Support to NSR Flood Protection Barrier Radiation Shielding Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Roof Slabs	Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Seismic Joint Filler	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 6
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27

Table 2.4-4:
CONTROL BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Structural Steel – beams, columns, plates, trusses	Shelter/protection for NSR Structural Support to NSR Missile Barrier Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 25,** **Table 3.5-1 Item 27**
Tube Track	Shelter/protection for NSR Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16**

2.4.2.3 Diesel Generator Building

The foundation system for the Diesel Generator Building, described in **FSAR Section 3.8.5.1.6** [**Reference 2.4-2**], consists of a reinforced concrete slab and grade beam system that is supported by reinforced concrete caissons drilled into competent bedrock. The foundations for the diesel generators extend from the operating floor level down to the basement floor mat.

The Diesel Generator Building is a Seismic Category I structure described in **FSAR Section 3.8.4.1.4**. The superstructure is a reinforced concrete shear wall (box type) structure containing three main floor levels above the foundation mat, and a roof, designed to withstand the various combinations of dead and live loads, OBE and SSE seismic loads, wind loads, tornado loads, and other generic design criteria loads as defined in the FSAR. The entire building is separated from other buildings to prevent load transfer during an OBE or SSE. The primary function of the Diesel Generator Building is to house the diesel generators that are needed to supply emergency onsite power in the event that offsite power is lost. The Diesel Generator Building is designed to withstand the various combinations or dead and live loads, design basis event loads, and other generic design criteria loads as defined in the FSAR.

The Diesel Generator Building is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides shelter/protection to safety-related components.
- Provides structural and/or functional support to safety-related equipment.

- Provides flood protection barrier (internal and external flooding event).
- Provides missile barrier (internally or externally generated).
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.

A complete list of Diesel Generator Building structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-5.

Table 2.4-5:
DIESEL GENERATOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Cable Tray & Conduit	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Caissons	Structural Support to NSR Structural Support to NNS	Table 3.5-2 Item 2
Compressible Joints & Seals	Flood Protection Barrier	Table 3.5-1 Item 6
Crane Rails & Girders	Structural Support to NNS	Table 3.5-1 Item 16
Duct Banks	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Electrical and Instrument Panels & Enclosures	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16

Table 2.4-5:
DIESEL GENERATOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Component Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Pads	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23, Table 3.5-1 Item 25
Expansion Anchors	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 16
Flood Barrier (Elastomers)	Flood Protection Barrier	Table 3.5-2 Item 3
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Flood, Pressure and Specialty Doors	Shelter/protection to NSR Flood Protection Barrier	Table 3.5-1 Item 16, Table 3.5-2 Item 5
Foundations	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Grating	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 27

Table 2.4-5:
DIESEL GENERATOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Hatches (Steel)	Rated Fire Barrier Shelter/protection to NSR Missile Barrier	Table 3.5-1 Item 16
HVAC Duct Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Instrument Line Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Instrument Racks & Frames	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Metal Partition Walls	Rated Fire Barrier	Table 3.5-1 Item 16
Missile Shields	Missile Barrier	Table 3.5-1 Item 16
Pipe Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Flood Protection Barrier Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Roof Slabs	Rated Fire Barrier Shelter/protection to NSR Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Seismic Joint Filler	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 6
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Structural Steel – beams, columns, plates, trusses	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25

Table 2.4-5:
DIESEL GENERATOR BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Sumps	Structural Support to NSR Flood Protection Barrier Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 23**

2.4.2.4 Fuel Handling Building

The foundation system for the Fuel Handling Building, described in **FSAR Section 3.8.5.1.5** [**Reference 2.4-2**], consists of a reinforced concrete mat formed by the bottom of the Spent Fuel Pool and Fuel Cask Pit and supported by reinforced concrete piers that extend to the fill concrete adjacent to the Reactor and Auxiliary Buildings, and by reinforced concrete caissons that extend to competent rock on the north and east sides.

The Fuel Handling Building is a Seismic Category I structure discussed in **FSAR Section 3.8.4.1.6**. The superstructure is a steel frame structure containing two main floor levels and a roof, designed to withstand the various combinations of dead and live loads, design basis event loads, and other generic design criteria loads as defined in the FSAR.

The Fuel Handling Building is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides shelter/protection to safety-related components.
- Provides structural and/or functional support to safety-related equipment.
- Provides flood protection barrier (internal and external flooding event).
- Provides pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events.
- Provides curbs for directing flow.
- Provides shielding against radiation.
- Provides missile barrier (internally or externally generated).
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.

A complete list of Fuel Handling Building structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-6.

Table 2.4-6:
FUEL HANDLING BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Cable Tray & Conduit	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Caissons	Structural Support to NSR Structural Support to NNS	Table 3.5-2 Item 2
Checkered Plate	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 27
Compressible Joints & Seals	Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding	Table 3.5-1 Item 6
Crane Rails & Girders	Structural Support to NSR	Table 3.5-1 Item 16
Electrical and Instrument Panels & Enclosures	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27

Table 2.4-6:
FUEL HANDLING BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Equipment Component Supports	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 25,** **Table 3.5-1 Item 27**
Equipment Pads	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 23,** **Table 3.5-1 Item 25**
Expansion Anchors	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 25,** **Table 3.5-1 Item 27**
Fire Barrier Penetration Seals	Rated Fire Barrier	**Table 3.3-1 Item 19**
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 17,** **Table 3.5-1 Item 23**
Fire Doors	Rated Fire Barrier	**Table 3.3-1 Item 19,** **Table 3.5-1 Item 16**
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 23**
Foundations	Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 17,** **Table 3.5-1 Item 21,** **Table 3.5-1 Item 22,** **Table 3.5-1 Item 23**
Fuel Transfer Canal Liner Plate	Pressure Boundary or Leak Barrier	**Table 3.5-1 Item 19**
Hatches (Concrete)	Rated Fire Barrier Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Missile Barrier	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 23**

Table 2.4-6:
FUEL HANDLING BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Hatches (Steel)	Rated Fire Barrier Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 27
HVAC Duct Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Instrument Line Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Instrument Racks & Frames	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Lead Shielding Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Masonry Block, Brick Walls, or Knockdown Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Radiation Shielding Structural Support to NNS	Table 3.5-1 Item 20
Metal Siding	Shelter/protection to NSR	Table 3.5-1 Item 16
Missile Shields	Missile Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 27
Neutron Absorbing Sheets in Spent Fuel Pool - Boraflex	Structural Support to NSR	Table 3.3-1 Item 9, Table 3.3-1 Item 12

Table 2.4-6:
FUEL HANDLING BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Piers (Concrete)	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Pipe Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Roof	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Seismic Joint Filler	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 6
Spent Fuel Pool Liner	Pressure Boundary or Leak Barrier	Table 3.5-1 Item 19, Table 3.5-1 Item 27
Spent Fuel Storage Rack	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27

**Table 2.4-6:
FUEL HANDLING BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Intended Functions	AMR Results
Structural Steel – beams, columns, plates, trusses	Shelter/protection to NSR Structural Support to NSR Missile Barrier Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 25,** **Table 3.5-1 Item 27**
Sumps	Structural Support to NSR Flood Protection Barrier Structural Support to NNS	**Table 3.5-1 Item 16,** **Table 3.5-1 Item 23**
Tube Track	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	**Table 3.5-1 Item 16**

2.4.2.5 Intermediate Building

The foundation system for the Intermediate Building, described in **FSAR Section 3.8.5.1.3** [**Reference 2.4-2**], consists of a reinforced concrete basement floor slab that acts in conjunction with a series of grade beams to transfer vertical loads to the reinforced concrete caissons, shear/bearing walls, and concrete piers. The shear/bearing wall foundations and reinforced concrete caissons are founded on competent bedrock. The piers are founded on fill concrete that extends beyond the Reactor Building and Auxiliary Building. Horizontal shears are transferred through the basement floor slab to the shear/bearing walls and to the Control Building base mat.

The Intermediate Building is a Seismic Category I structure described in **FSAR Section 3.8.4.1.3**. The superstructure is an L-shaped reinforced concrete shear wall (box type) structure containing two main floor levels above the foundation and extending up to the low roof. Above the low roof is a partial third floor of reinforced concrete and a high roof. The Intermediate Building is designed to withstand the various combinations of dead and live loads, design basis event loads, and other generic design criteria loads as defined in the FSAR.

The Intermediate Building is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides shelter/protection to safety-related components.

- Provides structural and/or functional support to safety-related equipment.
- Provides flood protection barrier (internal and external flooding event).
- Provides pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events.
- Provides spray shield or curbs for directing flow.
- Provides shielding against radiation.
- Provides missile barrier (internally or externally generated).
- Provides shielding against high-energy line breaks.
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.
- Provides pipe whip restraint.

A complete list of Intermediate Building structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-7.

Table 2.4-7:
INTERMEDIATE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Battery Racks	Structural Support to NSR	Table 3.5-2 Item 1
Blowout or Blow-off Panels	Structural Support to NSR	Table 3.5-1 Item 16
Cable Tray & Conduit	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27

**Table 2.4-7:
INTERMEDIATE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Intended Functions	AMR Results
Caissons	Structural Support to NSR Structural Support to NNS	Table 3.5-2 Item 2
Compressible Joints & Seals	Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding	Table 3.5-1 Item 6
Crane Rails & Girders	Structural Support to NNS	Table 3.5-1 Item 16
Duct Banks	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Electrical and Instrument Panels & Enclosures	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Equipment Component Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Equipment Pads	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23, Table 3.5-1 Item 25
Expansion Anchors	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19
Fire Barriers (Walls, Ceilings and Floors	Rated Fire Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23

Table 2.4-7:
INTERMEDIATE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 16
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Flood, Pressure and Specialty Doors	Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16, Table 3.5-2 Item 5
Foundations	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Hatches (Concrete)	Rated Fire Barrier Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Missile Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Hatches (Steel)	Rated Fire Barrier Shelter/protection to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 27
HVAC Duct Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Instrument Line Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27

Table 2.4-7:
INTERMEDIATE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Instrument Racks & Frames	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Jet Barriers	HELB Shielding	Table 3.5-1 Item 16, Table 3.5-1 Item 23, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Lead Shielding Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Metal Siding	Shelter/protection to NSR	Table 3.5-1 Item 16
Metal Spray Shields	Spray Shield or Curb	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Missile Shields	Missile Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Piers	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Pipe Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Pipe Whip Restraint	Structural Support to NSR HELB Shielding Structural Support to NNS Pipe Whip Restraint	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27

Table 2.4-7: INTERMEDIATE BUILDING COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Flood Protection Barrier Pressure Boundary or Leak Barrier Radiation Shielding Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Roof Slabs	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Pressure Boundary or Leak Barrier Radiation Shielding Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Seismic Joint Filler	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 6
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Structural Steel – beams, columns, plates, trusses	Shelter/protection to NSR Structural Support to NSR Missile Barrier HELB Shielding Structural Support to NNS Pipe Whip Restraint	Table 3.5-1 Item 16, Table 3.5-1 Item 25, Table 3.5-1 Item 27
Sumps	Structural Support to NSR Flood Protection Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23

Table 2.4-7:
INTERMEDIATE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Trenches	Shelter/protection to NSR Flood Protection Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Tube Track	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16

2.4.2.6 Turbine Building

The foundation mat for the Turbine Building is comprised of a reinforced concrete mat supported by Zone III fill (graded crushed stone) material. The reinforced concrete pedestal foundation mats for the Feedwater Pumps and Turbine Generators are founded on fill concrete over bedrock.

The Turbine Building is a non-Seismic Category I structure as described in **FSAR Section 3.8.4.1.1** [**Reference 2.4-2**]. The superstructure of steel framing, metal siding and metal roof deck is supported on a reinforced concrete substructure. The steel rigid frame structure is elastically supported at the operating floor, which acts as a diaphragm. The subsurface portion of the east, west and south walls are reinforced concrete. The north wall is structural steel framing, with no siding, that abuts the Control, Intermediate, and Diesel Buildings. The entire building is separated from other buildings to prevent load transfer during seismic events.

The Turbine Building is designed to withstand the various combinations of dead and live loads, seismic loads, wind loads, tornado loads, and other generic design criteria loads as defined in the FSAR. However, for earthquake loads and tornado wind loads, the Turbine Building is only designed to the extent required to prevent damage to Seismic Category I structures. The primary function of the Turbine Building is to house the turbine generators. The functional requirement of the building in the event of an earthquake or tornado is that no portion of the building collapses and results in damage to Seismic Category I structures.

The Turbine Building is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.

- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.

A complete list of Turbine Building structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-8.

Table 2.4-8:
TURBINE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Anchorage / Embedments (exposed surfaces)	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Cable Tray & Conduit	Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Compressible Joints & Seals	Flood Protection Barrier	Table 3.5-1 Item 6
Crane Rails & Girders	Structural Support to NNS	Table 3.5-1 Item 16
Duct Banks	Rated Fire Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Electrical and Instrument Panels & Enclosures	Structural Support to NNS	Table 3.5-1 Item 16
Embedments	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Component Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Pads	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23, Table 3.5-1 Item 25

Table 2.4-8:
TURBINE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Expansion Anchors	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 16
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Flood, Pressure and Specialty Doors	Pressure Boundary or Leak Barrier	Table 3.5-1 Item 16, Table 3.5-2 Item 5
Foundations	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Grating	Structural Support to NNS	Table 3.5-1 Item 16
Hatches (Concrete)	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Hatches (Steel)	Structural Support to NNS	Table 3.5-1 Item 16
HVAC Duct Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Instrument Line Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Instrument Racks & Frames	Structural Support to NNS	Table 3.5-1 Item 16
Masonry Block, Brick Walls, or Knockdown Walls	Rated Fire Barrier Structural Support to NNS	Table 3.5-1 Item 20

Table 2.4-8:
TURBINE BUILDING
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Metal Siding	Structural Support to NNS	Table 3.5-1 Item 16
Pipe Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Flood Protection Barrier Missile Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Roof	Structural Support to NNS	Table 3.5-1 Item 16
Seismic Joint Filler	Structural Support to NNS	Table 3.5-1 Item 6
Stair, Platform, & Grating Support	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Structural Steel – beams, columns, plates, trusses	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Sumps	Flood Protection Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Trenches	Flood Protection Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23

2.4.2.7 Service Water Pumphouse, Intake And Discharge Structures

2.4.2.7.1 Service Water Pumphouse

The foundation for the Service Water Pumphouse, described in FSAR Section 3.8.5.1.7 [Reference 2.4-2], consists of a reinforced concrete structural mat. The discharge pipe pits on the south side and the control areas on the west side of the Service Water Pumphouse are supported by buried reinforced concrete columns which extend to the supporting foundation mat. The entire structural mat is supported on compact fill that is in turn supported on a layer of in-situ soils (saprolite), then decomposed rock down to competent rock.

The Service Water Pumphouse is a Seismic Category I structure described in **FSAR Section 3.8.4.1.7**. The superstructure is a reinforced concrete building separated from the Service Water Intake Structure and from buried connecting pipes and electrical duct banks by flexible joints, which accommodate relative settlement and seismic movement.

The Service Water Pumphouse is designed to withstand the various combinations of dead and live loads, OBE and SSE seismic loads, wind loads, tornado loads, and other generic design criteria loads as defined in the FSAR. The primary function of the Service Water Pumphouse is to house the service water pumps that pump water from the Service Water Pond to supply the service water system. The Service Water Pumphouse is designed to withstand the various combinations of dead and live loads, design basis event loads, and other generic design criteria loads as defined in the FSAR.

The Service Water Pumphouse is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides shelter/protection to safety-related components.
- Provides structural and/or functional support to safety-related equipment.
- Provides flood protection barrier (internal and external flooding event).
- Provides spray shield or curbs for directing flow.
- Provides missile barrier (internally or externally generated).
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.
- Provides source of cooling water for plant shutdown.

A complete list of Service Water Pumphouse and Intake and Discharge Structures structural components subject to aging management review, along with their component intended functions, is provided in **Table 2.4-9**.

2.4.2.7.2 Service Water Intake And Discharge Structures

Service Water Intake Structure

The foundation for the Service Water Intake Structure, described in **FSAR Section 3.8.5.1.8** [**Reference 2.4-2**], consists of a reinforced concrete mat supported by compacted fill material, except for a portion of the inlet end, which rests on in-situ soils.

The Service Water Intake Structure is a Seismic Category I structure as described in **FSAR Section 3.8.4.1.8**. The structure is a reinforced concrete rectangular box culvert with two reinforced concrete wing walls at the intake end. The foundation mat forms the floor of the structure. An expansion joint separates the Service Water Intake Structure from the Service Water Pumphouse, which accommodates relative settlement and seismic movement. The structure extends into the Service Water Pond and is mostly buried in the West Embankment except for the intake end, which is submerged within the pond.

The Service Water Intake Structure is designed to withstand the various combinations of dead loads, OBE and SSE seismic loads, and other generic design criteria loads as defined in the FSAR. The primary function of the Service Water Intake Structure is to extend the point at which water is drawn from the Service Water Pond into the Service Water Pumphouse. The functional requirement of the Service Water Intake Structure during and following a design basis event is that it does not collapse and result in a loss of supply water from the Service Water Pond to the Service Water Pumphouse.

Service Water Discharge Structure

The foundation for the Service Water Discharge Structure, described in **FSAR Section 3.8.5.1.9**, consists of a reinforced concrete mat that bears partly on decomposed rock and partly on fill concrete that extends to the decomposed rock.

The Service Water Discharge Structure is a Seismic Category I structure as described in **FSAR Section 3.8.4.1.9**. The structure is a reinforced concrete rectangular basin mostly buried in the West Embankment of the Service Water Pond. The foundation mat forms the floor of the basin. A 15-foot high abutment wall forms the west end of the basin, and a 3-foot high sill wall forms the east end. Wing walls form the north and south sides of the basin. Two 30-inch diameter service water pipes terminate at the abutment wall and are connected to the Service Water Discharge Structure by flexible connections.

The Service Water Discharge Structure is designed to withstand the various combinations of dead loads, OBE and SSE seismic loads, and other generic design criteria loads as defined in the FSAR. The primary function of the Service Water Discharge Structure is to release service water into the Service Water Pond. The functional requirement of the Service Water Discharge Structure during and following a design basis event is that it does not collapse and result in an interruption of service water discharge.

Intended Functions

The Service Water Intake and Discharge Structures are in the scope of license renewal because they:

- Provide source of cooling water for plant shutdown.

A complete list of Service Water Pumphouse and Intake and Discharge Structures structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-9.

Table 2.4-9:
SERVICE WATER PUMPHOUSE, INTAKE AND DISCHARGE STRUCTURES COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25, Table 3.5-2 Item 8
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25, Table 3.5-2 Item 8
Cable Tray & Conduit	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25
Checkered Plate	Shelter/protection to NSR	Table 3.5-1 Item 18
Compressible Joints & Seals	Flood Protection Barrier	Table 3.5-1 Item 6
Crane Rails & Girders	Structural Support to NNS	Table 3.5-1 Item 18
Duct Banks	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-2 Item 9
Electrical and Instrument Panels & Enclosures	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25, Table 3.5-2 Item 8

Table 2.4-9:
SERVICE WATER PUMPHOUSE, INTAKE AND DISCHARGE STRUCTURES COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Equipment Component Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25
Equipment Pads	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25
Expansion Anchors	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25, Table 3.5-2 Item 8
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	Table 3.5-1 Item 18, Table 3.5-2 Item 9
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 18
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 18
Flood, Pressure and Specialty Doors	Shelter/protection to NSR Flood Protection Barrier	Table 3.5-1 Item 18
Foundations	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 22, Table 3.5-2 Item 7, Table 3.5-2 Item 9
Grating	Shelter/protection to NSR	Table 3.5-1 Item 18
Hatches (Concrete)	Rated Fire Barrier Shelter/protection to NSR Missile Barrier	Table 3.5-1 Item 18
HVAC Duct Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25
Instrument Line Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25

Table 2.4-9:
SERVICE WATER PUMPHOUSE, INTAKE AND DISCHARGE STRUCTURES COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Instrument Racks & Frames	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18
Intake Bays or Canals	Structural Support to NSR Source of Cooling Water	Table 3.5-1 Item 18
Intake Screens	Structural Support to NSR Source of Cooling Water	Table 3.5-1 Item 18, Table 3.5-2 Item 8
Missile Shields	Missile Barrier	Table 3.5-1 Item 18
Pipe Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25, Table 3.5-2 Item 8
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Flood Protection Barrier Structural Support to NNS Source of Cooling Water	Table 3.5-1 Item 18, Table 3.5-2 Item 7, Table 3.5-2 Item 9
Roof Slabs	Rated Fire Barrier Shelter/protection to NSR Missile Barrier Structural Support to NNS	Table 3.5-1 Item 18
Seismic Joint Filler	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 6
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25
Structural Steel – beams, columns, plates, trusses	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 18, Table 3.5-1 Item 25
Sumps	Structural Support to NSR Flood Protection Barrier Structural Support to NNS	Table 3.5-1 Item 18

2.4.2.8 Yard Structures

The following structures are included in Yard Structures:

- Condensate Storage Tank Foundation (**Section 2.4.2.8.1**)
- Fire Service Pumphouse (**Section 2.4.2.8.2**)
- Electrical Manhole MH-2 (**Section 2.4.2.8.3**)
- Earthen Embankments (Service Water Pond Dams, West Embankment, North Berm) (**Section 2.4.2.8.4**)
- Electrical Substation and Relay House (**Section 2.4.2.8.5**)

2.4.2.8.1 Condensate Storage Tank Foundation

The foundation for the Condensate Storage Tank is designed to satisfy Seismic Category I requirements as defined in **FSAR Sections 2.5.4.10.3** and **9.2.6** [**Reference 2.4-2**]. The foundation consists of a reinforced concrete mat supported by Zone III (graded crushed stone) fill material and an integral reinforced concrete ring wall that extends above the top of the foundation mat. The Condensate Storage Tank is secured to the foundation by anchor bolts embedded in the ring wall. The interior area of the ring wall is filled with clean dry sand to form a sand mat beneath the tank. A reinforced concrete valve pit for the Condensate Storage Tank drainpipe is integrated into the south side of the foundation.

The primary function of the Condensate Storage Tank foundation is to support the nuclear safety-related Condensate Storage Tank. The functional requirement of the foundation during and following a design basis event is that its failure would not result in a loss of the Condensate Storage Tank contents.

The Condensate Storage Tank foundation is in the scope of license renewal because it:

- Provides structural and/or functional support to safety-related equipment.

A complete list of Condensate Storage Tank foundation structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-10.

**Table 2.4-10:
YARD STRUCTURES (CONDENSATE STORAGE TANK FOUNDATION)
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS**

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Checkered Plate	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Expansion Anchors	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Foundation Dowels	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Foundations	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22
Instrument Line Supports	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Instrument Racks & Frames	Structural Support to NSR	Table 3.5-1 Item 16
Pipe Supports	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22
Stair, Platform, & Grating Support	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25

2.4.2.8.2 Fire Service Pumphouse

The Fire Service Pumphouse is a concrete block building described in the **Fire Protection Evaluation Report (FPER) Section 4.10** [**Reference 2.4-4**]. The building is founded upon the reinforced concrete Circulating Water Intake Structure. Hollow concrete blocks are used to form the exterior and interior walls of the building, and solid concrete blocks are used under steel framing members. The composite roof is a built-up insulated roof with gravel over steel decking and metal roof trusses.

A reinforced concrete slab, located on the east side of the Fire Service Pumphouse and founded upon the Circulating Water Intake Structure, is the foundation for the diesel engine-driven fire service pump fuel oil tank. The tank is secured to the foundation by embedded anchor bolts.

The primary function of the Fire Service Pumphouse is to house one electric motor-driven fire pump and one diesel engine-driven fire pump.

The Fire Service Pumphouse is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions.

A complete list of Fire Service Pumphouse structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-11.

Table 2.4-11:
YARD STRUCTURES (FIRE SERVICE PUMPHOUSE)
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NNS	**Table 3.5-1 Item 16, Table 3.5-1 Item 25**
Anchorage / Embedments (exposed surfaces)	Structural Support to NNS	**Table 3.5-1 Item 16, Table 3.5-1 Item 25**
Battery Racks	Structural Support to NNS	**Table 3.5-2 Item 1**

Table 2.4-11:
YARD STRUCTURES (FIRE SERVICE PUMPHOUSE)
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Cable Tray & Conduit	Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Electrical and Instrument Panels & Enclosures	Structural Support to NNS	Table 3.5-1 Item 16
Embedments	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Component Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Pads	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Expansion Anchors	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Fire Barrier Penetration Seals	Rated Fire Barrier	Table 3.3-1 Item 19
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Fire Doors	Rated Fire Barrier	Table 3.3-1 Item 19, Table 3.5-1 Item 16
Flood Curbs (Concrete)	Flood Protection Barrier Spray Shield or Curb	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Foundations	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Hatches (Steel)	Structural Support to NNS	Table 3.5-1 Item 16
HVAC Duct Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25

Table 2.4-11:
YARD STRUCTURES (FIRE SERVICE PUMPHOUSE)
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Instrument Line Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Instrument Racks & Frames	Structural Support to NNS	Table 3.5-1 Item 16
Masonry Block, Brick Walls, or Knockdown Walls	Rated Fire Barrier Structural Support to NNS	Table 3.5-1 Item 20
Pipe Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Reinforced Concrete – Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Flood Protection Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Structural Steel – beams, columns, plates, trusses	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Sumps	Flood Protection Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Trenches	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 23

2.4.2.8.3 Electrical Manhole MH-2

The below-grade foundation for Electrical Manhole MH-2 consists of a reinforced concrete mat supported by Zone I and II (earthen) fill material. The reinforced concrete exterior walls are set in from the profile of the foundation mat and extend above finished grade. The manhole is divided into two compartments by a reinforced concrete partition wall installed on the east-west axis. The above grade manhole cover is a reinforced concrete slab, containing two manways with galvanized steel covers for access into the manhole compartments.

Electrical Manhole MH-2 is a non-seismic structure described in **FPER Section 4.9** [**Reference 2.4-4**]. The structure contains nuclear safety-related Class 1E and non-nuclear safety-related electrical cables.

Electrical Manhole MH-2 is in the scope of license renewal because it:

- Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
- Provides shelter/protection to safety-related components.
- Provides structural and/or functional support to safety-related equipment.

A complete list of Electrical Manhole MH-2 structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-12.

Table 2.4-12:
YARD STRUCTURES (ELECTRICAL MANHOLE MH-2)
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Anchorage / Embedments (exposed surfaces)	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Cable Tray & Conduit	Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Duct Banks	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Embedments	Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Fire Barriers (Walls, Ceilings and Floors)	Rated Fire Barrier	Table 3.5-1 Item 16

Table 2.4-12:
YARD STRUCTURES (ELECTRICAL MANHOLE MH-2)
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Foundations	Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 23
Manhole Covers	Rated Fire Barrier Structural Support to NSR	Table 3.5-1 Item 16, Table 3.5-1 Item 21, Table 3.5-1 Item 22
Manholes	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Structural Support to NNS	Table 3.5-1 Item 16
Missile Shields	Shelter/protection to NSR Structural Support to NSR Missile Barrier	Table 3.5-1 Item 16, Table 3.5-1 Item 23
Reinforced Concrete - Beams, Columns, Floor Slabs, Walls	Rated Fire Barrier Shelter/protection to NSR Structural Support to NSR Flood Protection Barrier Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23

2.4.2.8.4 Earthen Embankments

Service Water Pond Dams and West Embankment

Four earthen embankments – three dams (North Dam, South Dam, and East Dam) and the West Embankment – form the Service Water Pond, a safety class impoundment. These homogeneous earth structures are Seismic Category I and are designed to satisfy the intent of Regulatory Guides 1.27 [**Reference 2.4-5**] and 1.29 [**Reference 2.4-6**].

The three dams and the West Embankment, which merges with the west abutments of the North and South Dams, are designed to be stable under static and dynamic conditions, OBE and SSE seismic loads, and for maximum wave run-up from the Monticello Reservoir as defined in **FSAR Section 2.5.6.1** [**Reference 2.4-2**]. The primary function of the earthen structures is to form the Service Water Pond, which provides water for the Service Water

System under normal and emergency conditions. The functional requirement, assuming a loss of the Monticello Reservoir during a design basis event, is that no dam or embankment failure would result in a loss of cooling water to the Service Water System.

North Berm

The shoreline along Monticello Reservoir north of the plant and west of the North Dam has an earthen dike (the North Berm) constructed three feet above site grade. The North Berm is classified as a non-seismic, non-nuclear safety-related structure whose primary function is to protect the site from external flooding. The functional requirement of the North Berm is to protect nuclear safety-related structures and components from any adverse effects due to flooding.

Further description of the North Berm is provided in **FSAR Sections 2.4.3**, **2.4.3.6.2**, and **2.4.10**.

Intended Functions

The North Dam, South Dam, East Dam and West Embankment are in the scope of license renewal because they:

- Provide flood protection barrier.
- Provide source of cooling water for plant shutdown.
- Impound water for ultimate heat sink during loss of Monticello Reservoir.

The North Berm is in the scope of license renewal because it:

- Provides flood protection barrier.

A complete list of earthen embankments structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-13.

Table 2.4-13:
EARTHEN EMBANKMENTS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Service Water Pond Dams (North Dam, South Dam, and East Dam) and West Embankment	Flood Protection Barrier Source of Cooling Water Impound Water	**Table 3.5-1 Item 18, Table 3.5-2 Item 10**

Table 2.4-13:
EARTHEN EMBANKMENTS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
North Berm	Flood Protection Barrier	**Table 3.5-1 Item 18, Table 3.5-2 Item 10**

2.4.2.8.5 Electrical Substation and Transformer Area

Components that are part of the plant system portion of the offsite power grid are within the scope of license renewal in accordance with the SBO scoping criterion, 10 CFR 54.4(a)(3). This power path includes portions of the power path from the power circuit breaker (PCB) in the substation to the safety-related buses. The power path includes (1) portions of the 230 kV substation system, and (2) portions of the Parr-Summer Safeguard 115 kV system. The Electrical Substation provides structural support and/or shelter to components relied on during a station blackout event.

The Electrical Substation yard, located South of the Turbine Building, contains power circuit breakers, transformers, buslines and electrical switching equipment. The transformer area within the site protected area is treated as part of the Electrical Substation for LR purposes.

The entire surface of the substation yard, with the exception of the paved roadways, is covered with several inches of “crusher run” stone and is enclosed by a perimeter fence. Busline and insulator supports are constructed of galvanized structural steel mounted on concrete footings. Power circuit breakers, transformers, and other electrical equipment are supported on concrete pads.

The Electrical Substation and Transformer Area are in the scope of license renewal because they:

- Provide structural support to non-nuclear safety-related components whose failure could prevent satisfactory accomplishment of any of the required regulated events functions.

A complete list of Electrical Substation and Transformer Area structural component types subject to aging management review, along with their component intended functions, is provided in Table 2.4-14.

Table 2.4-14:
YARD STRUCTURES (ELECTRICAL SUBSTATION AND TRANSFORMER AREA) COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Anchorage	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Anchorage / Embedments (exposed surfaces)	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Cable Tray & Conduit	Structural Support to NNS	Table 3.5-1 Item 16
Cable Tray & Conduit Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Electrical and Instrument Panels & Enclosures	Structural Support to NNS	Table 3.5-1 Item 16
Embedments	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Component Supports	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Equipment Pads (Buslines, PCBs, transformers)	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25
Reinforced Concrete - Foundations and Walls	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 17, Table 3.5-1 Item 21, Table 3.5-1 Item 22, Table 3.5-1 Item 23
Structural Steel - beams, columns, plates, trusses (Transmission Towers)	Structural Support to NNS	Table 3.5-1 Item 16, Table 3.5-1 Item 25

2.4.3 REFERENCES

2.4-1	10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 60 FR 22461, May 8, 1995.
2.4-2	VCSNS Final Safety Analysis Report (FSAR), through Amendment 02-01.
2.4-3	10 CFR 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.
2.4-4	VCSNS Fire Protection Evaluation Report (FPER), Amendment 02-01.
2.4-5	Regulatory Guide (RG) 1.27, Ultimate Heat Sink for Nuclear Power Plants, Revision 2.
2.4-6	Regulatory Guide (RG) 1.29, Seismic Design Classification, Revision 2.

2.5 SCOPING AND SCREENING RESULTS: ELECTRICAL AND INSTRUMENTATION AND CONTROL

The determination of electrical and instrumentation and control (I&C) systems and components within the scope of license renewal is made by initially identifying VCSNS electrical systems and then reviewing them to determine which systems and components satisfy one or more of the criteria in 10 CFR 54.4 [**Reference 2.5-1**]. The scoping and screening methodology is described in **Section 2.1** and the results of the plant level electrical and I&C system scoping are presented in **Section 2.2**.

A listing of electrical component commodity groups for the electrical and I&C systems within the scope of license renewal at VCSNS as well as active/passive determinations were developed following the guidance of NEI 95-10, Appendix B [**Reference 2.5-2**]. No commodity groups, beyond those listed in Appendix B of NEI 95-10, were identified for VCSNS. Only the commodity groups that perform a passive function are subject to aging management review.

The passive electrical commodity groups were reviewed to identify those commodity groups or components that are not subject to replacement based on a limited qualified life or specified time period. Most electrical components included in the VCSNS Environmental Qualification (EQ) Program do not meet the long-lived screening criterion of 10 CFR 54.21(a)(1)(ii). Consequently, the insulated cables and connections, terminal blocks, and electrical portions of electrical and I&C penetration assemblies within the scope of the VCSNS EQ program are not subject to an aging management review.

The results of the screening effort identified the following electrical and I&C commodity groups or subgroups that require an aging management review:

- Non-EQ Insulated Cables
- Non-EQ Connectors
- Non-EQ Splices
- Non-EQ Electrical Penetration Assemblies
- Non-EQ Terminal Blocks
- High Voltage Electrical Switchyard Bus
- High Voltage Transmission Conductors and Connections
- High Voltage Insulators

All of the other electrical and I&C commodities identified for VCSNS are either active, are subject to replacement based on a qualified life or specified time period, or do not perform any intended functions and are thus not subject to aging management review.

2.5.1 NON-EQ INSULATED CABLES & CONNECTIONS

The license renewal component intended function of insulated cables and connections is as indicated in Table 2.5-1. Examples of this type of component include power cable, instrument cable, control cable, electrical connectors and splices.

A complete list of the non-EQ insulated cables and connections components subject to aging management review, along with their component intended functions, is provided in Table 2.5-1.

The results of the aging management review of the non-EQ insulated cables and connections are provided in **Sections 3.6.1.1**, **3.6.1.2**, and **3.6.1.3**.

Table 2.5-1:
NON-EQ INSULATED CABLES AND CONNECTIONS COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Electrical Cable	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	**Table 3.6-1 Item 2, Table 3.6-1 Item 3, Table 3.6-1 Item 4, Table 3.6-2 Item 1**
Connectors	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	**Table 3.6-1 Item 2, Table 3.6-1 Item 3, Table 3.6-1 Item 5**
Splices	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	**Table 3.6-1 Item 2, Table 3.6-1 Item 3**

2.5.2 NON-EQ ELECTRICAL PENETRATION ASSEMBLIES

The license renewal component intended function of electrical penetration assemblies is to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals. The electrical penetration assemblies are evaluated in both the electrical area and the civil/structural area. The electrical evaluation is focused on the penetration assemblies as they carry the electrical current while the civil/structural evaluation reviews the structural design with respect to the containment pressure boundary.

A complete list of the non-EQ electrical penetration assembly component types subject to aging management review, along with their component intended functions, is provided in Table 2.5-2.

The results of the aging management review of the non-EQ Electrical Penetration Assemblies are provided in **Section 3.6.1.4**.

Table 2.5-2:
NON-EQ ELECTRICAL PENETRATION ASSEMBLIES COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Electrical Penetrations	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	**Table 3.6-2 Item 2**

2.5.3 NON-EQ TERMINAL BLOCKS

The license renewal component intended function of terminal blocks is to provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.

The results of the aging management review of the non-EQ terminal blocks are provided in **Section 3.6.1.5**.

A complete list of the non-EQ terminal block components subject to aging management review, along with their component intended functions, is provided in Table 2.5-3.

Table 2.5-3:
NON-EQ TERMINAL BLOCKS
COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
Terminal Block	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	**Table 3.6-1 Item 2, Table 3.6-1 Item 3**

2.5.4 HIGH VOLTAGE ELECTRICAL SWITCHYARD EQUIPMENT

The License Renewal component intended function of High Voltage electrical switchyard equipment is to maintain proper electrical continuity or insulation properties within the plant systems portion of the offsite power grid. The boundary of the plant systems portion of the offsite power grid, for the purposes of license renewal for the 230 KV substation Bus 1, ends at the first 230 KV power circuit breaker and at the circuit switchers for the 115 KV ESF line.

The results of the aging management review of the in-scope high voltage electrical switchyard equipment are provided in **Section 3.6.1.6**, **3.6.1.7**, and **3.6.1.8**.

A complete list of the HV electrical switchyard equipment subject to aging management review along with their component intended functions is provided in Table 2.5-4.

Table 2.5-4:
HIGH VOLTAGE ELECTRICAL SWITCHYARD EQUIPMENT COMPONENT TYPES SUBJECT TO AGING MANAGEMENT REVIEW AND THEIR INTENDED FUNCTIONS

Component Type	Intended Functions	AMR Results
High Voltage Electrical Switchyard Bus	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.6-2 Item 3
High Voltage Transmission Conductors and Connections	To provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals.	Table 3.6-2 Item 4
High Voltage Insulators	To insulate and support an electrical conductor.	Table 3.6-2 Item 5

2.5.5 REFERENCES

2.5-1	10 CFR Part 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, 60 FR 22461, May 8, 1995.
2.5-2	NEI 95-10, Revision 3, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, March 2001.

SECTION 3 - AGING MANAGEMENT REVIEW

3.0 AGING MANAGEMENT REVIEW RESULTS

For those structures and components that are identified as being subject to an aging management review, 10 CFR 54.21(a)(3) requires demonstration that the effects of aging will be adequately managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

This section describes the results of the aging management reviews of the components and structures determined, during the scoping and screening processes, to be subject to an aging management review. During the screening process, some structures and components (SCs) were incorporated into commodity groups based on similarity of their design or materials of construction. Use of commodity groups made it possible to address an entire group of SCs with a single evaluation. In the aging management reviews described in the following sections, further definition of commodity groups was performed based on design, material, environmental, and functional characteristics in order to disposition an entire group with a single aging management review.

Organization of this chapter parallels Chapter 3, “Aging Management Review Results” of NUREG-1800, Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (NUREG-1800), dated April 2001. The major sections of this Chapter are:

- 3.1 Aging Management of Reactor Vessel, Internals, and Reactor Coolant System
- 3.2 Aging Management of Engineered Safety Features
- 3.3 Aging Management of Auxiliary Systems
- 3.4 Aging Management of Steam and Power Conversion Systems
- 3.5 Aging Management of Containments, Structures, and Component Supports
- 3.6 Aging Management of Electrical and Instrumentation and Controls

Components and structures subject to an aging management review were evaluated to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation. The components, aging effects/mechanism, and aging management programs to be used for managing the effects of aging at VCSNS were compared to those listed in NUREG-1801, Generic Aging Lessons Learned, dated April 2001. The results are documented and discussed in the following sections using the format contained in Tables 3.1-1 through 3.6-1 of NUREG-1800. Aging management programs are described in Appendix B of this application.

3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

3.1.1 SYSTEM DESCRIPTION

VCSNS Reactor Coolant System components evaluated in this section of the application consist of the Reactor Vessel, Reactor Vessel Internals, Incore Instrumentation System, Pressurizer, Steam Generators, and associated reactor coolant system piping.

The RCS consists of three (3) closed reactor coolant loops connected in parallel to the reactor pressure vessel. Each loop contains one steam generator, one reactor coolant pump, connecting piping and instrumentation. A pressurizer is connected, via a surge line, to the A loop reactor vessel outlet (hot leg) pipe. All components of the Reactor Coolant System are located within the Reactor Building.

Portions of the Incore Nuclear Instrumentation System (thimble tube pressure boundary), Chemical and Volume Control System (regenerative heat exchanger and Reactor Coolant Pump seal injection and return lines), Safety Injection System (piping and valves), Residual Heat Removal System (piping and valves), Reactor Vessel Level Instrumentation/Inadequate Core Cooling Monitor System, and the Primary Sampling System (piping and isolation valves) are included in this section, because they form part of the Class 1 Reactor Coolant System boundary.

3.1.2 AGING MANAGEMENT REVIEW

3.1.2.1 Methodology

Aging management review of Reactor Coolant System components and commodities consisted of one of two approaches (or a combination of the two). The first was application of the methods described in Section 4.2 of NEI 95-10 [**Reference 3.1-1**]. The VCSNS AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to VCSNS materials of construction and component internal and external operating environments and was used to identify aging effects for components, and commodities. The results of the evaluation of materials and environment combinations, using the VCSNS methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The second approach involved applying the information contained in NRC-approved industry Generic Topical Reports applicable to Reactor Coolant System and related components. The Generic Topical Reports provide both a list of components/ commodities requiring aging management and related aging management programs.

In either case, the aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in NUREG-1801 [**Reference 3.1-2**] and evaluated for consistency with NUREG-1801 programs that are relied on for license renewal. The results are documented and discussed in Table 3.1-1 using the format suggested by the NRC Standard Review Plan for License Renewal (NUREG-1800) [**Reference 3.1-3**]. Aging management programs are described in Appendix B of this application.

3.1.2.2 Operating Experience

Site: VCSNS site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

Industry: An evaluation of industry operating experience published since the effective date of NUREG-1801 was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

On-Going: On-going review of plant-specific and industry operating experience is performed in accordance with the plant Operating Experience Program.

3.1.3 AGING MANAGEMENT PROGRAMS

3.1.3.1 Aging Management Programs Evaluated In NUREG-1801 That Are Relied On For License Renewal

Table 3.1-1 shows the component and commodity groups (combinations of materials and environments), and aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Reactor Coolant System. The table is based on Table 3.1-1 of NUREG-1800 and provides a discussion of the applicability of the component commodity group and details regarding the degree to which VCSNS aging management programs are consistent with those recommended in NUREG-1801. The discussion section includes (1) information regarding the applicability of NUREG-1801 component/commodity group to VCSNS, (2) any issues recommended in NUREG-1801 that require further evaluation, (3) details regarding VCSNS components to be included in the component/commodity group, and (4) any additional materials to be added to the component/commodity groups beyond those identified in NUREG-1801.

3.1.3.2 Further Evaluation Of Aging Management As Recommended By NUREG-1801

Further evaluation of aging management as recommended by NUREG-1801 has been incorporated into the “Discussion” column of Table 3.1-1. A cross-reference is provided to the section of the application where TLAAs are discussed.

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
1	Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	The TLAA is applicable to Class 1 components at VCSNS. See Section 4.3 of this application for the TLAA discussion for fatigue of Class 1 components.
2	Steam generator shell assembly	Loss of material due to pitting and crevice corrosion	Inservice inspection; water chemistry	Yes, detection of aging effects is to be further evaluated	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. IN 90-04 which is referenced for this NUREG-1801 item as the basis for enhanced detection of aging effects in the shell, addressed flaw growth at girth welds due to service loading. IN 90-04 contains only general indication that pits on the surface served as crack initiation sites, and not that pitting corrosion resulted in sufficient degradation to cause loss of component function. No subsequent industry experience has further identified pitting corrosion resulting in reportable indications for the shell. Cracking as the

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)					result of flaw growth is an aging effect/mechanism that is managed at VCSNS by the ISI Plan (**Appendix B.1.7**). The Chemistry Program (**Appendix B.1.4**) manages general corrosion, pitting corrosion and crevice corrosion. Consistent with NUREG-1801, this group contains alloy steel portions of the Steam Generators. (Specifically the upper and lower shell and the transition cone) In addition to this, VCSNS has included the elliptical head.
3	Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA	The TLAA is applicable to loss of fracture toughness of the pressure vessel at VCSNS. See **Section 4.2** of this application for the TLAA discussion of loss of fracture toughness of the pressure vessel.

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3 (cont.)	(E>1 MeV)				
4	Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. The upper shell and nozzles are not considered the limiting components for radiation embrittlement due to their physical distance from the active fuel assemblies. The Reactor Vessel Surveillance program (**Appendix B.1.24**) will manage radiation embrittlement by addressing the most limiting sub-components with respect to exposure to the greatest fluence postulated to occur during the period of extended operation. Consistent with NUREG-1801, this group contains stainless steel clad alloy steel reactor vessel components.

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
5	Westinghouse and Babcock & Wilcox (B&W) baffle/ former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. The Reactor Vessel Internals Inspection program (**Appendix B.2.4**) will manage the Loss of Fracture Toughness due to Neutron irradiation Embrittlement and Void Swelling for baffle former bolts. Consistent with NUREG-1801, this group contains stainless steel baffle former bolts for a Westinghouse plant.
6	Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC),	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/ inspected and detection of aging effects are to be further evaluated	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. Small Bore Class 1 Piping Inspections (**Appendix B.2.7**), the Chemistry Program (**Appendix B.1.4**), and In-Service Inspection (ISI) Plan (**Appendix B.1.7**), are

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
6 (cont.)		and thermal and mechanical loading			credited for management of these aging effects, and meet the intent of the NUREG-1801 listed programs. Consistent with NUREG-1801, this group. contains cracking of small bore ASME class 1 stainless steel piping.
7	Vessel shell	Crack growth due to cyclic loading	TLAA	Yes, TLAA	This group addresses a TLAA for underclad cracking of vessels with ASME SA-508 class 2 material. The vessel at VCSNS is constructed of ASME SA-533 grade B class 1 (not ASME SA-508) material, therefore this aging effect is not applicable.
8	Reactor internals	Changes in dimension due to void swelling	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. Some additional reactor internals components have been included in

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
8 (cont.)					NUREG-1801. Materials listed in NUREG-1801 and used at VCSNS are both SS and Ni alloy. However, the clevis inserts are Ni alloy at VCSNS instead of stainless steel as specified in NUREG-1801. The credited program is Reactor Vessel Internals Inspection (**Appendix B.2.4**). Consistent with NUREG-1801, this group contains void swelling of the stainless steel and Ni-alloy reactor internals components. Additionally, irradiation creep is an aging mechanism affecting baffle and former bolts.
9	PWR core support pads, instrument tubes (bottom head	Crack initiation and growth due to SCC and/or primary water stress corrosion cracking (PWSCC)	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and aging effect. Chemistry Program (**Appendix B.1.4**), Alloy 600 Aging Management Program (**Appendix B.1.1**) and In-Service Inspection (ISI) Plan

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
9 (cont.)	penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains				(**Appendix B.1.7**) are credited for management of these aging effects. Consistent with NUREG-1801, this group contains cracking of various Ni alloy components. At VCSNS only the core support pads and bottom head penetration tubes are included in this grouping.
10	Cast austenitic stainless steel (CASS) reactor coolant system piping	Crack initiation and growth due to SCC	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and aging effect. NUREG-1801 recommends evaluation of a plant-specific aging management program. The Chemistry Program (**Appendix B.1.4**) and In-Service Inspection (ISI) Plan (**Appendix B.1.7**) are credited for management of these aging effects.

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
10 (cont.)					Consistent with NUREG-1801, this group contains cracking and crack growth of the of cast stainless steel RCS piping and fittings.
11	Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth due to PWSCC	Inservice inspection; water chemistry	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. At VCSNS this group has been used to envelope inconel 82/182 weld metal for the pressurizer safe end rather than alloy 600 base metal for instrument penetrations and heater sleeves. Alloy 600 Aging Management Program (**Appendix B.1.1**) is credited for management of this aging effect. Consistent with NUREG-1801, this group contains cracking and crack growth due to primary water stress corrosion cracking of Nickel-Based Alloy Pressurizer instrument penetrations and heater sleeves.

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
12	Westing-house and B&W baffle former bolts	Crack initiation and growth due to SCC and irradia-tion-assisted stress corrosion cracking (IASCC)	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and aging effect. The credited aging management programs are Reactor Vessel Internals Inspection (**Appen-dix B.2.4**) and Chemistry Program (**Appendix B.1.4**). Consistent with NUREG-1801, this group con-tains cracking of the reactor internals stainless steel baffle former bolts.
13	Westing-house and B&W baffle former bolts	Loss of preload due to stress relaxation	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and aging effects. The credited aging management program is Reactor Vessel Internals Inspection (**Appen-dix B.2.4**). Consistent with NUREG-1801, this group

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
13 (cont.)					contains loss of preload of the stainless steel reactor internals baffle former bolts.
14	Steam generator feedwater impingement plate and support	Loss of section thickness due to erosion.	Plant specific	Yes, plant specific	This NUREG-1801 group addresses erosion of steam generator feedwater impingement plate and supports. These components do not have a license renewal intended function for VCSNS, therefore aging management is not required.
15	(Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking (ODSCC), and/or intergranular attack (IGA); or loss of material due to wastage	Steam generator tubing integrity; water chemistry	Yes, effectiveness of a proposed AMP is to be evaluated	This group addresses cracking of various aging effect/mechanisms of steam generator tube bundles, sleeves and plugs. VCSNS has Westinghouse Delta 75 steam generators with thermally treated alloy 690 tubes. Phosphate water chemistry control has not been used at VCSNS. The credited aging management programs are the Chemistry Program (**Appendix B.1.4**) and Steam Generator Management Program (**Appendix B.1.10**).

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
15 (cont.)		and pitting corrosion, and fretting and wear; or deformation due to corrosion at tube support plate intersections			
16	Tube support lattice bars made of carbon steel	Loss of section thickness due to flow-accelerated corrosion (FAC)	Plant specific	Yes, plant specific	This NUREG-1801 group addresses erosion of steam generator tube support lattice bars. This group is applicable to Combustion Engineering steam generators and does not apply to VCSNS.
17	Carbon steel tube support plate	Ligament cracking due to corrosion	Plant specific	Yes, effectiveness of a proposed AMP is to be evaluated	This NUREG-1801 group addresses cracking of carbon steel steam generator tube supports. The tube supports at VCSNS are made of 405 stainless steel The alloy 690 anti-vibration bars are also included in this group. The Chemistry Program (Appendix B.1.4) alone is

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
17 (cont.)					sufficient to manage cracking.
18	Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and credited program. A clarification to the aging effect requiring management is required; i.e. Loss of closure integrity rather than cracking is the effect requiring management and the additional mechanism of stress relaxation being managed. The credited aging management program is Reactor Head Closure Studs Program (**Appendix B.1.8**). Consistent with NUREG-1801, this group contains alloy steel reactor vessel closure head bolts for PWR's.

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
19	CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and credited program. This group addresses loss of fracture toughness (thermal embrittlement) due to thermal aging as an aging effect for cast austenitic stainless steel (CASS) where temperatures continuously exceed 482ºF. The Reactor Coolant System is designed to operate at 650ºF. The applicable components fabricated from CASS are certain Reactor Coolant System Class 1 valve bodies and the reactor coolant pump casing. The credited aging management program is the In-Service Inspection (ISI) Plan (**Appendix B.1.7**). Consistent with NUREG-1801, this group

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
19 (cont.)					contains loss of fracture toughness of the cast stainless steel ASME class 1 pumps and valves.
20	CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and credited program. This group addresses loss of fracture toughness (thermal embrittlement) due to thermal aging as an aging effect for cast austenitic stainless steel (CASS) where temperatures continuously exceed 482ºF. The Reactor Coolant System is designed to operate at 650ºF. The applicable components fabricated from CASS are the reactor coolant loop piping elbows and the 45-degree accumulator nozzle. Additionally, the control rod drive latch housings have been included in this group. In a May 9, 2000 letter, Christopher I. Grimes, Chief License Renewal and Standardization

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
20 (cont.)					Chief License Renewal and Standardization Branch clarified that not all CASS components are subject to thermal aging. The components fabricated from CASS were evaluated using the acceptance criteria set forth in the above NRC letter. The screening criteria are supported by the SER for WCAP-14575-A which concludes that if the screening criteria are met the material is determined “not susceptible” and no additional inspections or evaluations are required because the material has adequate toughness. Based on the material chemistry for the CASS components at VCSNS, these components are not susceptible to a loss of fracture toughness due to thermal aging. The components have low molybdenum content and have delta ferrite levels of less than 20%. Therefore an

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
20 (cont.)					aging management program is not required for loss of fracture toughness (thermal embrittlement) due to thermal aging of these components.
21	BWR piping and fittings; steam generator components	Wall thinning due to flow accelerated corrosion	Flow accelerated corrosion	No	This NUREG-1801 group addresses loss of material due to flow accelerated corrosion of steam generator steam and feed nozzles and safe ends. VCSNS has not identified flow accelerated corrosion as a valid aging mechanism for these components.
22	Reactor coolant pressure boundary (RCPB) valve closure	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation	Bolting integrity	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material and environment. VCSNS is a Westinghouse plant and only PWR and recirculating steam generator components are applicable. A clarification to the aging effect/

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
22 (cont.)	bolting, manway and holding bolting, and closure bolting in high-pressure and high-temperature systems	and growth due to cyclic loading and/or SCC	Bolting integrity	No	mechanisms requiring management is required. Loss of closure integrity rather than loss of preload or cracking is the effect requiring management and the additional mechanism of stress relaxation being managed. Wear is not considered a valid aging effect/ mechanisms requiring management for the control rod drive flange bolting. In-Service Inspection (ISI) Plan (**Appendix B.1.7**) is credited for management of these aging effects. Consistent with NUREG-1801, this group contains alloy steel and stainless steel class 1 bolting (other than reactor vessel closure head bolts) and steam generator class 2 bolting.
23	CRD nozzle	Crack initiation and growth due to PWSCC	Ni-alloy nozzles and penetrations; water chemistry	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
23 (cont.)					credited programs. The credited aging management program is the Alloy 600 Aging Management Program (**Appendix B.1.1**). Consistent with NUREG-1801, this group contains PWSCC of Ni-alloy control rod drive head penetration nozzles.
24	Reactor vessel nozzles safe ends and CRD housing; reactor coolant system components (except	Crack initiation and growth due to cyclic loading, and/or SCC, and PWSCC	Inservice inspection; water chemistry	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. The pressurizer relief tank is not in-scope for VCSNS. Consistent with NUREG-1801, the materials include stainless steel, stainless steel clad carbon steel, Ni alloy and cast stainless steel. The NUREG-1801 Table 1 “Component” column provides an exception for cast stainless steel, however, its “Item Number In GALL”

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
24 (cont.)	CASS and bolting)				column includes cast stainless steel pumps and valves. VCSNS has included these components in this group. Consistent with NUREG-1801, the credited aging management programs are the Chemistry Program (**Appendix B.1.4**) and In-Service Inspection (ISI) Plan (**Appendix B.1.7**). Whereas both the Chemistry Program and ISI are credited, ISI for this group is primarily directed at welded connections (and thereby crack growth). As such, the Chemistry Program itself provides adequate management for the non-welded portions of components/ component types within this group (such as the RCP main closure flange and casing and the pressurizer tubing, couplings, and immersion heater well assemblies). Additionally the Alloy 600 Aging Management Program

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
24 (cont.)					(Appendix B.1.1) is credited for management of primary water stress corrosion of Ni-alloy. Consistent with NUREG-1801, this group contains cracking and crack growth of many of the Class 1 components in a PWR. In addition to the listed components, other reactor vessel components have been included in the group.
25	Reactor vessel internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	Thermal aging and neutron irradiation embrittlement	No	No cast austenitic stainless steel reactor vessel internal components are in scope for VCSNS.
26	External surfaces of carbon steel	Loss of material due to boric acid corrosion	Boric acid corrosion	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
26 (cont.)	components in reactor coolant system pressure boundary				credited program. The pressurizer relief tank is not in-scope for VCSNS. The credited aging management program is Boric Acid Corrosion Surveillances (**Appendix B.1.2**). Consistent with NUREG-1801, this group contains boric acid corrosion of carbon (and alloy) steel reactor coolant system components.
27	Steam generator secondary manways and handholds (carbon steel)	Loss of material due to erosion	Inservice inspection	No	This NUREG-1801 group addresses erosion of steam generator manways and hand holes of once-through steam generators. VCSNS has Westinghouse recirculating steam generators.
28	Reactor internals, reactor vessel closure studs,	Loss of material due to wear	Inservice inspection	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. The materials listed in

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
28 (cont.)	and core support pads				NUREG-1801 and used at VCSNS are the same; however, the material for the clevis insert is alloy 600, instead of stainless steel. The components referenced in NUREG-1801 are the upper core plate alignment pins, vessel flange, radial keys, clevis inserts and flux thimbles. VCSNS has also included the head vessel alignment pins in this group. The Reactor Vessel Internals Inspection (**Appendix B.2.4**) is credited for reactor vessel Internals. The In-Service Inspection (ISI) Plan (**Appendix B.1.7**) is credited for the vessel flange. The Bottom Mounted Instrumentation Inspection (**Appendix B.1.3**) is credited for aging management of the bottom mounted flux thimbles. Consistent with NUREG-1801, this group contains wear of components (other than bolting)

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
28 (cont.)					of the reactor vessel and internals components that are routinely dissembled.
29	Pressurizer integral support	Crack initiation and growth due to cyclic loading	Inservice inspection	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. The credited aging management program is In-Service Inspection (ISI) Plan (**Appendix B.1.7**). This NUREG-1801 group addresses crack initiation and growth of the pressurizer integral support.
30	Upper and lower internals assembly (Westinghouse)	Loss of preload due to stress relaxation	Inservice inspection; loose part and/or neutron noise monitoring	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, and aging effect. The Reactor Vessel Internals Inspection (**Appendix B.2.4**) is credited for reactor vessel Internals. VCSNS does not credit Loose Parts

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
30 (cont.)					Monitoring or Neutron Noise Monitoring. Consistent with NUREG-1801, this group contains loss of preload of the stainless steel holddown spring and the Ni-alloy clevis insert bolts.
31	Reactor vessel internals in fuel zone region (except Westinghouse and B&W baffle former bolts)	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	PWR vessel internals; water chemistry	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, and aging effect. VCSNS is a Westinghouse plant and only the Westinghouse components are applicable. The Reactor Vessel Internals Inspection program (**Appendix B.2.4**) will manage the loss of fracture toughness for applicable components. The Chemistry Program is not credited for managing loss of fracture toughness. Consistent with NUREG-1801, this group contains loss of fracture toughness due to

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
31 (cont.)					irradiation embrittlement in components in close proximity to the core.
32	Steam generator upper and lower heads, tubesheets, and primary nozzles and safe ends	Crack initiation and growth due to SCC, PWSCC, and/or IASCC	Inservice inspection; water chemistry	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. NUREG-1801 group only identifies the steam generator primary side nozzles and safe ends. VCSNS has included all of the primary side of the steam generator except tubes, plug and sleeves. VCSNS is a Westinghouse plant and has recirculating steam generator. The credited aging management programs are Chemistry Program (**Appendix B.1.4**), In-Service Inspection Plan (**Appendix B.1.7**) and Alloy 600 Aging Management Program (**Appendix B.1.1**). Consistent with NUREG-1801, this group contains cracking of steam generator primary

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
32 (cont.)					nozzles and safe ends.
33	Vessel internals (except Westinghouse and B&W baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR vessel internals; water chemistry	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. VCSNS is a Westinghouse plant and only Westinghouse components are applicable. there is difference in material. The clevis inserts are alloy 600 at VCSNS instead of stainless steel as shown in NUREG-1801. Head/vessel alignment pins, secondary core support and spray nozzles for upper plenum cooling, are being included in this item. Although they are not addressed in this NUREG-1801 group, with respect to material, environment, aging effect requiring management, and credited program the sub-component AMR results are consistent with NUREG-1801. The Reactor Vessel Internals Inspection

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
33 (cont.)					program (**Appendix B.2.4**) and the Chemistry Program (**Appendix B.1.4**) will manage crack initiation and growth in these components. Consistent with NUREG-1801, this group contains crack initiation and growth of the stainless steel and Ni-alloy reactor internals except baffle former bolts.
34	Reactor vessel closure studs and stud assembly and stud assembly	Loss of material due to wear	Reactor head closure studs	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. The credited program is Reactor Head Closure Studs Program (**Appendix B.1.8**). Consistent with NUREG-1801, this group contains wear of the alloy steel reactor vessel closure stud assembly.

Table 3.1-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE REACTOR COOLANT SYSTEM EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
35	Reactor internals (Westinghouse upper and lower internal assemblies, CE bolts and tie rods)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. Material used at VCSNS is stainless steel. NUREG-1801 lists both stainless steel and Ni alloy. Whereas NUREG-1801 specifies the ASME Section XI ISI and Loose Parts Monitoring, VCSNS credits only the Reactor Vessel Internals Inspection (**Appendix B.2.4**). Consistent with NUREG-1801, this group contains loss of preload of the stainless steel reactor internals upper and lower support column bolts.

3.1.3.3 Aging Management Evaluations That Are Different From Or Not Addressed In NUREG-1801

Aging Management Evaluations that are different from or not addressed in NUREG-1801 are identified and discussed in Table 3.1-2. The standard six-column format has been utilized. The discussion column provides additional details regarding the aging management conclusions reached by VCSNS for the component type.

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
1	Piping and piping system components	Stainless Steel (including Cast Austenitic Stainless Steel)	Air-gas (moist air)	None Identified	None Required	This grouping includes external surfaces of stainless steel piping system components exposed to a moist air environment. At VCSNS, the ambient environment does not contain contaminants of sufficient concentration to cause aging effects that require aging management.
2	Piping and piping system components	Nickel-Based Alloy, Stainless Steel with Nickel-Based Weld	Air-gas (moist air)	None Identified	None Required	This grouping includes external surfaces of Nickel-Based Alloy piping system components exposed to a moist air environment. At VCSNS, the ambient environment does not contain contaminants of sufficient concentration to cause

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
2 (cont.)						aging effects that require aging management.
3	Steam Generator Components (Other than Shell - Upper and Lower Barrel, Transition Cone, Elliptical Head)	Carbon Steel	Treated Water	Loss of Material/ Crevice Corrosion, General Corrosion, Pitting Corrosion, Galvanic Corrosion	Chemistry Program	This grouping includes carbon steel components exposed to the treated water/ steam environment inside the steam generator. VCSNS has determined the Chemistry Program (**Appendix B.1.4**) provides adequate management for these secondary side components of the steam generator, which are not specifically addressed in NUREG-1801 chapter IV.
4	Instrumentation (Pressure	Stainless Steel	Reactor Building	Loss of Mechanical Closure Integrity/ Stress	In-Service Inspection (ISI) Plan	Loss of mechanical closure integrity due to stress relaxation, stress corrosion

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
4 (cont.)	Retaining Only) - Incore Thermocouples Seal			Relaxation, Wear		cracking, and or wear of bolted closures is an aging effect requiring evaluation for the in-core thermocouple seal assemblies. Stress relaxation (loss of preload) of bolting materials is a concern at the elevated temperatures at which the Reactor Coolant System operates. Stress corrosion cracking is a concern for high strength bolting (> 150 ksi) materials in Class 1 closure applications. Wear of bolted closures is a concern due to relative motion during infrequent (periodic) disassembly and reassembly operations. The

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
4 (cont.)						In-Service Inspection (ISI) Plan (**Appendix B.1.7**) is credited for management of these aging effects.
5	Piping and piping system components	Stainless Steel	Chemically Treated Borated Water	Loss of Material due to Crevice Corrosion, Pitting Corrosion	Chemistry Program	This grouping includes stainless steel piping and piping system components in chemically treated borated water. The Chemistry Program (**Appendix B.1.4**) alone provides aging management for loss of material of stainless steel in chemically treated borated water environment.
6	Piping and piping system components	Stainless Steel	Chemically Treated Borated Water	Crack Initiation and Growth/ Stress Corrosion Cracking	Chemistry Program	This grouping includes stainless steel piping and piping system components in chemically treated

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
6 (cont.)						borated water. The Chemistry Program (**Appendix B.1.4**) alone provides aging management for cracking of stainless steel in a chemically treated borated water environment.
7	Reactor Internals components, Reactor vessel, Pressurizer, RCP, Incore thermocouple seal	Stainless Steel and Ni-alloy	Chemically Treated Borated Water	Loss of Material due to Crevice Corrosion, Pitting Corrosion	Chemistry Program	This grouping includes stainless steel and Ni-alloy Reactor Coolant System components in chemically treated borated water. The Chemistry Program (**Appendix B.1.4**) alone provides aging management for loss of stainless steel material in chemically treated borated water environment.

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
8	Steam Generator Secondary Side Thermal Sleeves, Steam Flow Limiter	Nickel-Based Alloy (Alloy 690 TT)	Treated Water	Loss of Material due to Crevice Corrosion, Pitting Corrosion	Chemistry Program	This grouping includes Ni-alloy steam generator secondary side components. The Chemistry Program (**Appendix B.1.4**) provides adequate management for loss of material of these secondary side components of the steam generator in a treated water environment, which are not specifically addressed in GALL chapter IV.
9	Steam Generator Secondary Side Thermal Sleeves, Steam Flow Limiter	Nickel-Based Alloy (Alloy 690 TT)	Treated Water	Crack Initiation and Growth/ Stress Corrosion Cracking, Flaw Growth at Welds	Chemistry Program, The In-Service Inspection (ISI) Plan	This grouping includes Ni-alloy steam generator secondary side components. The Chemistry Program (**Appendix B.1.4**) and the In-Service Inspection (ISI) Plan (**Appendix B.1.7**)

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
9 (cont.)						provide adequate management for cracking of these secondary side components of these steam generator in a treated water environment, which are not specifically addressed in GALL chapter IV.
10	Steam Generator Feedwater Distribution Pipe & Fittings	Alloy Steel (Chrome Moly)	Treated Water	Loss of Material due to Crevice Corrosion, Pitting Corrosion; Crack initiation and growth/ Stress corrosion cracking	Chemistry Program, The In-Service Inspection (ISI) Plan	This grouping includes alloy feedwater distribution components. The Chemistry Program (**Appendix B.1.4**) and the In-Service Inspection (ISI) Plan (**Appendix B.1.7**) provide adequate management for loss of material and cracking of these secondary side components of the steam generator.

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
11	Reactor Internals components, Reactor vessel, Pressurizer, RCP, Incore thermocouple seal	Stainless Steel and Nickel-Based Alloy	Chemically Treated Borated Water	Crack Initiation and Growth/ Stress Corrosion Cracking	Chemistry Program	This grouping includes stainless steel and Ni-alloy Reactor Coolant System components in chemically treated borated water. The Chemistry Program (**Appendix B.1.4**) alone provides aging management for cracking of stainless steel and Ni-alloy in a chemically treated borated water environment.
12	Capillary tubes and associated components	Stainless Steel	Treated Water	None Identified	None Required	This group includes stainless steel capillary tubes and associated components exposed to treated water and is consistent in material and environment with NUREG-1801 closed-cycle cooling water

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
12 (cont.)						components. However, at VCSNS, no aging effects were determined to require management due to no means of oxygen or other contaminant entry existing for the deaerated distilled water inside these capillary tubes and associated components.
13	Tube Support Plates - Plates, AVBs, Flow Distribution Baffle - Secondary Side for Steam Generators	Stainless Steel / Nickel Based Tips & Rings	Treated Water	Loss of Material due to Crevice Corrosion, Pitting Corrosion; Crack Initiation and Growth/ Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking)	Chemistry Program	This grouping includes stainless steel Ni-alloy steam generator components in treated water. The Chemistry Program (**Appendix B.1.4**) alone provides aging management for loss of material and cracking of stainless steel and Ni-alloy in a

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
13 (cont.)						treated water/ steam environment.
14	Piping and piping system components	Stainless Steel	Treated Water	Loss of Material due to Crevice Corrosion, Pitting Corrosion	Chemistry Program	This grouping includes stainless steel piping and associated components exposed to treated water from the Reactor Makeup Water system for Pressurizer Relief Tank spray. It is consistent in material and environment, aging effect and credited program with NUREG-1801 closed-cycle cooling water components. The Chemistry Program (**Appendix B.1.4**) provides adequate management for this component, material and environment that is not specifically addressed in

Table 3.1-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE REACTOR COOLANT SYSTEM THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
14 (cont.)						NUREG-1801 chapter IV.
15	Piping and piping system components	Stainless Steel	Air-gas	None Identified	None Required	This grouping includes the internal surfaces of stainless steel piping system components exposed to a gas environment that is not subject to wetting. The gas environment does not contain contaminants of sufficient concentration to cause aging effects that require aging management.

3.1.4 REFERENCES

3.1-1	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.
3.1-2	NUREG-1801, “Generic Aging Lessons Learned Report,” Volumes 1 and 2, NRC, April 2001.
3.1-3	NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” NRC, April 2001.

3.2 AGING MANAGEMENT OF ENGINEERED SAFETY FEATURES

3.2.1 SYSTEM DESCRIPTION

Engineered Safety Features (ESF) are provided to mitigate the consequences of postulated accidents up to and including a design basis accident as discussed in **Chapter 15 of the FSAR**. The ESF systems provided for VCSNS have sufficient redundancy and independence of components and power sources that, under postulated accident conditions, the following are accomplished:

1. Core cooling limits the core thermal transient, thereby preventing excessive metal-water reactions and maintaining coolable core geometry.

2. Reactor Building structural integrity is maintained.

3. Radiation dose to the public is maintained within the limits of 10 CFR 100.

The ESF systems include the following systems:

- Chemical and Volume Control System
- Containment Isolation System
- Hydrogen Removal System
- Reactor Building Spray System
- Refueling Water System
- Residual Heat Removal System
- Safety Injection System

3.2.2 .AGING MANAGEMENT REVIEW

3.2.2.1 Methodology

Aging management review of Engineered Safety Features System components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [**Reference 3.2-1**]. The VCSNS AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to VCSNS materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the VCSNS methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in NUREG-1801 [**Reference 3.2-2**] and evaluated for consistency with NUREG-1801 programs that are relied on for license renewal. The results are documented and discussed in Table 3.2-1 using the format suggested by the NRC Standard Review Plan for License Renewal (NUREG-1800) [**Reference 3.2-3**].

3.2.2.2 Operating Experience

Site: VCSNS site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

Industry: An evaluation of industry operating experience published since the effective date of NUREG-1801 was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

On-Going: On-going review of plant-specific and industry operating experience is performed in accordance with the plant Operating Experience Program.

3.2.3 AGING MANAGEMENT PROGRAMS

3.2.3.1 Aging Management Programs Evaluated In NUREG-1801 That Are Relied On For License Renewal

Table 3.2-1 shows the component and commodity groups (combinations of materials and environments) and aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Engineered Safety Features Systems. The table is based on Table 3.2-1 of NUREG-1800 [**Reference 3.2-3**] and provides a discussion of the applicability of the component commodity group and details regarding the degree to which VCSNS aging management programs are consistent with those recommended in NUREG-1801. The discussion section includes (1) information regarding the applicability of NUREG-1801 component/commodity group to VCSNS, (2) any issues recommended in NUREG-1801 that require further evaluation, (3) details regarding VCSNS components to be included in the component/commodity group, and (4) any additional materials to be added to the component/commodity groups beyond those identified in NUREG-1801.

3.2.3.2 Further Evaluation Of Aging Management As Recommended By NUREG-1801

Further evaluation of aging management as recommended by NUREG-1801 has been incorporated into the “Discussion” column of Table 3.2-1. A cross-reference is provided to the section of the application where TLAAs are discussed.

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
1	Piping, fittings, and valves in emergency core cooling system	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	The TLAA is applicable to Class 2 and 3 piping systems at VCSNS. See **Section 4.3.2** for the TLAA discussion of Class 2 and 3 piping systems. VCSNS is a Westinghouse PWR plant.
2	Components in containment spray (PWR only), standby gas treatment system (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to general corrosion	Plant specific	Yes, plant specific	The VCSNS Containment Spray System is made of stainless steel, rather than carbon steel. Four systems at VCSNS have been identified whose only license renewal function is to provide containment isolation. These systems are: Auxiliary Coolant (Closed Loop) / CRDM Cooling Water (AC), Demineralized Water - Nuclear Service (DN), RB Leak Rate Testing (LR) and Nitrogen Blanketing (NG). AC, LR, and NG contain carbon steel components in scope. Inspections for Mechanical

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)					Components (**Appendix B.2.11**) are credited for detecting and managing the loss of material due to general corrosion on the external surfaces of carbon steel components. The air-gas environments inside NG and LR components do not contain contaminants of sufficient concentration to cause general corrosion severe enough to warrant managing. The Chemistry Program (**Appendix B.1.4**) is credited with providing aging management of corrosion of the interior of AC System components. Consistent with NUREG-1801, this group contains carbon steel in ambient air, and treated water environments. Additionally compressed gas is included in this group.
3	Components in	Loss of material due to pitting and	Plant specific	Yes, plant specific	This group contains the Refueling Water Storage Tank (RWST) and systems whose only

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3 (cont.)	containment spray (PWR only), standby gas treatment system (BWR only), containment isolation, and emergency core cooling systems	crevice corrosion	Plant specific	Yes, plant specific	license renewal function is to provide containment isolation. VCSNS has determined that loss of material of the underside of the RWST is not an aging effect requiring management as this stainless steel tank is not buried. Four systems at VCSNS have been identified whose only license renewal function is to provide containment isolation. These systems are: AC, DN, LR and NG. Pitting and crevice corrosion does not require aging management for the exterior environments of stainless steel components. Inspections for Mechanical Components (**Appendix B.2.11**) is credited for detecting and managing the loss of material due to pitting, crevice, and galvanic corrosion on the external surfaces of carbon steel

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3 (cont.)					components. The air-gas environments inside NG and LR components do not contain contaminants of sufficient concentration to cause pitting and crevice corrosion severe enough to warrant managing. The Chemistry Program (**Appendix B.1.4**) is credited with providing aging management of corrosion of the interior of AC and DN System components. Consistent with NUREG-1801, this group contains both carbon and stainless steel in ambient air, and treated water environments. Additionally compressed gas is included in this group.
4	Containment isolation valves and associated piping	Loss of material due to microbiologically influenced corrosion (MIC)	Plant specific	Yes, plant specific	Microbiologically influenced corrosion has been determined not to be a valid aging effects/mechanisms for the component/component type material and environment combination represented by this group. The four

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
4 (cont.)					systems at VCSNS that have been identified to provide containment isolation are not subject to wetting from raw water. These systems are: AC, DN, LR and NG.
5	High-pressure safety injection (charging) pump mini-flow orifice	Loss of material due to erosion	Plant specific	Yes, plant specific	NUREG-1801 V.D1.2-c addresses Emergency Core Cooling System orifices. The component/component type AMR results are consistent with the identified GALL item in material and environment. However, the identified GALL item recommends plant specific evaluation of the credited program. VCSNS considers this aging effect/mechanism a design problem, and does not have an identified aging management program for erosion of mini-flow orifices.
6	External surface of carbon steel	Loss of material due to general corrosion	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect and

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
6 (cont.)	components				credited program. Consistent with NUREG-1801, this group contains the external surfaces of carbon steel components. In addition to carbon steel, VCSNS has included cast iron in this group. Consistent with NUREG-1801, the aging mechanism for this group is general corrosion. NUREG-1801 recommends a plant specific evaluation of the credited program. The credited program/activity at VCSNS is Inspections for Mechanical Components (**Appendix B.2.11**).
7	Piping and fittings of CASS in emergency	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	The component/component type AMR results for VCSNS are not consistent with NUREG-1801 in material, environment, aging effect and program. VCSNS does not have cast

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)	core cooling systems				stainless steel in ESF Systems subject to temperatures over 482ºF.
8	Components serviced by open-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	The component/component type AMR results for VCSNS are not consistent with NUREG-1801 in material, environment, aging effect and program. VCSNS component/component type AMR results for components serviced by open-cycle cooling systems are contained in Auxiliary Systems, **Section 3.3**.
9	Components serviced by closed-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and aging effect. NUREG-1801 recommends a closed cycle cooling program. The credited program/activity at VCSNS is the Chemistry Program (**Appendix B.1.4**). The Chemistry Program has been in effect since initial plant startup

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
9 (cont.)					and has proven effective in maintaining systems' chemistry and detecting abnormal conditions. A review of operating experience confirms the effectiveness of the Chemistry Program to manage aging effects when continued into the period of extended operation. A verification program is, therefore, not warranted for the components/ component types in this group. The aging mechanisms for this group include loss of material due to general corrosion, crevice corrosion and pitting corrosion. In addition to these aging mechanisms, stress corrosion cracking is managed for this group at VCSNS. Consistent with NUREG-1801, this group includes carbon steel and stainless steel components in borated or treated water environments.

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
9 (cont.)					Consistent with NUREG-1801, this group includes heat exchanger components. VCSNS does not utilize heat exchangers in the Containment Spray System (Reactor Building Spray System at VCSNS). Some components may also be shown or listed in Auxiliary Systems, **Section 3.3**.
10	Pumps, valves, piping and fittings, and tanks in containment spray and emergency core cooling system	Crack initiation and growth due to SCC	Water chemistry	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment and aging effects. The credited program/activity at VCSNS is the Chemistry Program (**Appendix B.1.4**). Consistent with NUREG-1801, this group includes stainless steel and stainless steel cladding in borated or treated water environments.

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
10 (cont.)					Consistent with NUREG-1801, this group includes stress corrosion cracking of stainless steel and stainless steel clad pumps, valves, piping, fittings and tanks. In addition to this aging mechanism, crevice and pitting corrosion are also managed for this group at VCSNS.
11	Carbon steel components	Loss of material due to boric acid corrosion corrosion	Boric acid corrosion	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect and credited program. The credited program/activity at VCSNS is Boric Acid Corrosion Surveillances (**Appendix B.1.2**). Consistent with NUREG-1801, this group contains external surfaces of carbon and low alloy steel components/component types in an ambient air environment, where there is a potential for leaking and dripping of chemically

Table 3.2-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE ENGINEERED SAFETY FEATURES EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
11 (cont.)					treated borated water onto component surfaces.
12	Closure bolting in high-pressure or high-temperature systems	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	Loss of mechanical closure integrity is not considered an aging effect requiring evaluation for Non-Class 1 component bolted closures within the scope of license renewal at VCSNS. As such, the specific bolting/fastener materials of subject components/component types within the scope of license renewal were not itemized as a separate Non-Class 1 component/component type. Rather, bolting was treated as a “piece-part” (or sub-component/sub-part) of Non-Class 1 components/component types.

3.2.3.3 Aging Management Evaluations That Are Different From Or Not Addressed In NUREG-1801

Aging Management Evaluations that are different from or not addressed in NUREG-1801 are identified and discussed in Table 3.2-2. The standard six-column format has been utilized. The discussion column provides additional details regarding the aging management conclusions reached by VCSNS for the component type.

Table 3.2-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE ENGINEERED SAFETY FEATURES THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
1	Piping and piping system components (This includes tanks pump casings filter housings and heat exchanger components.)	Stainless Steel (including Cast)	Air-gas	None Identified	None Required	This grouping includes internal and external surfaces of stainless steel piping system components exposed to non-corrosive gas environment. At VCSNS, the ambient environment of the yard and sheltered environments do not contain contaminants of sufficient concentration to cause aging effects that require aging management.
2	Piping and piping system components	Carbon Steel	Air-gas (dry)	None Identified	None Required	This grouping includes external surfaces of carbon steel piping system components exposed to air that are not subject to wetting. It also includes the internal surface of these

Table 3.2-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE ENGINEERED SAFETY FEATURES THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
2 (cont.)						components when exposed to dry air or non-corrosive process gasses.
3	Recombiner H_2	Nickel-based Alloy	Air-gas	None Identified	None Required	This grouping contains the Post-accident H_2 recombiner. Nickel-based alloys do not require aging management in the air environment of the Reactor Building.
4	Piping and piping system components	Carbon Steel, copper-nickel	Oil	None Identified	None Required	This grouping includes oil-coated surfaces of system components exposed to lubricating oil that are not subject to wetting. These components are various components in the RCP lube oil systems that are continuously in service.

Table 3.2-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE ENGINEERED SAFETY FEATURES THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
5	Reactor Building Spray components associated with NaOH addition Piping and piping system components	Stainless Steel, Carbon Steel	Treated Water (Less Than 140°F with Caustic)	Loss of Material/ General Corrosion (carbon steel only), Crevice Corrosion, Pitting Corrosion; Cracking/ Stress Corrosion Cracking	Chemistry Program and Above Ground Tank Inspection	Consistent with NUREG-1801 for non-ESF systems, the Chemistry Program (**Appendix B.1.4**) is credited for management of the specified aging effects in the associated caustic treated water environment. However, due to the lack of oxygen control in the supply to the Sodium Hydroxide Storage Tank a one-time inspection is required as part of the Above Ground Tank Inspection (**Appendix B.2.1**) at VCSNS to ensure that carbon steel portions are not experiencing sufficient degradation to cause loss of intended

Table 3.2-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE ENGINEERED SAFETY FEATURES THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
5 (cont.)						function during the period of extended operation.
6	Capillary tubes and associated components	Stainless Steel	Treated Water	None Identified	None Required	This group includes stainless steel capillary tubes and associated components exposed to treated water and is consistent in material and environment with NUREG-1801 closed-cycle cooling water components. However, at VCSNS, no aging effects were determined to require management due to no means of oxygen or other contaminant entry existing for the deaerated distilled water inside these capillary tubes and associated components.

Table 3.2-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE ENGINEERED SAFETY FEATURES THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
7	Refueling Water Storage Tank (including attached piping)	Stainless Steel	Treated Water	Loss of Material due to Crevice Corrosion, Pitting Corrosion; Cracking Due To Corrosive Impacts Of Alternate Wetting And Drying	Chemistry Program, Above Ground Tank Inspection	The components in the grouping are partially consistent with a NUREG-1801 Engineered Safety Feature item with respect to material, environment, aging effect requiring management and aging management program (Chemistry Program). However, additional aging effects require management at VCSNS for this ESF grouping. NUREG-1801 does not address the corrosive impacts of alternate wetting and drying of vented tanks (or tanks without a cover gas) that could concentrate contaminants

Table 3.2-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE ENGINEERED SAFETY FEATURES THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
7 (cont.)						above bulk fluid concentrations and result in further degradation. The Chemistry Program (**Appendix B.1.4**) will manage the conditions required for a loss of material or cracking of the components in this grouping to occur in bulk fluid concentrations. In addition, the one-time Above Ground Tank Inspection (**Appendix B.2.1**) will manage the corrosive impacts of alternate wetting and drying to occur in stainless steel in the treated water environment. These programs,

Table 3.2-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE ENGINEERED SAFETY FEATURES THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
7 (cont.)						when continued into the period of extended operation, will provide reasonable assurance that the component intended function(s) will be maintained under all CLB conditions.

3.2.4 REFERENCES

3.2-1	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.
3.2-2	NUREG-1801, “Generic Aging Lessons Learned Report,” Volumes 1 and 2, NRC, April 2001.
3.2-3	NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” NRC, April 2001.

3.3 AGING MANAGEMENT OF AUXILIARY SYSTEMS

3.3.1 SYSTEM DESCRIPTION

The Auxiliary Systems are those systems used to support normal and emergency plant operations. The systems provide cooling, ventilation, sampling and other required functions.

The Auxiliary Systems include the following systems:

- Air Handling and Local Ventilation and Cooling Systems
- Boron Recycle System
- Chilled Water System
- Component Cooling Water System
- Diesel Generator Services Systems
- Fire Service System
- Fuel Handling System
- Gaseous Waste Processing System
- Instrument Air Supply System
- Liquid Waste Processing System
- Nuclear Plant Drains
- Non-Nuclear Plant Drains
- Nuclear Sampling System
- Radiation Monitoring System
- Reactor Makeup Water Supply System
- Roof Drains System
- Station Service Air System
- Service Water System
- Spent Fuel Cooling System
- Thermal Regeneration System

3.3.2 AGING MANAGEMENT REVIEW

3.3.2.1 Methodology

Aging management review (AMR) of Auxiliary Systems components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [**Reference 3.3-1**]. The VCSNS AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to VCSNS materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the VCSNS methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in NUREG-1801 [**Reference 3.3-2**] and evaluated for consistency with NUREG-1801 programs that are relied on for license renewal. The results are documented and discussed in Table 3.3-1 using the format suggested by the NRC Standard Review Plan for License Renewal (NUREG-1800) [**Reference 3.3-3**].

3.3.2.2 Operating Experience

Site: VCSNS site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

Industry: An evaluation of industry operating experience published since the effective date of NUREG-1801 was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

On-Going: On-going review of plant-specific and industry operating experience is performed in accordance with the plant Operating Experience Program.

3.3.3 AGING MANAGEMENT PROGRAM

3.3.3.1 Aging Management Programs Evaluated In NUREG-1801 That Are Relied On For License Renewal

Table 3.3-1 shows the component and commodity groups (combinations of materials and environments), and aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Auxiliary Systems. The table is based on Table 3.3-1 of NUREG-1800 [**Reference 3.3-3**] and provides a discussion of the applicability of the component commodity group and details regarding the degree to which VCSNS aging management programs are consistent with those recommended in NUREG-1801. The discussion section includes (1) information regarding the applicability of NUREG-1801 component/commodity group to VCSNS, (2) any issues recommended in NUREG-1801 that require further evaluation, (3) details regarding VCSNS components to be included in the component/commodity group, and (4) any additional materials to be added to the component/commodity groups beyond those identified in NUREG-1801.

3.3.3.2 Further Evaluation Of Aging Management As Recommended By NUREG-1801

Further evaluation of aging management as recommended by NUREG-1801 has been incorporated into the “Discussion” column of Table 3.3-1. A cross-reference is provided to the section of the application where TLAAs are discussed.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
1	Components in spent fuel pool cooling and cleanup	Loss of material due to general, pitting, and crevice corrosion	Water chemistry and one time inspection	Yes, detection of aging effects is to be further evaluated	The combinations of components, materials, and environments identified in NUREG-1801 are not applicable to VCSNS. There are no carbon steel components with elastomer linings in the Spent Fuel Cooling System that are used to perform a license renewal intended function.
2	Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific	The component /component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, and partially consistent in aging effects. The VCSNS plant specific program for managing aging is Inspections for Mechanical Components (**Appendix B.2.11**). This program inspects external surfaces. It is expected that the aging effects for these components/component types will be exhibited on external surfaces before being exhibited on internal surfaces, therefore an aging management program that

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)					detects and manages only aging effects of the external surfaces is sufficient. Also, loss of material due to wear is not considered an aging effect requiring management at VCSNS because mechanical components must perform their License Renewal intended functions without moving parts. Wear that occurs on non-moving components is considered to be caused by improper design and should be corrected by normal maintenance activities. Consistent with NUREG-1801, this group includes elastomers in an air environment. Consistent with NUREG-1801, this group includes flexible collars between ducts and fans. In addition to the Air Handling System, this group includes components/component types

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)					from the Diesel Generator System, where these components/component types are similar and the above discussions apply.
3	Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems (older BWR)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	The TLAA on fatigue is applicable to Class 2 and 3 piping systems, which includes the Chemical and Volume Control System at VCSNS. Refer to **Section 4.3.2** of this application for the discussion of fatigue of Class 2 and 3 piping systems. The VCSNS fuel handling cranes are adequately analyzed and designed for fatigue through the term of extended operation. Refer to **Section 4.7.3.3** of this application.
4	Heat exchangers	Crack initiation and growth due to	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
4 (cont.)	in reactor water cleanup system (BWR); high pressure pumps in chemical and volume control system (PWR)	SCC or cracking	Plant specific	Yes, plant specific	in material, environment, and aging effects. The VCSNS plant specific program for managing aging is the Chemistry Program (**Appendix B.1.4**). Consistent with NUREG-1801, the aging mechanism for this group is stress corrosion cracking. In addition, crevice corrosion and pitting corrosion are also managed for this group at VCSNS. Non-Class 1 Closure bolting is considered to be a piece-part of the components/component types as a whole at VCSNS. Therefore a bolting integrity program is not credited for aging management. As a piece-part of subject components/component types at VCSNS, the specific bolting/fastener materials were not itemized as a separate Non-Class 1 component or component type. Additionally, for carbon and alloy steel components, the aging

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
4 (cont.)					management program credited for managing external general corrosion of the applicable components/component types (e.g. Inspections for Mechanical Components [Appendix B.2.11]) will also inherently address their fasteners, thus requiring no separate action. Consistent with NUREG-1801, this group includes stainless steel in a chemically treated borated water environment. Consistent with NUREG-1801, this group includes pump casings.
5	Components in ventilation systems, diesel fuel oil system, and emergency diesel	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, and partially consistent in aging effects. The VCSNS plant specific programs/activities for managing aging are: Inspections for Mechanical Components (Appendix B.2.11), Maintenance Rule

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
5 (cont.)	generator systems; external surfaces of carbon steel components				Structures Program (**Appendix B.1.18**), Boric Acid Corrosion Surveillances (**Appendix B.1.2**), Preventive Maintenance Activities - Ventilation Systems Inspections (**Appendix B.1.26**), and Diesel Generator Inspections (**Appendix B.2.2**). Consistent with NUREG-1801, the aging mechanisms for this group include general corrosion, pitting corrosion, and external MIC. In addition to these aging mechanisms, boric acid corrosion, galvanic corrosion, heat exchanger fouling due to particulates, and alternate wetting and drying are managed for this group at VCSNS. For surfaces in contact with air or gas, crevice corrosion is not considered to be an aging mechanism at VSCNS because contaminants are not of sufficient concentration in the air or gas environments to initiate the mechanism. For carbon steel, thick-walled components

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
5 (cont.)					(≥3/8 inch) in the dry air-gas environments of the diesel generator system, general corrosion is conservatively estimated to reduce wall thickness by only 39 mils over the 60 year life of the plant, therefore, this aging mechanism will not affect the ability of these components to perform their intended functions and no program is needed to manage general corrosion of these components. At VCSNS, MIC is not considered to be an airborne contaminant, therefore, internal MIC in ventilation systems is not considered to be an aging mechanism. Consistent with NUREG-1801, this group includes carbon steel, galvanized steel, and copper-nickel in air or gas environments. In addition to these materials, this group includes copper, iron, cast iron, and ductile iron at VCSNS.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
5 (cont.)					Consistent with NUREG-1801, this group includes components in ventilation systems, diesel fuel oil systems, emergency diesel generator systems, and external surfaces of carbon steel components. In addition, this group includes components/component types from other systems where these components/component types are similar and the above discussions apply. These systems are the Fire Service System and the Component Cooling System.
6	Components in reactor coolant pump oil collect system of fire protection	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection	Yes, detection of aging effects is to be further evaluated	Whereas NUREG-1801 concerns reactor coolant pump oil collection system components/component types that are composed of carbon steel, copper, and brass, the reactor coolant pump oil collection system components/component types at VCSNS are composed of stainless steel.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
6 (cont.)					None of the components/component types of the reactor coolant pump oil collection will collect water in low spots since all are subject to high ambient conditions which would cause evaporation of any moisture. This fact is supported by a review of the operating experience, which reveals no aging effects for these components/component types. Therefore, no aging effects were determined to require management during the period of extended operation.
7	Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated.	The component /component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, and partially consistent in aging effects and credited program/ activity. The attributes of the credited program/ activity are not fully consistent with the corresponding program attributes as described in

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					NUREG-1801 because NUREG-1801 recommends augmentation of the Chemistry Program with a One-Time Inspection. The Chemistry Program (**Appendix B.1.4**) has been in effect since initial plant startup and has proven effective in maintaining systems chemistry and detecting abnormal conditions. A review of the operating experience confirms the effectiveness of the Chemistry Program (**Appendix B.1.4**) for fuel oil to manage aging effects when continued into the period of extended operation. A one-time inspection is, therefore, not warranted for these components/component types in this group. Consistent with NUREG-1801, the aging mechanisms for this group include general corrosion, pitting corrosion, crevice corrosion, and MIC. These aging mechanisms are caused by water pooling in tanks. Component/

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					component type AMR results for VCSNS show that water will not pool in the piping and other piping components, therefore, these aging mechanisms, with the exception of MIC, will not occur in these components. Only trace amounts of water are needed to promote MIC in fuel oil environments, so it is conservatively assumed to occur in piping and other piping components. In addition to these aging mechanisms, galvanic corrosion is managed for this group at VCSNS. These aging mechanisms will continue to be adequately managed through the period of extended operation. Biofouling is considered to be an aging mechanism for raw water environments only. Operating experience does not support biofouling in oil environments. Consistent with NUREG-1801, this group

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					contains carbon steel in a fuel oil environment. In addition to this material, this group contains stainless steel and copper at VCSNS. Consistent with NUREG-1801, this group includes diesel fuel oil tanks in the diesel fuel oil system and emergency diesel generator system. In addition, this group includes piping and other piping components from these systems. This group also includes tanks and piping and other piping components from the Fire Service System where these components/component types are similar and the above discussions apply.
8	Heat exchangers in chemical and volume control system	Crack initiation and growth due to SCC and cyclic loading	Water chemistry and a plant-specific verification program	Yes, plant specific	The component /component type AMR results for VCSNS are consistent with NUREG-1801 in material and environment, and partially consistent in aging effects and credited program/ activity. Consistent with NUREG-1801, the aging mechanisms for this group include

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
8 (cont.)					stress corrosion cracking. In addition to this aging mechanism, crevice corrosion, pitting corrosion, and heat exchanger fouling due to particulates are managed for this group at VCSNS. Cracking due to cyclic loading is not an aging effect requiring management during the period of extended operation at VCSNS. The Chemical and Volume Control System is operated continuously at steady state per design in a water-solid condition thereby precluding water hammer events; hence, there is no cyclic loading of the regenerative or letdown heat exchangers. The VCSNS plant specific program for managing aging is the Chemistry Program (**Appendix B.1.4**) which has been in effect since initial plant startup and has proven effective in maintaining systems chemistry and detecting abnormal conditions. A review of the operating experience

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
8 (cont.)					confirms its effectiveness in managing aging effects. A verification program is, therefore, not warranted for these components/component types in this group. Consistent with NUREG-1801, this group contains stainless steel in borated water and treated water environments. Consistent with NUREG-1801, this group contains the letdown and regenerative heat exchangers of the Chemical and Volume Control System.
9	Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity and loss of material due to general corrosion (Boral, boron steel)	Plant specific	Yes, plant specific	At VCSNS, the Boraflex neutron absorbing sheets will be replaced with Boral neutron absorbing sheets prior to the Refueling Outage 14 (September 2003). Boral does not degrade as a result of long-term exposure to radiation, and Boral is stable, durable, and corrosion resistant, therefore, there are no aging affects applicable to the Boral neutron

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
9 (cont.)					absorbing sheets in the spent fuel storage racks at VCSNS.
10	New fuel rack assembly	Loss of material due to general, pitting, and crevice corrosion	Structures monitoring	No	The New Fuel Rack Assembly at VCSNS does not perform an intended function and is not within scope of license renewal.
11	Spent fuel storage racks and valves in spent fuel pool cooling and cleanup	Crack initiation and growth due to stress corrosion cracking	Water chemistry	No	The AMR results for the spent fuel storage racks at VCSNS are consistent with NUREG-1801 in material, environment, aging effects, and credited programs/activities. Consistent with NUREG-1801, cracking due stress corrosion cracking is managed by the Chemistry Program (**Appendix B.1.4**). In addition, loss of material due to crevice corrosion and pitting corrosion are also managed by the Chemistry Program. However, the combinations of components, materials, and environments for carbon steel valves clad with stainless steel

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
11 (cont.)					identified in NUREG-1801 are not applicable to VCSNS. There are no carbon steel valves clad with stainless steel in the Spent Fuel Cooling System that are used to perform a license renewal intended function.
12	Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity due to Boraflex degradation	Boraflex monitoring	No	At VCSNS, the Boraflex neutron absorbing sheets will be replaced with Boral neutron absorbing sheets prior to Refueling Outage 14 (September 2003), therefore, this item is not applicable to VCSNS.
13	Closure bolting and external surfaces of carbon steel and low-alloy steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program/activity. Consistent with NUREG-1801, this group contains carbon and low alloy steel in an air

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
13 (cont.)					environment. Also included in this group are aluminum, galvanized steel, brass, bronze, copper, and cast iron. Consistent with NUREG-1801, this group contains external surfaces of components. Boric Acid Corrosion Surveillances (**Appendix B.1.2**) will continue to manage boric acid corrosion of external surfaces through the period of extended operation. Loss of mechanical closure integrity is not considered to be an aging effect requiring evaluation for Non-Class 1 component bolted closures within the scope for license renewal at VCSNS. As such, the specific bolting/fastener materials of subject components/component types within the scope of license renewal were not itemized as a separate Non-Class 1 component/component type. Rather, bolting was treated as a “piece-part” (or sub-component/sub-part) of Non-Class 1 components/

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
13 (cont.)					component types.
14	Components in or serviced by closed-cycle cooling water system	Loss of material due to general, pitting, and crevice corrosion, and MIC	Closed-cycle cooling water system	No	The component/component type AMR results in this group for VCSNS are consistent with NUREG-1801, in material, environment and aging effects (mechanisms) requiring management, as clarified. However, the Chemistry Program (**Appendix B.1.4**) is considered to provide adequate management in lieu of the Closed-Cycle Cooling Water System program that is recommended for this group by NUREG-1801. The Chemistry Program has been in effect since initial plant startup and has proven effective in maintaining systems chemistry and detecting abnormal conditions. A review of the operating experience confirms the effectiveness of this program to manage the conditions that could result in degradation during the period of extended operation.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
14 (cont.)					Consistent with Generic Letter 89-13, Service Water Problems Affecting Safety-Related Equipment evaluations, activities in addition to the existing chemistry controls are not warranted. Consistent with NUREG-1801, the component/component type AMR results in this group for VCSNS include components/component types in or serviced by closed-cycle cooling water systems. In addition, this group includes components/component types from systems that are not part of or not directly serviced by closed-cycle cooling water where the components/component types are similar and the same discussions apply. As such, the group includes components/component types from the Nuclear Sampling System, Radiation Monitoring System, Gaseous Waste

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
14 (cont.)					Processing System, and Reactor Coolant System (Non-Class 1 RCP thermal barrier flange and piping/tubing that are connected to the Component Cooling system). Consistent with NUREG-1801, this group includes carbon steel and stainless steel exposed to treated (non-borated) water at VCSNS, although NUREG-1801 indicates that heat exchanger components in this group are carbon steel, some stainless steel is in the heat exchangers in this group at VCSNS. Consistent with NUREG-1801, the aging effects (mechanisms) requiring management for this group include loss of material due to general (carbon steel only), crevice, and pitting corrosion. In addition to the NUREG-1801 aging effects, the following aging effects

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
14 (cont.)					(mechanisms) require management for applicable components/component types within this group at VCSNS: • Cracking due to stress corrosion cracking (SCC) of stainless steel in locations that conservatively have the potential to experience temperatures above 140°F during normal operation, • Cracking due to stress corrosion cracking (SCC) of carbon steel due to the presence of nitrite based corrosion inhibitors in the Chilled Water and Diesel Generator Cooling Water Systems, • Loss of material due to galvanic corrosion of carbon steel in contact with a more cathodic material, and • Heat exchanger tube fouling due to particulates in systems taking suction from the

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
14 (cont.)					bottom of a tank or reservoir.
15	Cranes including bridge and trolleys and rail system in load handling system	Loss of material due to general corrosion and wear	Overhead heavy load and light load handling systems	No	The VCSNS Aging Management Programs for this group are consistent with those reviewed and approved in NUREG-1801. The Material Handling System Inspection Program (**Appendix B.1.19**) is credited with managing loss of material for the steel rails and girders within the scope of license renewal, while the Maintenance Rule Structures Program (**Appendix B.1.18**) is responsible for managing the attributes associated with the aging of structures and structural components. These aging effects include loss of material due to corrosion, cracking and change in material properties, and they are detected by visual inspection of external surfaces. Wear is the result of movement of a material in

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
15 (cont.)					relation to another material in contact with the first. Wear can occur during the performance of active functions. According to the License Renewal Rule [10 CFR 54.21 (a)(1)(i)], structures and components subject to an aging management review must perform their intended functions without moving parts or without a change in configuration or properties. As such, loss of material due to wear is not an aging effect requiring further evaluation for license renewal, and is only considered a consequence of frequent or rough usage.
16	Components in or serviced by open-cycle cooling water systems	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling; buildup of deposit due to	Open-cycle cooling water system	No	The component/component type AMR results for VCSNS are consistent with this NUREG-1801 group in material and environment and are partially consistent with respect to aging effects and credited aging management program, as clarified below.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)		biofouling			Consistent with NUREG-1801, this group includes brass (not aluminum brass), copper, carbon steel and stainless steel at VCSNS. Brass and copper components/component types are also addressed later in this table for selective leaching of open and/or closed cycle cooling system components. Consistent with NUREG-1801, this group includes various components/component types in systems, or components serviced by, an open-cycle cooling water system, the Service Water System at VCSNS. Consistent with NUREG-1801, copper, brass and stainless steel components/component types within this group at VCSNS are not susceptible to a loss of material due to general corrosion and no management of that effect is

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					required for those components/component types. In addition to the aging effects specified in NUREG-1801, for this group, the credited aging management program at VCSNS also manages particulate fouling and loss of material due to erosion of susceptible components/component types. Furthermore, biofouling as is referenced in NUREG-1801, has been labeled as fouling due to biological materials at VCSNS. NUREG-1801 specifies an Open-Cycle Cooling Water System program. The VCSNS component/component type AMR results, for this group, credit the corresponding Service Water System Reliability and Inservice Testing program (**Appendix B.1.9**) to manage the specified aging effects during the period of extended operation.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
17	Buried piping and fittings	Loss of material due to general, pitting, and crevice corrosion, and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection	No Yes, detection of aging effects and operating experience are to be further evaluated	The component/component type AMR results, in this group, for VCSNS are consistent with NUREG-1801 in material, environment, aging effects (mechanisms) requiring management and credited aging management program, as clarified below. Consistent with NUREG-1801, the component/component type AMR results for VCSNS include carbon steel materials that are underground and wrapped/coated. In addition to the NUREG-1801 materials, this group includes cast iron and ductile iron (cement-lined internally) at VCSNS. In addition to the piping and fittings components/component types, this group includes tanks, and hydrants (including internal valve) at VCSNS. Also, in addition to the

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
17 (cont.)					NUREG-1801 aging effects (mechanisms) requiring management for this group, certain components/component types in this group at VCSNS are also susceptible to a loss of material due to galvanic corrosion. Consistent with NUREG-1801 options for this group, the Buried Piping and Tanks Inspection (**Appendix B.2.10**) activity is credited with the management of aging for susceptible components/component types.
18	Components in compressed air system	Loss of material due to general and pitting corrosion	Compressed air monitoring	No	No components/component types, that are subject to AMR, in compressed air systems at VCSNS were matched to NUREG-1801 items for this group since, by design, the compressed air systems at VCSNS are dry and oil-free. As such, a Compressed Air Monitoring program is not credited for aging management at VCSNS.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
19	Components (doors and barrier penetration seals) and concrete structures in fire protection	Loss of material due to wear; hardening and shrinkage due to weathering	Fire protection	No	The VCSNS Aging Management Programs for this group are generally consistent with those reviewed and approved in NUREG-1801. The VCSNS Fire Protection Program (**Appendix B.1.5**) contains many activities to achieve defense-in-depth and minimize the impact of a potential fire. The Fire Barrier and Fire Barrier Seal Inspections detect structural damage or degradation of fire barriers and fire barrier penetration sealing devices. Fire barriers include walls, ceilings and floors. The corresponding aging effects are cracking, separation from walls or components, separation of material layers, rupture or puncture of seals, shrinkage and voids. The Fire Door Inspections detect structural

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
19 (cont.)					damage or degradation of fire rated doors. Inspections are credited with managing loss of material of doors and door hardware for the period of extended operation. Excessive wear for door appurtenances such as latches, strike plates, hinges, sills and closing devices, and maintaining proper clearances (gaps) between the door, frame and threshold are also inspected, but these attributes are not credited for license renewal. Loss of material due to wear of the door hardware and hinges is not considered an aging effect but rather a consequence of frequent or rough usage.
20	Components in water-based fire protection	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling	Fire water system	No	The component/component type AMR results in this group for VCSNS are consistent with NUREG-1801 in material and environment, and are partially consistent in aging effects (mechanisms) and credited aging management program, as clarified below.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
20 (cont.)					Consistent with NUREG-1801, the component/component type AMR results for VCSNS include carbon steel, cast iron, bronze, and stainless steel. In addition to the NUREG-1801 materials, this group includes black steel, galvanized steel (conservatively considered similar to carbon/low alloy steel) and brass at VCSNS. Consistent with NUREG-1801, the aging effects (mechanisms) requiring management for this group includes loss of material due to general corrosion (carbon steel and cast iron only), crevice and/or pitting corrosion, galvanic corrosion, and microbilogically influenced corrosion (MIC). Biofouling, as it is named in NUREG-1801 for

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
20 (cont.)					this group, is referred to as “fouling due to biological materials” at VCSNS. In addition to the NUREG-1801 aging effects/mechanisms for this group, “fouling due to particulates” is managed by the credited aging management program at VCSNS. While not specified in NUREG-1801 for Fire Protection components in particular, certain components/component types in this group at VCSNS were also determined to be susceptible to a loss of material due to selective leaching based on their materials of construction (cast iron or brass) and environment. This aging effect/mechanism is also managed by the credited aging management program at VCSNS. The mechanical portion of the existing Fire

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
20 (cont.)					Protection Program (**Appendix B.1.5**) at VCSNS, referred to in this application as "Fire Protection Program - Mechanical", is credited for management of pertinent aging effects.
21	Components in diesel fire system	Loss of material due to galvanic, general, pitting, and crevice corrosion	Fire protection and fuel oil chemistry	No	Component/component type AMR results for the diesel fuel oil portion of the Fire Service system (that is tank and fuel supply line to the diesel driven fire pump) is included with the other diesel fuel oil supply portions in a separate portion of this table based on material, environment, aging effects (mechanisms) requiring management and credited aging management program similarities. As such, the NUREG-1801 items within this group are not considered applicable to VCSNS. Furthermore, the inclusion of fire service fuel oil supply with other fuel oil supply portions is conservative with respect to aging effects requiring management.

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
21 (cont.)					The pump casing of the diesel fire pump at VCSNS is not exposed to the fuel oil provided to the diesel engine, but to the fire service water. As such, the component/component type AMR results for VCSNS that include the casing of the diesel driven fire pump are included in the appropriate portion of this table.
22	Tanks in diesel fuel oil system	Loss or material due to general, pitting, and crevice corrosion	Above ground carbon steel tanks	No	The component AMR results for the external surface of the Fire Service system Diesel Fuel Oil Tank are consistent with this NUREG-1801 group with respect to material and environment and partially with respect to aging effect requiring management, as clarified below. However, an management program that includes a thickness measurement of the tank bottom surface is not required and an activity corresponding to the NUREG-1801 “Above

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
22 (cont.)					Ground Tank Inspection” is not credited. Due to environment differences (i.e. location underground or indoors), the component AMR results for other subject diesel fuel tanks are included in the appropriate portion of this table. Likewise, the internal surfaces of all these tanks are included in the appropriate portion of this table. Consistent with NUREG-1801, the carbon steel Fire Service system Diesel Fuel Oil Tank is susceptible to loss of material due to general corrosion. In addition to the NUREG-1801 aging effects requiring management, the Fire Service system Diesel Fuel Oil Tank is susceptible to a loss of material due to galvanic corrosion from a connection to a more cathodic material. However, the ambient

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
22 (cont.)					external conditions at VCSNS do not contain sufficient contaminants that could concentrate at wetted locations to cause localized crevice or pitting corrosion. Additionally, the VCSNS Fire Service system Diesel Fuel Oil Tank is mounted on stilts making all external surfaces accessible for inspection and the plant-specific Inspections for Mechanical Components (**Appendix B.2.11**) is credited with management of the specified aging effects.
23	Closure bolting	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and SCC	Bolting integrity	No	Non-Class 1 Closure bolting is considered to be a piece-part of the components/component types as a whole at VCSNS. Therefore a bolting integrity program is not credited for aging management. As a piece-part of subject components/component types at VCSNS, the specific bolting/fastener materials were not itemized as a separate Non-Class 1 component or component type. Additionally, for

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
23 (cont.)					carbon and alloy steel components, the aging management program credited for managing external general corrosion of the applicable components/component types (e.g. Inspections for Mechanical Components [Appendix B.2.11]) will also inherently address their fasteners, thus requiring no separate action.
24	Components (aluminum bronze, brass, cast iron, cast steel) in open-cycle and closed-cycle cooling water systems, and ultimate heat	Loss of material due to selective leaching	Selective leaching of materials	No	No component/component type AMR results at VCSNS are consistent with a NUREG-1801 item within this group. As such, a selective leaching of materials inspection activity is not credited for aging management. The component/component type AMR results for components in or directly serviced by the Service Water system (open-cycle cooling/ultimate heat sink), addressed previously in this table, indicate that selective leaching is not an aging effect to which they are susceptible due

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
24 (cont.)	sink				to the actual material of construction (e.g. no-alloying copper, inhibiting elements, etc.). Furthermore, although a NUREG-1801 item in this group indicates that cast steel pump casings in closed-cycle cooling systems are susceptible to selective leaching, available technical documentation and operating experience, as well as corresponding NUREG-1801 aging management program description, supports that cast steel actually refers to cast iron. VCSNS component/component type AMR results which were determined to have a susceptibility to selective leaching are addressed elsewhere in this table (brass, copper and cast iron Fire Service components/component types) and in Table 3.3-2 Item 28 of this application section (closed-cycle cooling components).

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
25	Fire barriers, walls, ceilings, and floors in fire protection	Concrete cracking and spalling due to freeze-thaw, aggressive chemical attack, and reaction with aggregates; loss of material due to corrosion of embedded steel	Fire protection and structures monitoring	No	The VCSNS Aging Management Programs for this group are generally consistent with those reviewed and approved in NUREG-1801. The VCSNS Fire Protection Program (**Appendix B.1.5**) contains many activities to achieve defense-in-depth and minimize the impact of a potential fire. The Fire Barrier Inspections detect structural damage or degradation of fire barriers. Fire barriers include walls, ceilings and floors. The corresponding aging effects are cracking (excluding hairline cracks), spalling and loss of material. The Maintenance Rule Structures Program (**Appendix B.1.18**) is responsible for managing the attributes associated with the aging of structures and structural components,

Table 3.3-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE AUXILIARY SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
25 (cont.)					including those in this grouping. These aging effects include loss of material due to corrosion, cracking and change in material properties, and are detected by visual inspection of external surfaces. Section 3.5 of the Application provides the primary discussions concerning structural monitoring.

3.3.3.3 Aging Management Evaluations That Are Different From Or Not Addressed In NUREG-1801

Aging Management Evaluations that are different from or not addressed in NUREG-1801 are identified and discussed in Table 3.3-2. The standard six-column format has been utilized. The discussion column provides additional details regarding the aging management conclusions reached by VCSNS for the component type.

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
1	Piping, tubing, valves (body/bonnet only), heat exchangers and associated components	Stainless Steel	Air-Gas (moist air)	None Identified	None Required	This grouping includes the external surface of stainless steel subject components/component types that are exposed to ambient conditions. These ambient conditions include the heat, cold, humidity and various forms of precipitation outdoors as well as the lesser heat, cold and humidity indoors (inside the Reactor Building and other in-scope plant buildings). Additionally, this grouping includes the internal surfaces of subject components/component types that are either open to the ambient conditions in the building/room in

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
1 (cont.)						which they are located or have been closed off from the process stream at ambient conditions. However, the ambient conditions in the indoor or outdoor environments at VCSNS do not contain contaminants in sufficient quantities to concentrate in wetted locations and cause corrosion/degradation of stainless steel.
2	Ductwork and HVAC pressure boundary components (including	Stainless Steel	Air-Gas (moist air)	None Identified	None Required	This grouping includes the internal and external surfaces of stainless steel ventilation system (Air Handling and Local Ventilation & Cooling)

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
2 (cont.)	housings, pipe, tubing and valves (body/bonnet only)					components/component types that are exposed to moist air (both ambient externally and conditioned internally). Stainless steel is not susceptible to corrosion/degradation in the absence of sufficient contaminant quantities, which are not found in the ambient building environments or outside environments at VCSNS.
3	Ductwork and HVAC pressure boundary components (including housings,	Galvanized Steel, Carbon Steel	Air-Gas (moist air)	None Identified	None Required	This grouping includes the internal surface of carbon and galvanized steel ventilation system (Air Handling and Local Ventilation & Cooling) components/component types that are

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
3 (cont.)	valves (body/ bonnet only)					exposed to moist air as well as the external surface of the exhaust air relief heads that are exposed to the heat, cold, humidity, and various forms of precipitation outdoors. Otherwise, the external surfaces of subject components/component types in this grouping are addressed in **Table 3.3-1 Item 5**. While the AMR results for subject components/component types in this grouping at VCSNS are consistent with NUREG-1801 with respect to material and environment, no

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
3 (cont.)						aging effects were determined to require management at VCSNS. This determination was made due to the lack of sufficient contaminants in the moist air environment to promote crevice, pitting, or microbiologically influenced corrosion in wetted locations, the resistance of galvanized steel to general corrosion in (non-wetted) air, and to the subject carbon steel components/component types being considered thick-walled such that any moisture related general corrosion will not result in loss of function during the period of extended operation.

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
4	Components/ component types in compressed gas service (CO_2 fire service extinguishing and HVAC Chillers refrigerant addition/ purge)	Black Steel, Brass, Bronze, Carbon Steel, Cast Iron, Copper, Stainless Steel	Air-Gas (dry)	None Identified	None Required	The material/environment combination for this grouping is not addressed for any item in GALL Chapters IV, V, VII or VIII. Also, no aging effects were determined to require management during the period of extended operation. This grouping includes the internal surfaces of subject components/component types in the Fire Service System (carbon dioxide extinguishing portion) and Chilled Water System (refrigerant addition and purge portion). At VCSNS, the process environment inside these portions is by design a dry

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
4 (cont.)						gas, free of moisture or contaminants, which could cause corrosion/degradation.
5	Pumps, piping, valves (body/bonnet only), and associated lubrication components (not including RCP Oil Collection)	Brass, Carbon Steel, Cast Iron, Copper, Copper-Nickel, Ductile Iron, Stainless Steel	Oil	None Identified	None Required	This grouping includes oil-coated surfaces of subject components/component types exposed to lubricating oil that are not subject to wetting.
6	Pumps, Pipe, Tube, Fittings, Valves (Body/Bonnet only) and	Stainless Steel	Treated Water (specifically chemically treated borated water)	Loss of Material due to Crevice Corrosion, Pitting Corrosion; Cracking due to Stress	Chemistry Program	This grouping includes subject component/component types in systems (Boron Recycle, Nuclear Sampling, Radiation Monitoring, Spent

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
6 (cont.)	associated components			Corrosion Cracking (SCC)		Fuel Cooling, Thermal Regeneration) whose process environment is borated water and that are either not closed-cycle cooling water systems as described in NUREG-1801, or if directly serviced by a closed-cycle cooling water system are the surface exposed to the medium being cooled. This grouping is consistent with NUREG-1801 Engineered Safety Feature items with respect to material, environment, aging effect requiring management, and credited program

**Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS
THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
6 (cont.)						(with the conservative addition of loss of material due to crevice and pitting corrosion). Only subject components/ component types in this grouping with process fluid temperatures above 140°F are susceptible to cracking due to SCC. As in NUREG-1801 for Engineered Safety Feature items, the Chemistry Program (**Appendix B.1.4**) is credited to manage the conditions that could result in the specified aging effects/mechanisms during the period of extended operation.

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
7	Nuclear Sampling system sample cooling tubes	Nickel-Based Alloy (Alloy 600)	Treated Water (including Borated Water, secondary side blowdown, Component Cooling Water)	Loss of Material due to Crevice Corrosion, Pitting Corrosion; Cracking due to Stress Corrosion Cracking (SCC); and Heat Exchanger Fouling due to Particulates (Component Cooling water only)	Chemistry Program	The components in this grouping at VCSNS are consistent with a NUREG-1801 Reactor Coolant item (**Table 3.1-1 Item 15**) with respect to material and environment and partially for aging effect requiring management. However, this material is not addressed in NUREG-1801, except for Reactor Coolant, and VCSNS evaluations have determined that the tubes in this grouping are not susceptible to primary water stress corrosion cracking (PWSCC) that is normally considered in a high temperature

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
7 (cont.)						borated water environment. The existing Chemistry Program (Appendix B.1.4) is credited for managing the conditions that could result in the specified aging mechanisms during the period of extended operation and provides reasonable assurance that the component intended functions will be maintained for all CLB conditions.
8	Pipe and Fittings	Stainless Steel, Carbon Steel	Embedded in Concrete	None Identified	None Required	This grouping includes the external surface of subject components/component types (Spent Fuel Cooling, Nuclear Plant Drains, and Service Water systems)

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
8 (cont.)						that are embedded in concrete. The concrete forms a tight seal that prevents exposure of the subject piping to aggressive chemical species that could cause corrosion.
9	Ductwork, Piping, Fittings and/or Tubing which enters or passes between buildings below the 425’ Elevation	Stainless Steel	Air-Gas (moist air)	Loss of Material Microbiologically Influenced Corrosion (MIC) {External Surface}	Maintenance Rule Structures Program	At VCSNS, piping or tubing component types which enter buildings or pass between buildings that do not share a common wall below the 425’ elevation are conservatively susceptible to the accumulation of external sulfate-reducing-bacteria resulting from groundwater intrusion. With the exception of those components types which pass

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
9 (cont.)						through a “fire seal” penetration, that are not susceptible to groundwater intrusion, these components may see a build-up of sulfate reducing bacteria at the wall/component interface and thereby be susceptible to localized external MIC. The Maintenance Rule Structures Program (Appendix B.1.18) will manage a loss of material due to external MIC for susceptible locations during the period of extended operation.
10	Piping, muffler, expansion joints in	Carbon Steel, Stainless Steel	Air-Gas (primarily moist air plus periodic	None Identified	None Required	This grouping includes the exhaust piping, mufflers and expansion joints

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
10 (cont.)	Diesel engine exhausts (Fire Service (FS) and Diesel Generator Services (DG) systems)		exhaust gases)			associated with the Diesel Driven Fire Service Pump and Emergency Diesel Generators. While this grouping is consistent with a NUREG-1801 item with respect to material and environment. However, due to these thick-walled components not being continuously exposed to hot exhaust gases, any general corrosion would be insufficient to cause loss of component intended function during the period of extended operation. Thin-walled carbon steel portions of these exhaust

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
10 (cont.)						trains are addressed in **Table 3.3-1 Item 5**.
11	Piping, tubing, valves (body/bonnet only) and associated components	Carbon Steel, Stainless Steel	Air-Gas	None Identified	None Required	This grouping includes various materials in the intake/starting air (Diesel Generator Services System), control air (Service Air System), and gaseous waste/effluent (Liquid and Gaseous Waste Processing Systems) that are normally exposed to an air or gas environment where moisture is not controlled through design. However, at VCSNS, these components/component types are stainless steel and not susceptible to strictly moisture related degradation or are

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
11 (cont.)						thick-walled carbon steel and not prone to a loss of function from strictly moisture related degradation since a protective coating is formed, slowing the corrosion rate, and the allowable minimum wall thickness would not be exceeded during the period of extended operation.
12	Piping, tubing, valves (body/bonnet only) and associated components	Aluminum, Brass, Copper	Air-Gas	None Identified	None Required	This grouping includes various materials in the intake/ starting air of the Diesel Generator Services System that are normally exposed to a moist air environment. However, moisture alone is insufficient to cause the degradation of

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
12 (cont.)						the materials within this grouping. Furthermore, the components/component types are in locations that are not susceptible to the wetting required for degradation.
13	Instrument Air system piping, tubing and valves	Aluminum, Brass, Carbon Steel, Cast Iron, Stainless Steel	Air-Gas	None Identified	None Required	This grouping includes various materials in the Instrument Air System (IA) that are normally exposed to compressed air that is, by design, dry, oil-free, and filtered. As such, no aging effects require management for the subject components/component types in this grouping during the period of extended operation since the design

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
13 (cont.)						precludes moisture collection in these components/ component types. The AMR results for component types in this system, as well as other compressed control air systems where moisture related aging is a concern during the period of extended operation are addressed separately below.
14	Piping, Tubing and Valves (body/ bonnet only)	Carbon Steel	Air-Gas (compressed air w/ moisture)	Loss of Material due to General Corrosion or Galvanic Corrosion (components in contact with more cathodic material)	Service Air System Inspection	This grouping includes subject component types in compressed air systems (Service Air, Instrument Air, Building Services Systems) where moisture is not controlled through system design. The Service Air

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
14 (cont.)						System Inspection (**Appendix B.2.6**) will manage loss of material due to general corrosion (oxygen and moisture) or galvanic corrosion for component types in this grouping. This program is a one-time inspection and will provide reasonable assurance during the period of extended operation that the component intended function(s) will be maintained under all CLB conditions
15	Gaseous Waste Processing system helical coils,	Stainless Steel	Air-Gas, Treated Water	Loss of Material due to Crevice and Pitting Corrosion; Cracking due to Stress	Waste Gas System Inspections	While somewhat consistent with NUREG-1801 with respect to material, environment and aging effect(s) requiring management, this

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
15	manifolds, pipe and valves (body/ bonnet only)			Corrosion Cracking (SCC)		grouping includes components/component types that are subject to collection of water (moisture) in an environment that is not controlled by the Chemistry Program (**Appendix B.1.4**). As such the one-time Waste Gas System Inspection (**Appendix B.2.8**) is credited for the management of the identified aging effects during the period of extended operation.
16	Fire Service system piping	Cement-lined ductile iron	Raw Water	None Identified	None Required	This grouping includes Fire Service Main Header piping that is exposed to raw water and is normally stagnant awaiting system actuation. While cement is

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
16 (cont.)						addressed for Civil/Structural components in NUREG-1801, the use of cement lined pipe in the Fire Service System precludes the degradation of that piping as the cement provides a protective layer for the piping and is not susceptible to the extreme conditions (temperature, aggressive chemicals, etc.) which may cause degradation of cement.
17	Pumps, Piping, Tubing, Fittings, Valves (body/ bonnet only)	Stainless Steel	Treated Water (including makeup water, secondary side blowdown, cooling water)	Loss of Material due to Crevice and Pitting Corrosion; Cracking due to Stress Corrosion	Chemistry Program	This grouping includes subject components/component types of auxiliary systems that are not closed cycle cooling systems (Nuclear Sampling, Reactor

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
17 (cont.)				Cracking (SCC)		Makeup Water) but whose process fluid is treated water. This grouping is consistent with NUREG-1801 Engineered Safety Features items with respect to material, environment, aging effect requiring management and credited program, with the conservative addition of crevice and pitting corrosion. As in NUREG-1801 for the associated Engineered Safety Features items, the Chemistry Program (**Appendix B.1.4**) manages the conditions that could lead to the specified aging effects/ mechanisms during the

**Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS
THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
17 (cont.)						period of extended operation.
18	Fire Service system nozzles, piping and fire hydrants	Galvanized Steel, Carbon Steel, Cast Iron	Air-Gas (air)	None Identified	None Required	This grouping includes fire service system portions (those not constructed of stainless steel) that are normally in a drained down condition awaiting system actuation and therefore normally exposed to ambient conditions rather than raw water. While the air is not strictly dry, the moisture content is expected to be very low. Additionally, the components are thick-walled such that the minimal resulting corrosion, if

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
18 (cont.)						any, would result in loss of component function during the period of extended operation.
19	Tanks (including Component Cooling Water Surge Tank, Chilled Water Expansion Tanks)	Carbon Steel	Air-Gas (moist air), Treated Water	Loss of Material due to Crevice, Pitting, General, Galvanic Corrosion and due to the Corrosive Impacts of Alternate Wetting and Drying	Chemistry Program, Above Ground Tank Inspection	The components in the grouping are partially consistent with a NUREG-1801 Closed Cycle Cooling Water item with respect to material, environment, and aging effect requiring management. However, additional aging effects require management at VCSNS for this Auxiliary grouping. NUREG-1801 does not address the corrosive impacts of alternate wetting and drying of vented tanks (or tanks without a

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
19 (cont.)						cover gas) that could concentrate contaminants above bulk fluid concentrations and result in further degradation. The Chemistry Program (**Appendix B.1.4**) will manage the conditions required for a loss of material of the components in this grouping to occur in bulk fluid (treated water) concentrations. In addition, the one-time Above Ground Tank Inspection (**Appendix B.2.1**) will manage a loss of material due to the corrosive impacts of alternate wetting and drying as well

**Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS
THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
19 (cont.)						as general or galvanic corrosion in the tank and attached components above the normal water level.
20	Tanks (including Reactor Makeup Water Storage Tank, Nuclear Sampling Flush Water Storage Tank)	Stainless Steel	Air-Gas (moist air), Treated Water	Loss of Material due to Crevice or Pitting Corrosion; Cracking due to Stress Corrosion Cracking (SCC); Corrosive Impacts of Alternate Wetting and Drying	Chemistry Program, Above Ground Tank Inspection	The components in the grouping are partially consistent with a NUREG-1801 Engineered Safety Feature item with respect to material, environment, aging effect requiring management and aging management program (Chemistry Program [Appendix B.1.4]). However, additional aging effects require management at VCSNS for this Auxiliary grouping. NUREG-1801 does not

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
20 (cont.)						address the corrosive impacts of alternate wetting and drying of vented tanks (or tanks without a cover gas) that could concentrate contaminants above bulk fluid concentrations and result in further degradation. The Chemistry Program (Appendix B.1.4) will manage the conditions required for a loss of material or cracking of the components in this grouping to occur in bulk fluid concentrations. In addition, the one-time Above Ground Tank Inspection

**Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS
THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
20 (cont.)						(Appendix B.2.1) will manage the corrosive impacts of alternate wetting and drying to occur in stainless steel in the treated water environment.
21	Piping, Valves (body/ bonnet only), Heat Exchanger tubes	Stainless Steel	Liquid Waste/ Drain Water (including borated, treated water)	Loss of Material due to Crevice and Pitting Corrosion and/or Cracking due to Stress Corrosion Cracking (SCC)	Liquid Waste System Inspection	This grouping includes subject component types in systems (Liquid Waste Processing, Nuclear Plant Drains) that contain/transport liquid waste or drain water that has the potential for contamination. This grouping is consistent with a NUREG-1801 Engineered Safety Feature item with respect to material, environment, and aging effect requiring

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
21 (cont.)						management with the addition of loss of material due to crevice and pitting corrosion. However, the one-time Liquid Waste System Inspection (**Appendix B.2.3**) will provide reasonable assurance that the component intended function(s) will be maintained under CLB conditions during the period of extended operation.
22	Drain Pan and Drain Piping	Stainless Steel	Treated Water (condensate quality water with traces of boric acid)	Loss of Material due to Crevice and Pitting Corrosion	Reactor Building Cooling Unit Inspection	This grouping includes the loop seal drain lines on the Reactor Building Cooling Units (RBCUs), Roof Drain System, and is consistent with an Engineered Safety Feature NUREG-1801 item

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
22 (cont.)						with respect to material and environment. However, due to temperatures well below 140°F, the drain lines are not susceptible to cracking due to SCC but are susceptible to a loss of material due to crevice or pitting corrosion. The one-time Reactor Building Cooling Units Inspection (**Appendix B.2.5**) will provide reasonable assurance that the component function of these drains will be maintained during the period of extended operation.
23	Sight glass (body only)	Glass	Air-Gas (including moist air, starting air), Oil	None Identified	None Required	This grouping includes the external and internal surfaces of sight glasses and

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
23 (cont.)						glass portions of pressure retaining instruments that is inert and not susceptible to age related degradation.
24	Motor Bearing Cooler Fins	Aluminum	Air-Gas (moist air)	Heat Exchanger Fouling due to Particulates	Preventive Maintenance Activities – Ventilation Systems Inspection	This grouping includes the fins of the fin/tube type Component Cooling Water Pump motor integral coolers. Particulate fouling of the air side can occur from the accumulation and build-up of dust, dirt or debris on and between the fins of air coolers. The Preventive Maintenance Activities - Ventilation Systems Inspections (Appendix B.1.26) will manage heat exchanger fouling due to particulates for these

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
24 (cont.)						aluminum components during the period of extended operation, as well as for the other portions of the coolers, as included in **Table 3.3.1 Item 14**.
25	Valves (body/ bonnet only) and piping components	Aluminum, Brass, Copper	Air-Gas (moist air)	None Identified	None Required	This grouping includes the external surface of subject components/component types that are in locations (such as the Diesel Generator Building) where exposure to leaking borated water is not a possibility and the ambient conditions in those locations do not contain contaminants in sufficient quantities to result in other forms of corrosion to the materials in this

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
25 (cont.)						grouping.
26	Flexible Hose and Coupling	Rubber	Oil/Fuel Oil, Treated Water	None Identified	None Required	This grouping includes the internal surface of rubber components/component types (Diesel Generator Services System) that are not considered to be susceptible to degradation in fluid environments due to the lack of excessive temperatures and to the change in material properties of elastomers being more closely tied to external conditions such as ultraviolet radiation. The external surfaces of these components are addressed as applicable in

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
26 (cont.)						Table 3.3.1 Item 2.
27	Piping, tubing and fittings	Brass, Copper	Fuel Oil	Loss of Material due to Microbiologically Influenced Corrosion (MIC)	Chemistry Program	This grouping includes subject components/component types exposed to fuel oil in locations that are not susceptible to water pooling (Diesel Generator Services System). Although the components/component types in this grouping are not susceptible to water pooling, and thereby to related degradation mechanisms, the Chemistry Program (Appendix B.1.4) is conservatively credited for the management of the conditions that could lead to MIC in the fuel oil environment.

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
28	Cooling unit tubes and fittings	Brass, Copper, Copper Nickel	Treated Water	Loss of Material due to crevice, pitting and/or galvanic corrosion, as well as erosion-corrosion, and selective leaching; Heat exchanger fouling due to particulates; Cracking due to stress corrosion cracking (SCC)	Chemistry Program & Heat Exchanger Inspections	This grouping includes subject cooling components/ component types with a treated water process fluid on one side and a non water process fluid on the other (Air Handling, Local Ventilation and Cooling, Chilled Water, Component Cooling, Diesel Generator Services Systems). Also included are brass components exposed to closed cycle cooling water that are susceptible to selective leaching due to their material compositions. The material, environment, aging effect requiring management for this grouping

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
28 (cont.)						are consistent with a NUREG-1801 item for heat exchangers between open-cycle systems and closed-cycle systems, except that the surface not exposed to treated (closed-cycle) water is not an open-cycle system, as described in NUREG-1801, and therefore not within the scope of the corresponding aging management program. At VCSNS, the Chemistry Program (**Appendix B.1.4**) is considered to provide adequate management of the conditions which could result in crevice, pitting or

Table 3.3-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE AUXILIARY SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
28 (cont.)						galvanic corrosion during the period of extended operation. In addition to the Chemistry Program (**Appendix B.1.4**), the one-time Heat Exchanger Inspection (**Appendix B.2.12**) is credited for the characterization of a loss of material due to erosion-corrosion, heat exchanger fouling due to particulates (components in systems taking suction from the bottom of a tank), if any, and loss of material due to selective leaching.

3.3.4 REFERENCES

3.3-1	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.
3.3-2	NUREG-1801, “Generic Aging Lessons Learned Report,” Volumes 1 and 2, NRC, April 2001.
3.3-3	NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” NRC, April 2001.

3.4 AGING MANAGEMENT OF STEAM AND POWER CONVERSION SYSTEMS

3.4.1 SYSTEM DESCRIPTION

The Steam and Power Conversion Systems act as a heat sink to remove heat from the nuclear steam supply system and convert the heat generated in the reactor to the plant's electrical output.

The Steam and Power Conversion Systems include the following systems:

- Auxiliary Boiler Steam and Feedwater System
- Condensate System
- Emergency Feedwater System
- Extraction Steam System
- Feedwater System
- Gland Seal Steam System
- Main Steam System
- Main Steam Dump System
- Steam Generator Blowdown System

3.4.2 AGING MANAGEMENT REVIEW

3.4.2.1 Methodology

Aging management review of Steam and Power Conversion Systems components and commodities involved consideration and evaluation of the materials, environments, and stressors that are associated with each component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [**Reference 3.4-1**]. The VCSNS AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to VCSNS materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the VCSNS methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in NUREG-1801 [**Reference 3.4-2**] and evaluated for consistency with NUREG-1801 programs that are relied on for license renewal. The results are documented and discussed in Table 3.4-1 using the format suggested by the NRC Standard Review Plan for License Renewal (NUREG-1800) [**Reference 3.4-3**].

3.4.2.2 Operating Experience

Site: VCSNS site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

Industry: An evaluation of industry operating experience published since the effective date of NUREG-1801 was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

On-Going: On-going review of plant-specific and industry operating experience is performed in accordance with the plant Operating Experience Program.

3.4.3 AGING MANAGEMENT PROGRAM

3.4.3.1 Aging Management Programs Evaluated In NUREG-1801 That Are Relied On For License Renewal

Table 3.4-1 shows the component and commodity groups (combinations of materials and environments), and aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Steam and Power Conversion Systems. The table is based on Table 3.4-1 of NUREG-1800 [**Reference 3.4-3**] and provides a discussion of the applicability of the component commodity group and details regarding the degree to which VCSNS aging management programs are consistent with those recommended in NUREG-1801. The discussion section includes (1) information regarding the applicability of NUREG-1801 component/commodity group to VCSNS, (2) any issues recommended in NUREG-1801 that require further evaluation, (3) details regarding VCSNS components to be included in the component/commodity group, and (4) any additional materials to be added to the component/commodity groups beyond those identified in NUREG-1801.

3.4.3.2 Further Evaluation Of Aging Management As Recommended By NUREG-1801

Further evaluation of aging management as recommended by NUREG-1801 has been incorporated into the “Discussion” column of Table 3.4-1. A cross-reference is provided to the section of the application where TLAAs are discussed.

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
1	Piping and fittings in main feedwater line, steam line and auxiliary feedwater (AFW) piping (PWR only)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	The TLAA is applicable to Class 2 and 3 piping at VCSNS. See **Section 4.3.2** for the TLAA discussion of Class 2 and 3 piping.
2	Piping and fittings, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head, and	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated	The component /component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effects, and partially consistent in program. The attributes of the credited program/activity are not fully consistent with the corresponding program attributes as described in NUREG-1801 because NUREG-1801 recommends augmentation of the Chemistry Program with a One Time Inspection. The Chemistry Program has

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)	shell (except main steam system)				been in effect since initial plant startup and has proven effective in maintaining systems chemistry and detecting abnormal conditions. A review of the operating experience confirms the effectiveness of the Chemistry Program (**Appendix B.1.4**) for treated water to manage aging effects when continued into the period of extended operation. A one-time inspection is, therefore, not warranted for the majority of components/component types in this group. The only exception is the Condensate Storage Tank which is inspected by the Above Ground Tank Inspections (**Appendix B.2.1**) activity. This activity also inspects the interior of the tank above the water line. Consistent with NUREG-1801, the aging mechanisms for this group include general corrosion, pitting corrosion, and crevice corrosion. In addition to these aging mechanisms, galvanic corrosion,

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)					stress corrosion cracking, and corrosive effects of alternate wetting and drying are managed at VCSNS. Consistent with NUREG-1801, this group includes stainless steel and carbon steel in treated water. In addition to these materials, this group includes low alloy steel and nickel-based metal at VCSNS. Consistent with NUREG-1801, this group includes piping and fittings, valve bodies, pump casings, and tanks of various systems within the steam and power conversion grouping. In addition, this group includes components/component types from the Nuclear Sampling System where these components/component types are similar and the above discussions apply.
3	AFW piping	Loss of material due to general,	Plant specific	Yes, plant specific	The combinations of components, materials, and environments identified in NUREG-1801

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3 (cont.)		pitting, and crevice corrosion, MIC, and biofouling			are not applicable to VCSNS. The AFW (Emergency Feedwater at VCSNS) piping at VCSNS is not exposed to untreated water. The Service Water System provides emergency backup to the Emergency Feedwater System through automatic isolation valves that normally provide boundary isolation between the treated water of the Emergency Feedwater System and the untreated water of the Service Water System. For a description of the aging effects of carbon steel in an untreated water environment refer to Aging Management of Auxiliary Systems in **Section 3.3**.
4	Oil coolers in AFW system (lubricating oil side possibly contaminated	Loss of material due to general (carbon steel only), pitting, and crevice corrosion,	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material and environment. Water and contaminants will not intrude into the oil environments for these components/component

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
4 (cont.)	with water)	and MIC			types. This supported by a review of the operating experience, which reveals no aging effects for these components/component types. Therefore, no aging effects were determined to require management during the period of extended operations. Consistent with NUREG-1801, this group contains carbon steel and stainless steel in an oil environment. In addition to these materials, this group contains cast iron at VCSNS. Consistent with NUREG-1801, this group contains AFW (Emergency Feedwater System at VCSNS) bearing oil system components.
5	External surface of carbon steel components	Loss of material due to general corrosion	Plant specific	Yes, plant specific	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, and aging effect. However, NUREG-1801 recommends plant

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
5 (cont.)					specific evaluation of the credited program. The credited program/activity at VCSNS is Inspections for Mechanical Components (**Appendix B.2.11**). Consistent with NUREG-1801 this program/activity will detect and manage loss of material due to general corrosion. In addition, this program will detect and manage loss of material due to galvanic corrosion. An additional program, Maintenance Rule Structures Program (**Appendix B.1.18**), will detect and manage loss of material due to microbiologically influenced corrosion (MIC) on external surfaces at susceptible locations (in contact with groundwater). Consistent with NUREG-1801, this group contains carbon steel in an ambient, moist air environment. In addition to carbon steel, this group contains low alloy steel and cast iron at

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
5 (cont.)					VCSNS. Consistent with NUREG-1801, this group contains external surfaces for various components/component types.
6	Carbon steel piping and valve bodies	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and program. Consistent with NUREG-1801, this group contains carbon steel in a treated water environment. In addition to carbon steel, this group contains low alloy steel at VCSNS. Consistent with NUREG-1801, this group includes valve bodies and piping and fittings of various systems within the steam and power conversion grouping. The Flow Accelerated Corrosion Monitoring Program (**Appendix B.1.6**) manages this aging effect/mechanism at VCSNS.

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7	Carbon steel piping and valve bodies in main steam system	Loss of material due to pitting and crevice corrosion	Water chemistry	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. Consistent with NUREG-1801, the aging mechanisms for this group are pitting corrosion and crevice corrosion. In addition to these mechanisms, general corrosion and galvanic corrosion are managed at VCSNS. Consistent with NUREG-1801, this group contains carbon steel in a treated water environment. Consistent with NUREG-1801, this group contains valve bodies and piping and fittings. The Chemistry Program (**Appendix B.1.4**) manages this aging effect/mechanism at VCSNS.
8	Closure bolting in	Loss of material due to general	Bolting integrity	No	Non-Class 1 Closure bolting is considered to be a piece-part of the components/component

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
8 (cont.)	high-pressure or high-temperature systems	corrosion; crack initiation and growth due to cyclic loading and/ or SCC.			types as a whole at VCSNS. Therefore a bolting integrity program is not credited for aging management. As a piece-part of subject components/component types at VCSNS, the specific bolting/fastener materials were not itemized as a separate Non-Class 1 component or component type. Additionally, for carbon and alloy steel components, the aging management program credited for managing external general corrosion of the applicable components/component types (e.g. Inspections for Mechanical Components [Appendix B.2.11]) will also inherently address their fasteners, thus requiring no separate action.
9	Heat exchangers and coolers/ condensers serviced by	Loss of material due to general (carbon steel only), pitting, and crevice corrosion,	Open-cycle cooling water system	No	Open Cycle Cooling Water Systems as described by NUREG-1801 are not used in any Steam and Power Conversion Systems at VCSNS.

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
9 (cont.)	open-cycle cooling water	MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	Open Cycle Cooling Water Systems as described by NUREG-1801 are not used in any Steam and Power Conversion Systems at VCSNS.
10	Heat exchangers and coolers/ condensers serviced by closed-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Closed-cycle cooling water system	No	Closed Cycle Cooling Water Systems as described by NUREG-1801 are not used in any Steam and Power Conversion Systems at VCSNS.
11	External surface of above ground condensate storage tank	Loss of material due general (carbon steel only), pitting, and crevice corrosion	Above ground carbon steel tanks	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, and aging effect. However, rather than an above ground carbon steel tanks program as recommended by NUREG-1801, general corrosion on external surfaces is detected and managed by

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
11 (cont.)					Inspections for Mechanical Components (Appendix B.2.11) at VCSNS. Also, at VCSNS the ambient environment in the yard environment does not contain contaminants in sufficient quantities that could be concentrated in wetted locations to cause pitting corrosion and crevice corrosion, therefore, loss of material due to pitting corrosion and crevice corrosion has not been identified as an aging effect. Consistent with NUREG-1801, this group contains carbon steel in a yard environment. Consistent with NUREG-1801, this group contains external surfaces of the above ground Condensate Storage Tank.
12	External surface of buried condensate	Loss of material due to general, pitting, and crevice corrosion, and	Buried piping and tanks surveillance or Buried piping and	No Yes, detection of	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. The condensate storage

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
12 (cont.)	storage tank and AFW piping	MIC	tanks inspection	aging effects and operating experience are to be further evaluated	tank is above ground, however, there is underground piping in the AFW (Emergency Feedwater System at VCSNS). Consistent with NUREG-1801, this group contains carbon steel in an underground (buried) environment. Consistent with NUREG-1801, this group includes external surfaces of piping and fittings. The Buried Pipe and Tanks Inspection program (**Appendix B.2.10**) will manage these aging effects/mechanisms at VCSNS.
13	External surface of carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	The component/component type AMR results for VCSNS are consistent with NUREG-1801 in material, environment, aging effect, and credited program. Consistent with NUREG-1801, this group

Table 3.4-1:

SUMMARY OF AGING MANAGEMENT PROGRAMS FOR THE STEAM & POWER CONVERSION SYSTEMS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
13 (cont.)					contains external surfaces of carbon and low alloy steel components/component types in an air environment where there is a potential for leaking and dripping chemically treated borated water. In addition to these metals, this group also includes cast iron at VCSNS. The Boric Acid Corrosion Surveillances (**Appendix B.1.2**) will manage this aging effect/mechanism at VCSNS.

3.4.3.3 Aging Management Evaluations That Are Different From Or Not Addressed In NUREG-1801

Aging Management Evaluations that are different from or not addressed in NUREG-1801 are identified and discussed in Table 3.4-2. The standard six-column format has been utilized. The discussion column provides additional details regarding the aging management conclusions reached by VCSNS for the component type.

**Table 3.4-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE STEAM & POWER CONVERSION SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
1	Piping and piping system components	Stainless Steel	Ambient, moist air	None Identified	None Required	This grouping includes external surfaces of stainless steel piping system components exposed to a moist air environment. At VCSNS, the ambient environment of the yard and sheltered environments do not contain contaminants of sufficient concentration to cause aging effects that require aging management
2	Piping and piping system components	Aluminum, Brass	Ambient, moist air	Boric Acid Corrosion / Aggressive Chemical Attack	Boric Acid Corrosion Surveillances	The Boric Acid Corrosion Surveillances (**Appendix B.1.2**) will manage boric acid corrosion on external surfaces of components that are susceptible to this aging mechanism. No other aging mechanisms were

Table 3.4-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE STEAM & POWER CONVERSION SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
2 (cont.)						identified for this material/ environment combination at VCSNS.
3	Piping and piping components and heat exchanger tubes	Aluminum, Brass	Oil	None Identified	None Required	Water and contaminants will not intrude into the oil environments for these components/component types. This is supported by review of the operating experience, which reveals no aging effects for these components/component types. Therefore, no aging effects were determined to require management during the period of extended operations.
4	EFWP Turbine (Casing	Carbon Steel	Ambient, moist air	General Corrosion	Preventive Maintenance Activities:	The Emergency Feedwater Pump turbine is normally in

**Table 3.4-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE STEAM & POWER CONVERSION SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
4 (cont.)	Only), Valve (Body Only) EFWP Turbine Governor valve				Terry Turbine	a standby condition whereby the internal surfaces of these components are exposed to an ambient, moist air environment. Preventive Maintenance Activities: Terry Turbine (**Appendix B.1.25**) will manage this aging effect.
5	Valves (Bodies)	Stainless Steel	Treated Water	Crevice Corrosion, Pitting Corrosion, Stress Corrosion Cracking	Chemistry Program	The Chemistry Program (**Appendix B.1.4**) will manage crevice corrosion, pitting corrosion, and stress corrosion cracking for stainless steel valve bodies in a treated water environment during the period of extended operation.

Table 3.4-2:
SUMMARY OF AGING MANAGEMENT EVALUATIONS FOR THE STEAM & POWER CONVERSION SYSTEMS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program/Activity	Discussion
6	Heat Exchanger EF Pump Turbine Lube Oil - Tube	Brass	Treated Water	Crevice Corrosion, Galvanic Corrosion, Pitting Corrosion, Selective Leaching, Stress Corrosion Cracking	Chemistry Program, Heat Exchanger Inspections	The Chemistry Program (**Appendix B.1.4**) will manage crevice corrosion, pitting corrosion, and galvanic corrosion of brass heat exchanger tubes in a treated water environment during the period of extended operation. Heat Exchanger Inspections (**Appendix B.2.12**) will manage selective leaching of brass heat exchanger tubes in a treated water environment during the period of extended operation.

3.4.4 REFERENCES

3.4-1	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.
3.4-2	NUREG-1801, “Generic Aging Lessons Learned Report,” Volumes 1 and 2, NRC, April 2001.
3.4-3	NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” NRC, April 2001.

3.5 AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

3.5.1 CONTAINMENT, STRUCTURES, AND COMPONENT SUPPORTS DESCRIPTIONS

The Containment, Structures and Component Supports applicable to License Renewal at VCSNS include the following (for plant location see **Site Facilities Drawing**):

- Reactor Building (Internal Structures and Class 1 Component Supports)
- Auxiliary Building (includes Hot Machine Shop, Refueling Water Storage Tank & Reactor Makeup Water Storage Tank foundations and West Penetration Access Area)
- Control Building
- Intermediate Building (includes East Penetration Access Area)
- Diesel Generator Building
- Fuel Handling Building
- Turbine Building
- Service Water Pumphouse, Intake and Discharge Structures
- Yard Structures (includes Fire Service Pumphouse, Condensate Storage Tank foundation, and Electrical Manhole MH-2)
- Earthen Embankments (includes Service Water Pond North Dam, South Dam, East Dam, West Embankment and North Berm)
- Electrical Substation and Relay House

3.5.1.1 Reactor Building

The Reactor Building is a post tensioned, reinforced concrete structure with an integral steel liner. The Reactor Building consists of a cylindrical wall, a shallow dome roof and a foundation mat with a depressed incore instrumentation pit under the reactor vessel. The foundation mat bears on fill concrete that extends to competent rock. At the underside of the Reactor Building foundation mat a tendon access gallery is formed into the top of the fill concrete. A retaining wall, extending approximately one-quarter of the way around the Reactor Building, protects the below grade portions of the Reactor Building wall from the subgrade. Adjacent buildings surround the remaining three-quarters of the Reactor Building. The internal structures of the Reactor Building consist of the primary shield wall surrounding and supporting the reactor vessel; secondary shield walls surrounding and laterally supporting each steam generator and the pressurizer; refueling cavity and fuel transfer canal; mezzanine floor and operating floor, both consisting of concrete slabs supported by structural steel framing; polar crane supports; and concrete basement slab supported by the structural foundation mat.

3.5.1.2 Auxiliary Building

The Auxiliary Building superstructure is a reinforced concrete shear wall (box type) structure containing five main floor levels above the foundation and extending up to elevation 485'-0" (designated as the roof). Above this level is another story composed primarily of a metal clad structural steel braced frame, but with limited areas continuing the reinforced concrete construction employed below. The foundation is comprised of a reinforced concrete structural mat which is supported on fill concrete down to competent bedrock. A waterproofing membrane is provided between the structural mat concrete and fill concrete because of the depth of the foundation below the ground water table.

3.5.1.3 Control Building

The Control Building superstructure is a steel frame structure with concrete exterior shear walls containing four main floor levels and a concrete roof. The foundation system for the Control Building (CB) is comprised of a reinforced concrete mat which is supported on fill concrete down to competent bedrock.

3.5.1.4 Intermediate Building

The Intermediate Building superstructure is a "L" shaped reinforced concrete shear wall (box type) structure containing two main floor levels above the foundation and extending up to a low roof. Above the low roof is a partial third floor of reinforced concrete. The foundation system for the Intermediate Building (IB) is comprised of a reinforced concrete basement floor slab that acts in conjunction with a series of grade beams to transfer vertical loads to reinforced concrete caissons, shear/bearing walls, and concrete piers. The shear/bearing wall foundations and reinforced concrete caissons are founded on competent bedrock. The piers are founded on fill concrete that extends beyond the Reactor Building and Auxiliary Building.

3.5.1.5 Diesel Generator Building

The Diesel Generator Building superstructure is a reinforced concrete shear wall (box type) structure containing three main floor levels above the foundation mat. The operating floor is where the diesel generators are located, with their foundations extending down to the basement floor mat. The foundation system for the Diesel Generator Building consists of a reinforced concrete slab and grade beam system that is supported by reinforced concrete caissons drilled into competent bedrock.

3.5.1.6 Fuel Handling Building

The Fuel Handling Building superstructure is a steel frame superstructure containing two main floor levels and a roof. The foundation system for the Fuel Handling Building is comprised of a reinforced concrete mat formed by the bottom of the Spent Fuel Pool and Fuel Cask Pit. The foundation mat is supported by reinforced concrete piers that extend to the fill concrete adjacent to the Reactor and Auxiliary Buildings and by reinforced concrete caissons that extend to competent rock on the north and east sides.

3.5.1.7 Turbine Building

The Turbine Building is a non Seismic Category I building. The superstructure (steel framing, metal siding and metal roof deck) is supported on a reinforced concrete substructure. The foundation for the Turbine Building is mostly comprised of a reinforced concrete mat supported by Zone III fill (graded crushed stone) material. The turbine generator pedestal foundation mat is founded on fill concrete.

3.5.1.8 Service Water Structures

The Service Water Pumphouse (SWPH) superstructure is a reinforced concrete building (containing three floor levels) constructed within the West Embankment of the Service Water Pond. The foundation is comprised of a reinforced concrete structural mat. The entire structural mat is supported on compact fill which extends from the underside of the mat. The compacted fill is supported on in-situ soils (saprolite), to decomposed rock, to competent rock. The SWPH structure is separate from the Service Water Intake Structure (SWIS) and from buried connecting pipes and electrical duct banks by flexible joints, which accommodate relative settlement and seismic movement. The SWIS is a reinforced concrete rectangular box culvert (tunnel) with two reinforced concrete wing walls at the intake end. The foundation for the SWIS forms the floor of the tunnel and is comprised of a reinforced concrete mat. The mat is supported by compacted fill material except for a portion of the inlet end which rests on in-situ soils. The Service Water Discharge Structure (SWDS) is a reinforced concrete rectangular basin mostly buried within the West Embankment of the Service Water Pond. The foundation is comprised of a reinforced concrete mat. The base mat bears partly on decomposed rock and partly on fill concrete that extends to the decomposed rock.

3.5.1.9 Yard Structures

Yard Structures include the Condensate Storage Tank Foundation, Electrical Manhole EH-2, and the Fire Service Pumphouse. The Condensate Storage Tank (CST) foundation consists of a reinforced concrete mat which is supported by a Zone III fill (graded crushed stone) material. A reinforced concrete ring wall extends above the foundation mat which secures the CST with anchor bolts. Electrical Manhole EH-2 is a reinforced concrete structure which contains electrical duct banks for Class 1E cables. The manhole is mostly embedded below the yard grade, with reinforced concrete walls and foundation mat. The Fire Service Pumphouse is a concrete block building (with composite roof) founded upon the reinforced concrete Circulating Water Intake Structure (CWIS). This non-safety structure is included within License Renewal in compliance with 10CFR50.48 (Fire Protection). The Electrical Substation yard, with the exception of the paved roadways, is covered with several inches of "crusher run" stone and is enclosed by a perimeter fence. The Electrical Substation and Transformer Area are included within License Renewal in compliance with the NRC revised staff position on the scoping of SBO equipment for license renewal dated 4/1/2002.

3.5.1.10 Earthen Embankments

The Earthen Embankments (North Dam, South Dam, and East Dam and the West Embankment) form the Service Water Pond. These earthen structures are Seismic Category I and were designed to satisfy the intent of Regulatory Guides 1.27 and 1.29. The three dams and West Embankment are homogeneous earth structures with fill material consisting of residual soil and saprolite which was excavated from local borrow sources. The shoreline along Monticello Reservoir north of the plant and west of the North Dam has an earthen dike (North Berm) constructed above the site grade. The North Berm is classified as a non-seismic, non-nuclear safety-related structure. The primary function of the North Berm is to protect the site from the potential of external flooding from Monticello Reservoir (due to probable maximum precipitation and wave run-up).

3.5.1.11 Class 1 Component Supports

Class 1 Component Supports are those supports for major equipment and Class 1 piping that are subject to aging management review including: Class 1 piping supports and major equipment supports (pressurizer base flange and upper lateral supports; reactor vessel supports; steam generator vertical, lower lateral, and upper lateral supports; and reactor coolant pump lateral and vertical support assemblies).

3.5.2 AGING MANAGEMENT REVIEW

3.5.2.1 Methodology

Aging management review of Containment, Structures and Component Supports involved consideration and evaluation of the materials, environments, and stressors that are associated with each structure, component, or commodity grouping under review, as discussed in Section 4.2 of NEI 95-10 [**Reference 3.5-1**]. The VCSNS AMR methodology follows the approach recommended in NEI 95-10 and is based on generic industry guidance for determining aging effects for both mechanical and civil/structural components. The guidance represents a set of rules that allow the evaluator to identify aging effects for a given material and environment combination. The material and environment-based rules in the generic industry guidance documents are derived from known age-related degradation mechanisms and industry operating experience. The guidance was reviewed for applicability to VCSNS materials of construction and component internal and external operating environments and was used to identify aging effects for components, structures, and commodities. The results of the evaluation of materials and environment combinations, using the VCSNS methodology, are aging effects; and, if the aging effects adversely affect intended functions, the results are aging effects requiring management for the applicable components and commodities. Aging effects that require management are correlated to aging management programs.

The aging management review identifies one or more aging management programs to be used to demonstrate that the effects of aging will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. The programs to be used for managing the effects of aging were compared to those listed in NUREG-1801 [**Reference 3.5-2**] and evaluated for consistency with NUREG-1801 programs that are relied on for license renewal. The results are documented and discussed in Table 3.5-1 using the format suggested by the NRC Standard Review Plan for License Renewal (NUREG-1800) [**Reference 3.5-3**].

3.5.2.2 Operating Experience

Site: VCSNS site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) Maintenance Rule Data Base, and (4) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

Industry: An evaluation of industry operating experience published since the effective date of NUREG-1801 was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

On-Going: On-going review of plant-specific and industry operating experience is performed in accordance with the plant Operating Experience Program.

3.5.3 AGING MANAGEMENT PROGRAM

3.5.3.1 Aging Management Programs Evaluated In NUREG-1801 That Are Relied On For License Renewal

Table 3.5-1 shows the aging management groups (combinations of components, materials and aging effects) and the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the Containment, Class I Structures, and Component Supports at VCSNS. Note that this table only includes those components, materials and aging effects that are applicable to a PWR. The VCSNS comparison to NUREG-1801 (including clarifications and exceptions) is included in the “Discussion” column.

3.5.3.2 Further Evaluation Of Aging Management As Recommended By NUREG-1801

Further evaluation of aging management as recommended by the NUREG-1801 has been incorporated into the “Discussion” column of Table 3.5-1.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
Common Components of All Types of PWR Containment					
1	Penetration sleeves, penetration bellows, and dissimilar metal welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	VCSNS does not evaluate fatigue for penetration sleeves, bellows or dissimilar metal welds; therefore, a TLAA evaluation is not applicable. **Penetration Sleeves** meet the requirements of ASME Section III, comply with GDC-51, and behave in a non-brittle manner. **Penetration Bellows** are used in hot penetrations at VCSNS but do not provide containment isolation since they are located within the penetration on the exterior side of containment. Hot penetrations are sealed on the inside of containment by a flat plate welded to both the penetration sleeve and process pipe (similar to cold penetrations), thus providing

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
1 (cont.)					containment isolation without the use of a resilient or flexible seal. **Dissimilar Metal Welds** are materials and not components. VCSNS penetration sleeves and process pipes use similar (SA) materials.
2	Penetration sleeves, bellows, and dissimilar metal welds	Cracking due to cyclic loading; crack initiation and growth due to SCC	Containment inservice inspection (ISI) and containment leak rate test	Yes, detection of aging effects is to be evaluated	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), 10 CFR 50 Appendix J Leak Rate Testing (**Appendix B.1.12**) and Containment ISI Program - IWE/IWL (**Appendix B.1.16**) are consistent with those reviewed and approved in NUREG-1801. Stress Corrosion Cracking (SCC) requires a combination of a corrosive environment, susceptible materials, and high tensile stresses. (1) VCSNS penetration sleeves are not

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)					subject to high tensile stresses or aggressive chemicals during normal operation, while similar metal welds are used between penetration sleeves and process pipes; therefore, SCC is not an applicable aging effect requiring management. (2) VCSNS hot penetration bellows do not perform a pressure boundary function nor incorporate a flexible seal assembly on the inboard side of containment. They do provide structural and/or functional support for process piping on the outboard side of containment; therefore, in the unlikely event of SCC in the bellows, the intended functions are not affected.
3	Penetration sleeves, penetration bellows, and	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), 10 CFR 50 Appendix J Leak Rate Testing (**Appendix B.1.12**) and

**Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3 (cont.)	dissimilar metal welds				Containment ISI Program - IWE/IWL (**Appendix B.1.16**) are consistent with those reviewed and approved in NUREG-1801.
4	Personnel airlock and equipment hatch	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), 10 CFR 50 Appendix J Leak Rate Testing (**Appendix B.1.12**) and Containment ISI Program - IWE/IWL (**Appendix B.1.16**) are consistent with those reviewed and approved in NUREG-1801.
5	Personnel airlock and equipment hatch	Loss of leak tightness in closed position due to mechanical wear of locks, hinges, and closure mechanisms	Containment leak rate test and plant technical specifications	No	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), 10 CFR 50 Appendix J Leak Rate Testing (**Appendix B.1.12**) and Containment ISI Program - IWE/IWL (**Appendix B.1.16**) are consistent with those reviewed and approved in NUREG-1801.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
5 (cont.)					Operation of hatches is governed by VCSNS Technical Specifications. Plant operational experience has not identified any fretting or seal degradation. Locks, hinges, and closure mechanisms are active components; therefore, mechanical wear is not considered an aging effect.
6	Seals, gaskets, and moisture barriers	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI and Containment leak rate test	No	Loss of sealing is not considered an aging effect, but rather a consequence of elastomer degradation which is managed by the programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), 10 CFR 50 Appendix J Leak Rate Testing (**Appendix B.1.12**), Containment ISI Program - IWE/IWL (**Appendix B.1.16**) and Maintenance Rule Structures Program (**Appendix B.1.18**) which are consistent with those reviewed and

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
6 (cont.)					approved in NUREG-1801. Component types include: moisture barrier, compressible joints and seals used for seismic gaps, and fire barrier seals. These components are managed by the Containment ISI, IWE and Maintenance Rule Structures Programs.
PWR Concrete (Reinforced and Prestressed) and Steel Containment					
7	Concrete elements: foundation, dome, and wall	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide, aggressive chemical attack, and	Containment ISI	Yes, if aging mechanism is significant for inaccessible areas	**ACCESSIBLE AREAS** VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**) and Containment ISI Program - IWE/IWL (**Appendix B.1.16**) are consistent with those reviewed and approved in NUREG-1801.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)		corrosion of embedded steel			**Leaching of Calcium Hydroxide** from reinforced concrete becomes significant only if the concrete is exposed to flowing water. Resistance to leaching is enhanced by using a dense concrete with low permeability and well cured. The VCSNS containment structure is not exposed to flowing water and designed in accordance with ACI-318 and constructed in accordance with ACI-301 and ASTM Standards, which provides a good quality, dense, low permeability concrete. Leaching has been identified in the accessible containment Tendon Access Gallery (due to groundwater infiltration) and is managed by the programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**) and Containment ISI Program - IWE/IWL (**Appendix B.1.16**).

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					**Aggressive Chemical Attack** becomes significant to concrete exposed to an aggressive environment (Chlorides > 500 ppm, Sulfates > 1500 ppm, and pH < 5.5). Resistance to mild acid attack is enhanced by using a dense concrete with low permeability and a low water-to-cement ratio of less than 0.50. The VCSNS containment structure uses a dense, low permeable concrete with a maximum water-to-cement ratio of 0.48, which provides an acceptable degree of protection against aggressive chemical attack. VCSNS is not located in areas exposed to sulfate or chloride attack, nor located near industrial plants whose emissions would change environmental parameters and cause degradation to concrete. The water chemical

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					analysis results confirm that the site groundwater is mildly acidic but considered to be non-aggressive. Therefore, loss of material due to aggressive chemical attack is not an aging effect requiring management for accessible containment concrete structures. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. **Corrosion of Embedded Steel** becomes significant if exposed to an aggressive environment (Concrete pH < 11.5 and Chlorides > 500 ppm). Corrosion is not significant if the concrete has a low water-to-cement ratio, low permeability, and designed in accordance with ACI Standards (ACI-318 or ACI-349). The

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					design and construction of the VCSNS containment structure (in accordance with accepted ACI Standards) prevents corrosion of embedded steel from occurring; therefore, this aging effect does not require management for accessible areas. **<u>INACCESSIBLE AREAS</u>** Inaccessible Areas at VCSNS do not require a plant-specific aging management program for leaching of calcium hydroxide, aggressive chemical attack or corrosion of embedded steel due to the following: • Containment concrete surfaces are not exposed to flowing water and the in-place concrete is constructed to design

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					requirements in accordance with ACI recommendations (at the time of construction) which produced a dense concrete with low permeability. Specific reference to ACI 201.2R-77 is not made since the plant was designed and constructed prior to 1977. • Concrete is not exposed to a below grade environment which is considered aggressive. Refer above to “Aggressive Chemical Attack” for “Accessible Areas.” Additionally, VCSNS used a concrete design mix with maximum water-cement ratio of 0.44 - 0.48 which is specified by ACI Standards to be chemically resistant and watertight. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is

Table 3.5-1:
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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
7 (cont.)					not aggressive.
8	Concrete elements: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures monitoring	No, if within the scope of the applicant's structures monitoring program	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), Containment ISI Program - IWE/IWL (**Appendix B.1.16**) and Maintenance Rule Structures Program (**Appendix B.1.18**) are consistent with those reviewed and approved in NUREG-1801. The VCSNS containment foundation is constructed directly on competent bedrock and is not subject to settlement; therefore, aging management is not required.
9	Concrete elements: foundation	Reduction in foundation strength due to erosion of	Structures monitoring	No, if within the scope of the applicant's	The VCSNS containment foundation does not use porous concrete and is not subject to flowing water; therefore, aging management is

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
9 (cont.)		porous concrete subfoundation		structures monitoring program	not required.
10	Concrete elements: foundation, dome, and wall	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete containment that exceed specified temperature limits	The VCSNS containment concrete elements are not exposed to temperatures which exceed the thresholds for degradation; therefore, reduction of strength and modulus due to elevated temperatures are not aging effects requiring management.
11	Prestressed containment: tendons and anchorage components	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21 (c)	Yes, TLAA	VCSNS Aging Management Programs: Containment ISI Program - IWE/IWL (**Appendix B.1.16**) and Tendon Surveillance Program (**Appendix B.3.3**) are consistent with those reviewed and approved in NUREG-1801. VCSNS Containment tendons have been determined to be a TLAA in accordance with 10 CFR 54.3. Refer to **Section 4.5**.

**Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
12	Steel elements: liner plate and containment shell	Loss of material due to corrosion in accessible and inaccessible areas	Containment ISI and containment leak rate test	Yes, if corrosion is significant for inaccessible areas	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), 10 CFR 50 Appendix J Leak Rate Testing (**Appendix B.1.12**), Containment Coating Monitoring And Maintenance Program (**Appendix B.1.15**) and Containment ISI Program - IWE/IWL (**Appendix B.1.16**) are consistent with those reviewed and approved in NUREG-1801. Corrosion for inaccessible areas (embedded containment liner) is not significant because: • Concrete meeting the requirements of ACI-318 or ACI-349 and the guidance of ACI-201.2R was used for the containment concrete in contact with the embedded containment liner.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
12 (cont.)					• The concrete is monitored under Maintenance Rule Structures Program and IWL to ensure that it is free of penetrating cracks. • The moisture barrier is monitored under IWE for aging degradation. • Borated water leakage in the containment structure is not a common occurrence and is monitored under the aging management program Boric Acid Corrosion Surveillances (**Appendix B.1.2**).
13	Steel elements: protected by coating	Loss of material due to corrosion in accessible areas only	Protective coating monitoring and maintenance	No	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B1.11**), Containment Coating Monitoring and Maintenance Program (**Appendix B.1.15**), Containment ISI Program - IWE/IWL (**Appendix B.1.16**) and Maintenance Rule Structures Program (**Appendix B.1.18**) are consistent with those

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SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
13 (cont.)					reviewed and approved in NUREG-1801.
14	Prestressed containment: tendons and anchorage components	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI	No	VCSNS Aging Management Programs: Containment ISI Program - IWE/IWL (**Appendix B.1.16**) and Tendon Surveillance Program (**Appendix B.3.3**) are consistent with those reviewed and approved in NUREG-1801.
15	Concrete elements: foundation, dome, and wall	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI	No	**ACCESSIBLE AREAS** VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**), Containment ISI Program - IWE/IWL (**Appendix B.1.16**), and Maintenance Rule Structures Program (**Appendix B.1.18**) are consistent with those reviewed and approved in NUREG-1801.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
15 (cont.)					**Freeze-thaw** is not an aging effect requiring management for the containment structure at VCSNS, since it is not exposed to saturated water conditions and is designed and constructed to acceptable ACI and ASTM Standards. **Reaction with Aggregates** for the containment structure at VCSNS is mitigated by carefully designed and selected concrete constituents; therefore, expansion and cracking due to reaction with aggregates is not an aging effect requiring management. **<u>INACCESSIBLE AREAS</u>** **Inaccessible Areas** at VCSNS do not require

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
15 (cont.)					a plant-specific aging management program for freeze-thaw or reaction with aggregates due to the following: • VCSNS lies within the “moderate” weathering region of the U. S. (as defined in ASTM C33) and its containment concrete is not exposed to saturated water conditions near the ground surface which eliminates freeze-thaw considerations. Additionally, The concrete used at VCSNS is designed with entrained air content of between 4% and 6% in conformance with ACI-301, and IWL inspections have not identified any concrete degradation related to freeze-thaw. • Aggregates used in concrete at VCSNS were carefully selected (using local

Table 3.5-1:
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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
15 (cont.)					quarries) to mitigate aggregate reactions, incorporating design specifications in conformance with accepted ACI and ASTM Standards.
Class I Structures					
16	All Groups except Group 6: accessible interior/exterior concrete and steel components	All types of aging effects	Structures monitoring	No, if within the scope of the applicant's structures monitoring program	**ACCESSIBLE AREAS** VCSNS Aging Management Programs: described in Containment Coating Monitoring and Maintenance Program (**Appendix B.1.15**), Containment ISI Program - IWE/IWL (**Appendix B.1.16**) and Maintenance Rule Structures Program (**Appendix B.1.18**) are consistent with those reviewed and approved in NUREG-1801. **Freeze-thaw** is not an aging effect requiring

**Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					management for these groups of structures at VCSNS, since they are not exposed to saturated water conditions and are designed and constructed to acceptable ACI and ASTM Standards. **Leaching of Calcium Hydroxide** from reinforced concrete becomes significant only if the concrete is exposed to flowing water. Resistance to leaching is enhanced by using a dense concrete with low permeability. These groups of structures at VCSNS are not exposed to flowing water and designed in accordance with ACI-318 and constructed in accordance with ACI-301 and ASTM Standards, which provides a good quality, dense, low permeability concrete.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					**Reaction with Aggregates** for concrete for these groups of structures at VCSNS are mitigated by carefully designed and selected concrete constituents; therefore, expansion and cracking due to reaction with aggregates is not an aging effect requiring management. **Corrosion of Embedded Steel** becomes significant if exposed to an aggressive environment (Concrete pH < 11.5 and Chlorides > 500 ppm). Corrosion is not significant if the concrete has a low water-to-cement ratio, low permeability, and designed in accordance with ACI Standards (ACI-318 or ACI-349). The design and construction of these groups of structures at VCSNS prevents corrosion of embedded steel from occurring; therefore, this aging effect does not require management for

**Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES
AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					accessible areas. **Aggressive Chemical Attack** becomes significant to concrete exposed to an aggressive environment (Chlorides > 500 ppm, Sulfates > 1500 ppm, and pH < 5.5). Resistance to mild acid attack is enhanced by using a dense concrete with low permeability and a low water-to-cement ratio of less than 0.50. These groups of structures at VCSNS use a dense, low permeable concrete with a maximum water-to-cement ratio of 0.48, which provides an acceptable degree of protection against aggressive chemical attack. VCSNS is not located in areas exposed to sulfate or chloride attack, nor located near industrial plants whose emissions would change environmental parameters and cause degradation to

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					concrete. The water chemical analysis results confirm that the site groundwater is considered to be non-aggressive; therefore, loss of material due to aggressive chemical attack is not an aging effect requiring management. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. **Corrosion** of structural steel components is managed by the Maintenance Rule Structures Program (**Appendix B.1.18**). **Lubrite** materials are not used at VCSNS in the reactor pressure vessel supports; therefore, aging management is not required.

**Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					**<u>INACCESSIBLE AREAS</u>** **Inaccessible Areas** at VCSNS do not require a plant-specific aging management program for freeze-thaw, leaching of calcium hydroxide, reaction with aggregates, corrosion of embedded steel or aggressive chemical attack due to the following: • VCSNS lies within the “moderate” weathering region of the U. S. (as defined in ASTM C33) and its concrete is not exposed to saturated water conditions near the ground surface which eliminates freeze-thaw considerations. Additionally, the concrete used at VCSNS is designed with entrained air content of between 4% and 6% in conformance with ACI-301, and Maintenance Rule

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					inspections have not identified any degradation related to freeze-thaw. • Concrete surfaces are not exposed to flowing water and the in-place concrete is constructed with design requirements in accordance with ACI recommendations (at the time of construction) which produced a dense concrete with low permeability. Specific reference to ACI 201.2R-77 is not made since the plant was designed and constructed prior to 1977. • Aggregates used in concrete at VCSNS were carefully selected (using local quarries) to mitigate aggregate reactions, incorporating design specifications in conformance with accepted ACI and ASTM Standards. • Concrete is not exposed to a below grade

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
16 (cont.)					environment which is considered aggressive. Additionally, VCSNS used a concrete design mix with maximum water-cement ratio of 0.44 - 0.48 which is specified by ACI Standards to be chemically resistant and watertight. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive.
17	Groups 1-3, 5, 7-9; inaccessible concrete components, such as	Aging of inaccessible concrete areas due to aggressive chemical attack, and corrosion of	Plant specific	Yes, if an aggressive below-grade environment exists	**Aggressive Chemical Attack** becomes significant to concrete exposed to an aggressive environment (Chlorides > 500 ppm, Sulfates > 1500 ppm, and pH < 5.5). Resistance to mild acid attack is enhanced by using a dense concrete with low permeability and a low water-to-cement ratio of less than 0.50. These groups of structures at VCSNS use a dense,

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
17 (cont.)	exterior walls below grade and founda-tion	embedded steel			low permeable concrete with a maximum water-to-cement ratio of 0.48, which provides an acceptable degree of protection against aggressive chemical attack. The water chemical analysis results confirm that the site groundwater is considered to be non-aggressive. Therefore, loss of material due to aggressive chemical attack is not an aging effect requiring management for inaccessible concrete components. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive. **Corrosion of Embedded Steel** becomes significant if exposed to an aggressive environment (Concrete pH < 11.5 and Chlorides

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
17 (cont.)					> 500 ppm). Corrosion is not significant if the concrete has a low water-to-cement ratio, low permeability, and designed in accordance with ACI Standards (ACI-318 or ACI-349). The design and construction (in accordance with accepted ACI Standards) of these groups of structures at VCSNS prevents corrosion of embedded steel from occurring; therefore, this aging effect does not require management for inaccessible areas.
18	Group 6: all accessible/ inaccessible concrete, steel, and earthen	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of water-control structures or FERC/US Army Corp of Engineers dam inspection	No	**ACCESSIBLE AREAS** VCSNS Aging Management Programs: Maintenance Rule Structures Program (**Appendix B.1.18**) and Service Water Pond Dam Inspection Program (**Appendix B.1.21**) are consistent with those reviewed and approved in NUREG-1801. In addition, VCSNS

**Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)	components	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	and maintenance	No	incorporates other programs: Service Water Structures Survey Monitoring Program (**Appendix B.1.22**) and Underwater Inspection Program (SWIS and SWPH) (**Appendix B.1.23**). **Freeze-thaw** is an aging effect requiring management for these structures at VCSNS, since they are exposed to saturated water conditions. This aging effect is minimized since these structures are designed and constructed to acceptable ACI and ASTM Standards. **Leaching of Calcium Hydroxide** from reinforced concrete becomes significant only if the concrete is exposed to flowing water. Resistance to leaching is enhanced by using a dense concrete with low permeability and well

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					cured. These structures at VCSNS are designed in accordance with ACI-318 and constructed in accordance with ACI-301 and ASTM Standards, which provides a good quality, dense, low permeability concrete. **Reaction with Aggregates** for concrete for these structures at VCSNS are mitigated by carefully designed and selected concrete constituents; therefore, expansion and cracking due to reaction with aggregates is not an aging effect requiring management. **Corrosion of Embedded Steel** becomes significant if exposed to an aggressive environment (Concrete pH < 11.5 and Chlorides > 500 ppm). Corrosion is not significant if the concrete has a low water-to-cement ratio, low

Table 3.5-1:
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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					permeability, and designed in accordance with ACI Standards (ACI-318 or ACI-349). The design and construction (in accordance with accepted ACI Standards) of these structures at VCSNS prevents corrosion of embedded steel from occurring; therefore, this aging effect does not require management for accessible areas. **Aggressive Chemical Attack** becomes significant to concrete exposed to an aggressive environment (Chlorides > 500 ppm, Sulfates > 1500 ppm, and pH < 5.5). Resistance to mild acid attack is enhanced by using a dense concrete with low permeability and a low water-to-cement ratio of less than 0.50. These structures at VCSNS use a dense, low permeable concrete with a maximum water-to-cement

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SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					ratio of 0.48, which provides an acceptable degree of protection against aggressive chemical attack. VCSNS is not located in areas exposed to sulfate or chloride attack, nor located near industrial plants whose emissions would change environmental parameters and cause degradation to concrete. The water chemical analyses results confirm that the site groundwater, pond and reservoir are considered to be non-aggressive; therefore, loss of material due to aggressive chemical attack is not an aging effect requiring management. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not aggressive.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					**Settlement** of these structures at VCSNS is monitored and managed by the programs: Service Water Pond Dam Inspection Program (**Appendix B.1.21**) and Service Water Structures Survey Monitoring Program (**Appendix B.1.22**). **Abrasion and Cavitation** due to flowing water are considered insignificant at VCSNS due to the low flow conditions designed for these structures; therefore, aging management is not required. **Corrosion** of structural steel components is managed by the programs: Maintenance Rule Structures Program (**Appendix B.1.18**) and Underwater Inspection Program (SWIS and

Table 3.5-1:
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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					SWPH) (**Appendix B.1.23**). **Earthen Structures** are monitored and managed by the programs: Service Water Pond Dam Inspection Program (**Appendix B.1.21**) and Service Water Structures Survey Monitoring Program (**Appendix B.1.22**). **INACCESSIBLE AREAS** **Inaccessible Areas** at VCSNS do not require a plant-specific aging management program for freeze-thaw, leaching of calcium hydroxide, reaction with aggregates, corrosion of embedded steel or aggressive chemical attack due to the following: • VCSNS lies within the “moderate”

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SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					weathering region of the U. S. (as defined in ASTM C33). These concrete structures are exposed to saturated water conditions near the ground surface; however, the concrete used at VCSNS is designed with entrained air content of between 4% and 6% in conformance with ACI-301, and Maintenance Rule inspections have not identified any degradation related to freeze-thaw. • Portions of concrete surfaces for these structures are exposed to flowing water; however, the in-place concrete was constructed with design requirements in accordance with ACI recommendations at the time of construction which produce a dense concrete with low permeability. Specific reference to ACI 201.2R-77 is not

**Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					made since the plant was designed and constructed prior to 1977. • Aggregates used in concrete at VCSNS were carefully selected (using local quarries) to mitigate aggregate reactions, incorporating design specifications in conformance with accepted ACI and ASTM Standards. • Concrete is not exposed to a below grade environment which is considered aggressive. Additionally, VCSNS used a concrete design mix with maximum water-cement ratio of 0.44 - 0.48 which is specified by ACI Standards to be chemically resistant and watertight. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
18 (cont.)					environment is not aggressive.
19	Group 5: liners	Crack initiation and growth due to SCC; loss of material due to crevice corrosion	Water chemistry and monitoring of spent fuel pool water level	No	VCSNS Aging Management Programs: Chemistry Program (**Appendix B.1.4**) and Maintenance Rule Structures Program (**Appendix B.1.18**) are consistent with those reviewed and approved in NUREG-1801.
20	Groups 1-3, 5, 6: all masonry block walls	Cracking due to restraint, shrinkage, creep, and aggressive environment	Masonry wall	No	Masonry walls at VCSNS are inspected in accordance with the Maintenance Rule Structures Program (**Appendix B.1.18**). No masonry walls are used in nuclear safety-related structures at VCSNS.
21	Groups 1-3, 5, 7-9: foundation	Cracks, distortion, and increases in component stress	Structures monitoring	No, if within the scope of the applicant's structures monitoring	VCSNS Aging Management Program: Maintenance Rule Structures Program (**Appendix B.1.18**) is consistent with those reviewed and approved in NUREG-1801.

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
21 (cont.)		level due to settlement		program	Concrete Tanks are not used at VCSNS; therefore, aging management is not required.
22	Groups 1-3, 5-9: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures monitoring	No, if within the scope of the applicant's structures monitoring program	The VCSNS structure foundations do not use porous concrete; therefore, aging management is not required. Concrete Tanks are not used at VCSNS; therefore, aging management is not required.
23	Groups 1-5: concrete	Reduction in strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete that exceed specified temperature limits	The VCSNS structural concrete elements are not exposed to temperatures which exceed the thresholds for degradation; therefore, reduction of strength and modulus due to elevated temperatures are not aging effects requiring management.
24	Group 7, 8: liners	Crack initiation and growth due to	Plant specific	Yes	Group 7 (Concrete Tanks) are not used at VCSNS; therefore, aging management is not

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
24 (cont.)		SCC; loss of material due to crevice corrosion			required. Note that the combinations of components, materials, and environments identified in NUREG-1801 for Group 8 (Steel Tanks) are not applicable to VCSNS; therefore, aging management is not required.
Component Supports					
25	All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted	Aging of component supports	Structures monitoring	No, if within the scope of the applicant's structures monitoring program	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**) and Maintenance Rule Structures Program (**Appendix B.1.18**) are consistent with those reviewed and approved in NUREG-1801. Concrete structures and concrete components can be subjected to cyclic loading and

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
25 (cont.)	connections, etc.				therefore, can be subjected to fatigue degradation. However, concrete components have good fatigue strength properties for hundreds or thousands or cycles of below yield load application (high cycle low-level loads). For components that may be subjected to vibratory or cyclic loading, proper design eliminates or compensates for vibration and cyclic loading. In addition, vibration characteristically leads to cracking in a short period of time, on the order of hours to days of operation. For example, a component with 1 Hertz vibratory load will be subjected to 10^7 cycles in four months of service, so that failure, should it occur, is probable early in life for vibratory stresses above the endurance limit. Because this time period is short when compared to the overall plant operational life, any cracking

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
25 (cont.)					would be identified and corrected to prevent recurrence long before the period of extended operation. This type of degradation is limited to a small set of components and is corrected as discovered with inspections of similar locations and configurations to ensure the event is location specific or a one-time event. The potential for cracking induced by other cyclic loads, such as thermal cycling of the supported system, is implicitly considered in structural steel design through the specification of conservative design allowable stresses that account for a minimum of 10^5 load cycles. VCSNS concrete components are designed in accordance with accepted ACI Standards and have good low cycle fatigue properties. Plant

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SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
25 (cont.)					experience did not identify any concrete degradation due to service-induced loads. Therefore, cracking due to fatigue is not an aging effect requiring management for concrete components.
26	Groups B1.1, B1.2, and B1.3: support members: anchor bolts, and welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	TLAA is not applicable since a CLB fatigue analyses does not exist for these component types at VCSNS.
27	All Groups: support members: anchor bolts and welds	Loss of material due to boric acid corrosion	Boric acid corrosion	No	VCSNS Aging Management Programs: Boric Acid Corrosion Surveillances (**Appendix B.1.2**), 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**) and Maintenance Rule Structures Program

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
27 (cont.)					(**Appendix B.1.18**) are consistent with those reviewed and approved in NUREG-1801.
28	Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators	Loss of material due to environmental corrosion; loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI	No	VCSNS Aging Management Programs: 10 CFR 50 Appendix J General Visual Inspection (**Appendix B.1.11**) and Maintenance Rule Structures Program (**Appendix B.1.18**) are consistent with those reviewed and approved in NUREG-1801. Loss of mechanical function, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads; and elastomer hardening are not considered as aging effects but rather design issues.
29	Groups B1.1: high strength low-alloy	Crack initiation and growth due to SCC	Bolting integrity	No	Bolting integrity at VCSNS is inspected in accordance with the ISI Program (IWF): ASME Section XI ISI Program – IWF

Table 3.5-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
29 (cont.)	bolts				(Appendix B.1.13). Industry experience has shown that high strength bolts (bolts with tensile strength greater than 150 ksi) installed in Class 1 component supports could be susceptible to SCC in humid environments like the Reactor Building. The key factors necessary for SCC include high-strength materials, moist environments, and a high level of sustained tensile stress. Operating experience also shows that improperly heat-treated anchor bolts have been susceptible to SCC, especially when under a high preload (full preload of 70% of ultimate strength). Anchor bolts are also exposed to concrete where chlorides can leach-out and attack the intergranular structure of the bolts over time. Therefore based on

Table 3.5-1:
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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
29 (cont.)					industry experience, stress corrosion cracking is a potential aging effect for the ASTM A490 high strength anchor bolts used in the Class 1 component supports at VCSNS. However, SCC of high strength anchor bolts should also be considered as a negligible aging effect at VCSNS since the following conditions apply: • ASTM A490 anchor bolt material is properly heat-treated by conforming to ASTM Specification A490 through a certified mill test report. • Anchor bolts are tightened snug-tight as defined by AISC; therefore, for bolts greater than 1" in diameter, a significant preload (in the order of 70% of ultimate

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AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
29 (cont.)					strength) is not practical to develop. • Anchor bolts do not have a high level of sustained tensile stress as evidenced by lower faulted condition design loads due to elimination of dynamic effects subsequent to postulated High Energy Line Break (HELB) of the Reactor Coolant System primary coolant piping.

3.5.3.3 Aging Management Evaluations That Are Different From Or Not Addressed In NUREG-1801

Table 3.5-2 contains Containment, Structures and Component Supports aging management review results that are not specifically addressed in NUREG-1801. This table includes component types, materials, environments and aging effects requiring management, along with the programs and activities for managing aging.

Table 3.5-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
1	Battery Racks	Carbon Steel	Internal	Loss of Material (General Corrosion)	Battery Rack Inspection Program	Battery Racks are not explicitly identified in GALL as a structural component type. VCSNS uses a Battery Rack Inspection Program (**Appendix B.1.14**) to inspect for corrosion or other physical damage to ensure integrity, thus providing acceptable aging management. This is a plant specific program which is not addressed in the GALL.
2	Caissons (Foundations)	Concrete	Below Grade	None	Not Applicable	Reinforced Concrete Caissons are used to support the foundations of the Intermediate, Diesel Generator, and Fuel Handling

**Table 3.5-2:
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THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
2 (cont.)						Buildings at VCSNS. Concrete Caissons are inaccessible since they are completely surrounded by backfill (beneath the structure foundations) and embedded in underlying bedrock. Aging management programs are not required since the below grade environment is considered to be non-aggressive. Periodic monitoring of the below grade water chemistry will be conducted during the period of extended operation to demonstrate that the below-grade environment is not

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SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS
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AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
2 (cont.)						aggressive. Settlement is not considered significant since the caissons are structurally embedded within the underlying bedrock.
3	Flood Barriers	Elastomers	Internal, External (Below Grade)	Cracking, Change in Material Property	Fire Protection Program, Maintenance Rule Structures Program	The VCSNS Flood Barrier Inspection Program (**Appendix B.1.17**) is not evaluated in the GALL and is plant specific. Flood barrier inspections are performed as part of the Fire Protection Program (**Appendix B.1.5**) and Maintenance Rule Structures Program (**Appendix B.1.18**).

Table 3.5-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
4	Lubrite Plates (Class 1 Pipe Hanger Supports)	Lubricant	Internal	None	Not Applicable	Lubrite plates have been used in a few Class 1 pipe hanger supports at VCSNS. Lubrite plates are not used in the reactor pressure vessel support shoes as described in the GALL. Lubrite materials for nuclear applications are designed to resist deformation, have a low coefficient of friction, resist softening at elevated temperatures, resist corrosion, withstand high intensities of radiation, and will not score or mar; therefore, they are not susceptible to aging effects requiring management.

Table 3.5-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
5	Pressure Doors	Carbon Steel	Internal, External	Loss of Material (General Corrosion)	Pressure Door Inspection Program, Fire Protection Program	The VCSNS Pressure Door Inspection Program (**Appendix B.1.20**) is not evaluated in the GALL and is plant specific. Pressure door inspections are performed as part of the Pressure Door Inspection and Fire Protection Programs.
6	RHR and Spray Isolation Chamber Valve Guard Pipes	Carbon Steel	Below Grade	Loss of Material (Microbiologically Induced Corrosion [MIC])	Containment ISI Program – IWE/ IWL	VCSNS has identified MIC as an aging effect requiring management for these component types. MIC has not been evaluated within the GALL. The Containment ISI Program (IWE) was effective in identification of this aging effect and

Table 3.5-2:
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AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
6 (cont.)						will be used in the future to provide acceptable aging management.
7	Service Water Intake Structure (SWIS)	Concrete	Raw Water (Flowing)	Loss of Material (Abrasion, Cavitation)	Underwater Inspection Program (SWIS & SWPH)	Underwater inspections are not evaluated in the GALL. VCSNS has a plant specific Underwater Inspection Program (SWIS and SWPH) (Appendix B.1.23) as part of the CLB which monitors the SWIS for settlement cracking, abrasion and cavitation. These underwater visual inspections are conducted every 5 years and provide for acceptable aging management.

Table 3.5-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
8	Service Water Pump-house (SWPH)	Carbon Steel	Raw Water	Loss of Material (General Corrosion, Pitting, MIC)	Maintenance Rule Structures Program, Underwater Inspection Program (SWIS & SWPH)	For Water-Control Structures, the GALL only identifies corrosion as an aging mechanism and identifies RG 1.127 (XI.S7) as the acceptable AMP. VCSNS has identified additional aging effects for this component type and material, and uses the Maintenance Rule Structures Program (**Appendix B.1.18**) and Underwater Inspection Program (SWIS and SWPH) (**Appendix B.1.23**) as the primary AMPs for acceptable aging management. VCSNS uses the RG 1.127 Service Water Pond Dam

Table 3.5-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
8 (cont.)						Inspection Program (Appendix B.1.21) inspections only for supplementary review.
9	Service Water Pumphouse (SWPH), Service Water Intake Structure (SWIS)	Concrete	Raw Water Earthen Backfill	Cracking (Settlement)	Maintenance Rule Structures Program, Underwater Inspection Program (SWIS & SWPH), Service Water Structures Survey Monitoring Program.	For Water-Control Structures, the GALL identifies RG 1.127 (XI.S7) as the acceptable AMP. VCSNS uses the Maintenance Rule Structures Program (Appendix B.1.18), Underwater Inspection Program (SWIS and SWPH) (Appendix B.1.23) and Service Water Structures Survey Monitoring Program (Appendix B.1.22) as the primary AMPs for acceptable aging

Table 3.5-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR STATION CONTAINMENT, OTHER STRUCTURES AND COMPONENT SUPPORTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
9 (cont.)						management. VCSNS uses the RG 1.127 Service Water Pond Dam Inspection Program (**Appendix B.1.21**) inspections only for supplementary review.
10	Service Water Pond Dams, North Berm	Earthen	External	Loss of Material (Erosion, seepage, piping); Cracking (Settlement)	Service Water Pond Dam Inspection Program (RG 1.127), Maintenance Rule Structures Program	For Water-Control Structures, the GALL identifies RG 1.127 (XI.S7) as the acceptable AMP. VCSNS uses the Maintenance Rule Structures Program (**Appendix B.1.18**) to supplement the RG 1.127 Service Water Pond Dam Inspection Program (**Appendix B.1.21**) inspections for acceptable aging management.

3.5.4 REFERENCES

3.5-1	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.
3.5-2	NUREG-1801, “Generic Aging Lessons Learned Report,” Volumes 1 and 2, NRC, April 2001.
3.5-3	NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” NRC, April 2001.

3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

3.6.1 ELECTRIC DESCRIPTION

VCSNS has performed an aging management review on the following electrical/I&C commodity groups:

- Non-EQ Insulated Cables
- Non-EQ Connectors
- Non-EQ Splices
- Non-EQ Electrical Penetration Assemblies
- Non-EQ Terminal Blocks
- High Voltage Electrical Switchyard Bus
- High Voltage Transmission Conductors and Connections
- High Voltage Insulators

3.6.1.1 Non-EQ Insulated Cables

Non-EQ insulated cables include power cables, control cables, and instrument cables. For VCSNS, these applications are defined to be at the following voltage levels:

- Low Voltage Cables: 480 VAC, 240/120 VAC, 125 VDC (and less)
- Medium Voltage Cables: 7.2 kV
- High Voltage Cables: Greater Than 7.2 kV (none in scope)

In order to facilitate the review of the cables at VCSNS, the cables are placed into two categories: (1) power cable and (2) I&C cable. The power cable category includes all 7.2 kV cables and the 480 VAC power cables. The I&C category includes the 480 VAC control cable, all 240/120 VAC cable, and all DC cables (125 VDC and less). Depending upon their application, cables utilized as switchboard wire are placed into one of these two categories, typically as I&C cable. It should be noted at this time that VCSNS purchased nearly all of its electric power cable, control cable, and instrument cable (with the exception of certain communication cables, cables ordered for specific non-safety applications, and special cables ordered subsequently for specific modifications) to 10CFR50.49 Harsh EQ standards.

The worst case cable insulation possible in application used in license renewal is polyethylene with a 60 year service limiting temperature of 131°F. The non-EQ insulated cables will be subject to an aging management program as described in **Table 3.6.1**.

3.6.1.2 Non-EQ Electrical Connectors

Cable connections are used to connect the cable conductors with other cables or with a variety of electrical devices (e.g., instruments, motors, etc.). The various types of insulated cable connections (or terminations) are identified in the Cable Aging Management Guideline (AMG) [Section 3.3.2 of **Reference 3.6-1**]. The Cable AMG describes the cable termination grouping as follows:

- Compression connectors
- Fusion connectors
- Plug-in / Multi-pin connectors

A variety of plant documents were reviewed to identify electrical connectors in use at VCSNS, including procurement records, plant drawings, EQ binders, and plant maintenance documents. This review provided strong reasonable assurance that all types of connectors have been identified and that the bounding materials for the connectors at VCSNS have also been identified. Connectors are included in the Non-EQ Insulated Cables and Connections Inspection Program.

3.6.1.3 Non-EQ Electrical Splices

Many of the splices at VCSNS are delineated in a calculation which identifies all safety-related (Class 1E) 7.2 kV and 480V splices in the plant. The BOM table for the electric cables lists common splice and tape materials ordered for VCSNS during plant construction. The identification of VCSNS splices included a review of EQ documentation, procurement records, and design basis documents. This review provided strong reasonable assurance that all splice types and materials applicable to VCSNS (which may be subject to aging management review) have been identified. Non-EQ splices are included in the Non-EQ Insulated Cables and Connections Inspection Program.

3.6.1.4 Non-EQ Electrical Penetration Assemblies

Electrical penetration assemblies are utilized to carry electrical circuits through the Reactor Building containment wall while maintaining pressure-tight integrity. They provide the electrical continuity of the circuit and the pressure boundary for containment integrity. The scope of the review in this report applies only to the electrical function of the penetration assemblies. The pressure-retaining function of the penetration assemblies is addressed in **Section 2.4** of this application for the Reactor Building.

All the electrical penetrations at VCSNS have been listed in the VCSNS EQ program, whether or not they carry Class 1E circuits. The non-Class 1E electrical penetrations are classified as category “B1, B2” components with respect to EQ (i.e., they must not fail and prevent the accomplishment of a NSR function) and are administratively included in the EQ program in order to credit the portion of the EQ testing which justifies the pressure-retaining function of the penetrations. VCSNS utilizes D.G. O'Brien electrical penetration for its non-Class 1E applications.

The D.G. O'Brien electrical penetration assemblies are subject to aging management review. This review provides for their identification and also for the listing of the organic materials found during the review. Because there are D.G. O'Brien electrical penetration assemblies that are part of the VCSNS EQ program and have been evaluated in detail for that purpose, there is reasonable assurance that all their organic materials have been identified and properly evaluated with respect to aging for the non-EQ installations. An additional review has shown Non-EQ Electrical Penetrations at VCSNS to be located in areas inside and outside of the RB which have less severe environments, that are clearly enveloped by material properties and aging testing and evaluation done through the manufacturer. The Non-EQ Electrical Penetrations at VCSNS are not included in the Non-EQ Insulated Cables and Connections Inspection Program. The evaluation of the Non-EQ Electrical Penetrations at VCSNS is further documented in **Table 3.6-2 Item 2**.

3.6.1.5 Non-EQ Terminal Blocks

A terminal block consists of an insulating base with fixed metallic points for landing wires (conductors) or for connecting terminal rings (lugs). Terminal blocks are typically installed in an enclosure such as a control board, MCC, motor, terminal box, or a panel.

A complete list of the specific terminal blocks used at VCSNS does not exist in one file or location; however, a review of the Bills of Material and other plant documents (EQ files, etc.) for general electrical equipment revealed that the following suppliers have terminal blocks in use at VCSNS: GE, Kulka, Marathon, States, and Weidmuller.

From the Cable AMG [**Reference 3.6-1**], the most common materials used in the insulating base are phenolic, melamine resin, and nylon. The material with the least thermal and radiation resistance is nylon. Because there is no single document that lists all terminal block manufacturers, materials, and locations for VCSNS, nylon was chosen as the bounding material for the evaluation of the terminal blocks, due to its limited radiation resistance. By choosing nylon as the limiting material with respect to the plant environmental conditions, there is reasonable assurance that the terminal blocks at VCSNS are properly evaluated with respect to aging. Non-EQ terminal blocks are included, as appropriate, in the Non-EQ Insulated Cables and Connections Inspection Program.

3.6.1.6 High Voltage Electrical Switchyard Bus

High Voltage (HV) electrical switchyard bus is uninsulated, unenclosed, rigid electrical conductor used in switchyards and switching stations to connect two or more elements of an electrical power circuit such as active disconnect switches and passive transmission conductors. The review of switchyard bus included the bus itself as well as the hardware used to secure the bus to high-voltage insulators.The in scope switchyard bus at VCSNS is constructed of aluminum tubing or copper rods, and supported on station post insulators with aluminum cast fastening hardware.

For the ambient environmental conditions at VCSNS, no significant aging effects have been identified that would cause a loss of function for the extended period of operation. The potential effects of surface oxidation and vibration are not considered significant for the VCSNS installation. No aging management program for HV electrical switchyard bus is required.

3.6.1.7 High Voltage Transmission Conductors and Connections

Transmission conductors are uninsulated, stranded electrical cables used in switchyards, switching stations and transmission lines to connect two or more elements of an electrical power circuit such as active disconnect switches, power circuit breakers and transformers to passive switchyard bus. The review of transmission conductors included the transmission conductors and the hardware used to secure the conductors to a high-voltage insulator or to switchyard bus. Transmission conductors are supported by passive high-voltage strain or suspension insulators. Transmission conductors and connection hardware at VCSNS are made of aluminum reinforced with galvanized steel.

For the ambient environmental conditions at VCSNS, no significant aging effects related to conductor corrosion or wind loading vibration or sway on connections have been identified that would cause a loss of function for the extended period of operation. No aging management program for HV transmission conductors and connections is required.

3.6.1.8 High Voltage Insulators

HV switchyard post insulators and strain or suspension insulators as typically used on transmission towers are insulating materials in a form designed to (a) support a conductor physically and (b) separate the conductor electrically from another conductor or object. The insulators evaluated for license renewal are those used to support and insulate high voltage electrical components in switchyards, switching stations and transmissions such as transmission conductors and switchyard bus. HV insulators serve as an intermediate support between a supporting structure (such as a transmission tower or support pedestal) and the switchyard bus or transmission conductor. Materials of construction include porcelain, metal (insulator cap and pin) and cement to join the cap or pins to the porcelain.

For the ambient environmental conditions at VCSNS, no significant aging effects related to airborne contaminants or mechanical wear have been identified that would cause a loss of function for the extended period of operation. No aging management program for HV insulators is required.

3.6.2 AGING MANAGEMENT REVIEW

3.6.2.1 Methodology

The AMR methodology for the electrical discipline for VCSNS is summarized in the following points:

- Evaluation of the electrical component commodity groups (subject to AMR) to identify the organic materials subject to age-related degradation
- Identification and evaluation of the 60-year service-limiting environmental parameters for these organic materials
- Identification and evaluation of the aging mechanisms and effects to determine which require review
- Identification and evaluation of the service conditions (i.e., the operating environments and locations) for the electrical component commodity groups
- Evaluation of the industry and plant-specific operating experience for the electrical component commodity groups
- Aging management program evaluation (following NUREG-1801)
- Demonstration of aging management

The review of the VCSNS electrical component commodity groups with respect to aging mechanisms and effects was performed based upon the guidance of various industry documents, primarily SAND96-0344, *Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations* [**Reference 3.6-1**]. This document provides detailed materials analysis for cable and termination materials exposed to nuclear power plant environments. It also provides guidance for performing aging management reviews pursuant to 10 CFR 54.

The methodology used for the aging management review of the electrical commodity groups employs the "Plant Spaces" approach in which the plant is segregated into areas (or spaces) where common bounding environmental parameters can be assigned. The VCSNS plant operating environments are delineated as "Environmental Zones." Each bounding environmental zone is evaluated against the material of the commodity groups most susceptible to aging to determine if the components will be able to maintain their intended function through the period of extended operation. With respect to the electrical components, the environmental parameters of interest are temperature, radiation and moisture.

The intended functions of the electrical component commodity groups under review are as follows:

- To electrically connect or insulate two sections of an electrical circuit and/or to provide for continuity or insulation of electrical circuits.
- The electrical penetration assemblies also have a structural function to provide a leak-tight barrier for containment isolation; this is evaluated in **Section 2.4.1.3** of this application.

3.6.2.2 Operating Experience

Site: VCSNS site-specific operating experience was reviewed. The site-specific operating experience included a review of (1) Corrective Action Program, (2) Licensee Event Reports, (3) interviews with Systems Engineers. No additional aging effects requiring management were identified beyond those identified using the methods described in the previous Section.

Industry: An evaluation of industry operating experience published since the effective date of NUREG-1801 was performed to identify any additional aging effects requiring management. No additional aging effects requiring management were identified beyond those identified using the methods described in Section 3.6.2.1.

On-Going: On-going review of plant-specific and industry operating experience is performed in accordance with the plant Operating Experience Program.

3.6.3 AGING MANAGEMENT PROGRAM

3.6.3.1 Aging Management Programs Evaluated In NUREG-1801 That Are Relied On For License Renewal

Table 3.6.1 shows the aging management groups (combinations of components, materials and aging effects) and the aging management programs evaluated in NUREG-1801 that are relied on for license renewal of the electrical and instrumentation and control components at VCSNS. Note that this table only includes those components, materials and aging effects that are applicable to a PWR. The VCSNS comparison to NUREG-1801 (including clarifications and exceptions) is included in the “Discussion” column.

3.6.3.2 Further Evaluation Of Aging Management As Recommended By NUREG-1801

Further evaluation of aging management as recommended by the NUREG-1801 has been incorporated into the “Discussion” column of Table 3.6-1.

Table 3.6-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
1	Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Degradation due to various aging mechanisms	Environmental Qualification of Electrical equipment	Yes, TLAA	The TLAA for electrical equipment in the EQ Program is discussed in **Section 4.4** of this application.
2	Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure caused by	Aging management program for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	No	VCSNS applies the Non-EQ Insulated Cables and Connections Inspection Program (**Appendix B.2.9**) to these cables and connections even though the environmental conditions within the plant are not severe enough to show that the aging effects associated with elevated temperature are significant for the cable and connector/termination insulation materials.

Table 3.6-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
2 (cont.)		thermal/thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet [UV] sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion			
3	Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR;	Aging management program for Electrical Cables and Connections not subject to 10 CFR 50.49 EQ requirements	No	VCSNS applies the Non-EQ Insulated Cables and Connections Inspection Program (**Appendix B.2.9**) to these cables and connections even though the environmental conditions within the plant are not severe enough to show that the aging effects associated with elevated temperature are significant for the

Table 3.6-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3 (cont.)	requirements that are sensitive to reduction in conductor insulation resistance	electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion			insulation materials. Additional information is provided in **Table 3.6-2 Item 1**.
4	Inaccessible medium-voltage (2kV to 15kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR	Formation of water trees; localized damage leading to electrical failure (breakdown of insulation) caused by moisture intrusion and water trees	Aging management program for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No	The aging management review for medium voltage cables exposed to moisture and voltage stressors concluded that aging management at VCSNS is not required. No instances of power cable failure at VCSNS due to moisture intrusion were found.

**Table 3.6-1:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS
EVALUATED IN NUREG-1801 THAT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Group	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
4 (cont.)	50.49 EQ requirements				
5	Electrical connectors not subject to 10 CFR 50.49 EQ requirements that are exposed to borated water leakage	Corrosion of connector contact surfaces caused by intrusion of borated water	Boric acid corrosion	No	With regard to NUREG-1801 Chapter VI item A.2.1, the aging management for electrical connectors which may experience boric acid corrosion will be addressed by the Boric Acid Corrosion Surveillance (**Appendix B.1.2**) program at VCSNS. A separate electrical program will not be developed. The procedures which comprise the VCSNS Boric Acid Corrosion Surveillance program are sufficient to address the NUREG-1801 standard.

3.6.3.3 Aging Management Evaluations That Are Different From Or Not Addressed In NUREG-1801

Table 3.6-2 contains electrical and instrument and control components aging management review results that are not specifically addressed in NUREG-1801. This table includes component types, materials, environments and aging effects requiring management, along with the programs and activities for managing aging.

Table 3.6-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
1	Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance	Various organic insulating materials.	Maximum ambient temperature potential is 131°F Maximum 60 year gamma radiation potential is 1.95E10 Rads	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/thermoxidative degradation of organics; radiation induced oxidation; moisture intrusion.	Non-EQ Insulated Cables and Connections Inspection Program	NUREG-1801 recommends an aging management program specifically for cables with sensitive, low-level signals. VCSNS applies the Non-EQ Insulated Cables and Connections Inspection Program (**Appendix B.2.9**) (**Table 3.6-1 Item 2**). The visual inspection of instrument as well as power and control cables is considered a better means to identify age-related degradation due to localized ambient thermally and radiologically induced stress prior to significant

**Table 3.6-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS
THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL**

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
1 (cont.)						loss of insulation resistance.
2	Non-EQ Electrical Penetration Assemblies	Polysulfone insulation, Silicon Rubber o-rings, Fiberglass epoxy, Neoprene gaskets, Molycote thread sealant, Polycarbonate view port.	Maximum ambient temperature is 121°F for a short time in Env. Zones PAA-03 and PAI-02. Maximum 60 year gamma radiation is 3.0E06 Rads in Env. Zone RB-05.	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance; electrical failure caused by thermal/Thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet sensitive materials only) of	None required.	Non-EQ Electrical Penetrations at VCSNS are located in areas inside and outside of the RB in environments which are clearly enveloped by material properties and aging testing and evaluation done through the manufacturer. Cable that utilize these penetrations are addressed in Table 3.6-1 Item 2.

Table 3.6-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
2 (cont.)				organics; radiation-induced oxidation; moisture intrusion		
3	Electrical Switchyard Bus	Aluminum, copper	Outdoor environment in air and rainwater	Change in material properties leading to increased resistance and heating due to oxidation. Also cracking due to vibration.	None	For the ambient environmental conditions at VCSNS, no aging effects have been identified that could cause a loss of function. No aging management is required.
4	High Voltage Transmission Conductors and	Aluminum, Steel	Outdoor environment in air and rainwater	Loss of conductor strength due to corrosion. Also wear or fatigue	None	For the ambient environmental condition at VCSNS, no significant aging effects related to corrosion or wind

Table 3.6-2:
SUMMARY OF AGING MANAGEMENT PROGRAMS FOR ELECTRICAL AND INSTRUMENTATION AND CONTROL COMPONENTS THAT ARE DIFFERENT FROM OR NOT ADDRESSED IN NUREG-1801 BUT ARE RELIED ON FOR LICENSE RENEWAL

AMR Item	Component Type	Material	Environment	Aging Effect / Mechanism	Program Activity	Discussion
4 (cont.)	Connections	Aluminum, Steel	Outdoor environment in air and rainwater	due to wind loading vibration or sway.		loading vibration or sway for conductors or connections have been identified that could cause a loss of function. No aging management program is required.
5	High Voltage Insulators	Porcelain, metal cap or pin, cement	Outdoor environment in air and rainwater	Surface contamination or cracking due to airborne contaminants. Also loss of material due to mechanical wear.	None	For the ambient environmental conditions at VCSNS, no significant aging effects related to airborne contaminants or mechanical wear have been identified that could cause a loss of function. No aging management program is required.

3.6.4 REFERENCES

3.6-1	U. S. Department of Energy Report SAND 96-0344, Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations, September 1996.

SECTION 4 - TIME-LIMITING AGING ANALYSIS

4.0 TIME-LIMITED AGING ANALYSIS

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSIS

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 as those licensee calculations and analyses that meet six specific criteria. 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses be provided as part of an application for a renewed license.

10 CFR 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.

10 CFR 54.21 (c)

(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.

4.1.1 TLAA PROCESS OVERVIEW

The process used to identify the time-limited aging analyses for Virgil C. Summer Nuclear Station (VCSNS) is consistent with the guidance provided in NEI 95-10 [**Reference 4.8.5.1**]. Calculations and analyses that meet the six criteria of 10 CFR 54.3 were identified by searching the current licensing basis, which includes the FSAR, Technical Specifications, engineering calculations, technical reports, docketed licensing correspondence, and applicable Westinghouse Topical Reports (WCAPs).

4.1.2 EXEMPTIONS TO 10 CFR 50.12

10 CFR 54.21(c) also requires that an application for a renewed license include a list of current plant-specific exemptions granted pursuant to 10 CFR 50.12 that are based on time-limited aging analyses as defined in 10 CFR 54.3.

Each exemption granted to VCSNS was evaluated. The evaluation established whether the exemption is still in effect, and if so, whether the exemption is based on a time-limited aging analysis. No current 10 CFR 50.12 exemptions based on a time-limited aging analysis as defined in 10 CFR 54.3 have been identified for VCSNS.

4.1.3 TLLA SUMMARY

The VCSNS calculations and evaluations that met all six of the criteria listed in 10 CFR 54.3 are listed in Table 4.1-1 and discussed in Sections 4.2 through 4.7 of this Application. Selected TLAAs that did not meet all six criteria for VCSNS are also listed and discussed based on their inclusion in NUREG-1801 [**Reference 4.8.4.2**] or NUREG-1800 [**Reference 4.8.4.1**].

Three options are provided in 10 CFR 54.21(c)(1) for the applicant to address each plant specific time-limited aging analyses identified. For each TLAA identified, these regulations direct that the applicant must 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.

Table 4.1-1:
VCSNS SPECIFIC - ACTUAL TLAA SUMMARY

GROUP	DESCRIPTION	SECTION	TLAA OPTION
Reactor Vessel Neutron Embrittlement	Upper Shelf Energy	4.2.1	Option ii
	Pressurized Thermal Shock	4.2.2	Option ii
	Pressure-Temperature (P-T) Limits	4.2.3	Option ii
Metal Fatigue	ASME Section III, Class I	4.3.1	Option iii
	ASME Section III, Class 2 and 3 Piping Fatigue	4.3.2	Option i
EQ	Environmental Qualification	4.4	Option iii
Concrete Containment Tendons	Concrete Containment (Reactor Building) Tendon Prestress Analysis	4.5	Option iii
Containment Liner Plate, Metal Containments, And Penetration Fatigue Analysis	Containment (Reactor Building) Liner	4.6.1	Option ii
	Metal Containments	4.6.2	Not a TLAA
	Containment (Reactor Building) Isolation Bellows	4.6.3.1	Not a TLAA
	Containment (Reactor Building) Isolation - Fracture Toughness And Effects Radiation	4.6.3.2	Not a TLAA
Other	Reactor Coolant Pump Flywheel	4.7.1	Option i
	Leak-Before-Break Analyses	4.7.2	Option ii
	Crane Load Cycle Limit	4.7.3	Option i
	Service Water Intake Structure Settlement	4.7.4	Option ii

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The regulations governing reactor vessel integrity are contained in the following sections of 10 CFR Part 50:

10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation," 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 50.60.

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," contains fracture toughness requirements for protection against pressurized thermal shock.

4.2.1 UPPER-SHELF ENERGY

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires that reactor vessel beltline materials must have an initial, pre-irradiation, Charpy Upper Shelf Energy of no less than 75 ft-lb and must maintain a Charpy Upper Shelf Energy 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 Upper Shelf Energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code) [**Reference 4.8.5.2**].

The VCSNS-calculated beltline fluence is one of the factors used in the determination of the decrease in Charpy Upper Shelf Energy due to radiation embrittlement and thermal aging of the reactor vessel. Since upper shelf energy is a measure of the fracture toughness of a material, a decrease in the upper shelf energy of reactor vessel materials implies a reduction in fracture toughness of the vessel.

Charpy Upper Shelf Energy values are calculated for beltline region materials using NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [**Reference 4.8.4.3**]. In response to NRC Generic Letter 92-01, "Reactor Vessel Structural Integrity," Revision 1, and based upon examinations of the first three VCSNS surveillance capsules, VCSNS reported [**References 4.8.2.1** and **4.8.4.4**] the end of current license (32 EFPY) upper-shelf energy and limiting beltline component to be 67.5 ft-lb for intermediate plate A9154-1 (transverse-orientation).

Revision to the analyses is required to calculate Charpy Upper-Shelf Energy for the end of the extended operating period. The two remaining VCSNS reactor surveillance capsules require additional exposure to neutron fluence in order to provide data that correlates to estimated fluence on the vessel at the end of the period of extended operation. Following adequate capsule exposure, a capsule will be withdrawn and analyzed. The Charpy Upper Shelf Energy will be recalculated for additional fast neutron fluence corresponding to the end of the extended operating period. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to calculate the RPV Charpy Upper Shelf Energy for the end of the extended operating period.

4.2.2 PRESSURIZED THERMAL SHOCK

Beltline fluence is one of the factors used to determine the margin for pressurized thermal shock due to radiation embrittlement and thermal aging of the reactor vessel. The margin is the difference between the maximum nil ductility reference temperature in the limiting beltline component (RT_{PTS}) and the screening criterion established in accordance with 10 CFR 50.61. The method for calculating RT_{PTS} values is consistent with NRC Regulatory Guide 1.99, [**Reference 4.8.4.3**].

$$RT_{PTS} = RT_{NDT}\ (\text{Unirradiated}) + M + [(CF) * (FF)]$$

Where the product of the Chemistry Factor (CF) and Fluence Factor (FF) is alternatively called ΔRT_{PTS}. RT_{NDT} is the reference temperature for a reactor vessel material in the preservice or unirradiated condition. M is the margin added to account for the uncertainties in the value of RT_{NDT}.

The Chemistry Factor accounts for the effects of copper and nickel content on radiation embrittlement. If the specific material composition of the vessel beltline materials is available some conservatism may be eliminated from the RT_{PTS} determination. The calculated RT_{PTS} is refined in conjunction with analysis of each successive surveillance capsule.

The RT_{PTS} values for VCSNS were calculated for the current 40-year operating term. Per WCAP-15103, all of the beltline materials in the VCSNS reactor vessel have end-of-life (EOL) 32 EFPY and life extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings, and longitudinal welds and 300°F for circumferential welds. These results show that VCSNS will not exceed the RT_{PTS} screening criteria during the current (32 EFPY) or extended (48 EFPY) operating license. The RT_{PTS} value will be recalculated when one of the two remaining VCSNS surveillance capsules is removed from the vessel. VCSNS intends to remove at least one of the surveillance capsules when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall at the end of the extended operating period. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to calculate the RT_{PTS} data for the end of the extended operating period.

4.2.3 PRESSURE-TEMPERATURE (P-T) LIMITS

Beltline fluence is one of the factors used to calculate revisions to pressure-temperature limits for heatup and cooldown due to radiation embrittlement of the reactor vessel. This input is based on calculation of an Adjusted Reference Temperature (ART) using methodology of NRC Regulatory Guide 1.99, [**Reference 4.8.4.3**]. This methodology is very similar to that used to calculate RT_{PTS}. However, the calculation of ART also considers attenuation of the fast neutron fluence through the vessel wall to the depth of the postulated flaw.

The formula for calculating ART is listed below.

$$ART = IRT_{NDT} + \Delta RT_{NDT} + M$$

Where ΔRT_{NDT} is the product of the Chemistry Factor (CF) and attenuation-corrected fast neutron fluence factor. RT_{NDT} is the reference temperature for a reactor vessel material in the pre-service or unirradiated condition. M is the margin added to account for the uncertainties in the value of RT_{NDT}. IRT_{NDT} is the initial RT_{NDT}, the reference temperature for the unirradiated material as defined in paragraph NB-2332 of ASME Section III.

The Chemistry Factor accounts for the effects of copper and nickel content on radiation embrittlement. If the specific material composition data is available some conservatism may be eliminated from the ART. The calculated ART is refined in conjunction with analysis of each successive surveillance capsule. Allowable pressure-temperature curves are generated for steady state and each finite cooldown rate specified, assuming a reference flaw at the inside (most limiting) surface of the reactor vessel. A composite cooldown limit curve is constructed as the minimum of each of these curves. Similarly, allowable pressure-temperature curves are generated for steady state and each finite heatup rate specified considering each of the worst-case reference flaw locations (outside vessel surface and inside vessel surface).

The current VCSNS heatup and cooldown curves are based on calculations for the current 40-year operating term. VCSNS will revise the calculated value of ART and associated pressure-temperature limits for heatup and cooldown when one of the two remaining VCSNS surveillance capsules is removed from the vessel. VCSNS intends to remove at least one of the surveillance capsules when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall at the end of the extended operating period. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to develop pressure-temperature limits for the period of extended operation.

4.3 METAL FATIGUE

4.3.1 ASME SECTION III, CLASS 1

The issue of thermal fatigue for ASME Section III Class 1 [**Reference 4.8.5.4**] components has been identified as a time-limited aging analysis for VCSNS. All six of the criteria in 10 CFR 54.3 are satisfied. Class 1 components have been designed with the transient cycle assumptions in Table 5.2-2 of the Final Safety Analysis Report [**Reference 4.8.3.2**].

Currently, there are no components in the scope of the VCSNS Inservice Inspection (ISI) Program that contain flaws which exceed acceptance standards and require analysis to demonstrate acceptance.

The VCSNS ISI Program involves the monitoring of thermal transients. The Thermal Fatigue Management Program described in **Appendix B.3.2** of this application is equivalent to the corresponding program described and evaluated in Section X.M1 of NUREG-1801 [**Reference 4.8.4.2**]. However, enhancements to the program are warranted to incorporate the new guidance in EPRI Report MRP-47 concerning “Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application,” which was published in October 2001[**Reference 4.8.5.3**].

GSI-190, Fatigue Evaluation of Metal Components for 60-year Plant life, relates to environmental effects on fatigue of reactor coolant system components for 60 years and was closed by the NRC [**Reference 4.8.2.6**]. In the closure letter, the NRC concluded that licensees should address the effects of the reactor coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The VCSNS Thermal Fatigue Management Program will be revised by the end of the current license term (40 years) to base future projections on 60 years of operation and to account for environmental effects of the reactor coolant environment on RCS components. The Thermal Fatigue Management Program as revised will meet the corresponding program described in NUREG-1801 (GALL) [**Reference 4.8.4.2**], Section X.M.1. This program meets the requirement of 10 CFR 54.21(c)(1) by the utilization of option (iii).

4.3.2 ASME SECTION III, CLASS 2 AND 3 PIPING FATIGUE

Piping systems, designed in accordance with ASME Section III, Class 2 and Class 3 [**Reference 4.8.5.4**] or ANSI B31.1 [**Reference 4.8.5.5**], utilize allowable stress values based on a stress reduction factor. VCSNS **FSAR Table 3.2-3** states the versions of ASME Section III and ANSI B 31.1 that apply to different plant components. Table 4.3-1 of NUREG-1800 [**Reference 4.8.4.1**] includes Class 2 and 3 piping fatigue. The stress range reduction factors range from 1 (no reduction) for 7000 cycles or less, to a stress range reduction factor of 0.5 for 100,000 cycles or more.

For all of the systems reviewed, except the Post-Accident Sampling and Nuclear Sampling Systems, the number of thermal cycles is related to the heatup and cooldown of the plant (steam and primary), which ideally, occurs once a cycle (18 months). Even if this is conservatively assumed to occur once a month for 60 years, then the total thermal cycles would only be 720, which is approximately one–tenth of the allowed 7000 cycles. Therefore, it is conservative to assume that Class 2 and 3 and ANSI B31.1 systems are adequately evaluated for fatigue for the period of extended operation.

The estimated number of thermal cycles at 60 years on the Post-Accident Sampling and Nuclear Sampling Systems system is 6668 cycles which is less than the 7000 allowed thermal cycles by a small margin. This magnitude of the estimate is due to past operational practices of obtaining frequent samples by use of the “B” RCS Loop sampling line. In order to ensure that the number of cycles remained below 7000, procedural controls will be implemented to ensure that sampling activities are discontinued on the “B” RCS Loop sampling line for all but emergency situations and/or revise the calculations to verify the acceptability of a higher safety factor.

Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (i) for this area because piping fatigue for ASME Section III, Class 2 and 3, and ANSI B31.1 is adequately analyzed for the period of extended operation.

4.4 ENVIRONMENTAL QUALIFICATION (EQ)

The environmental qualification analyses for electrical equipment included in the Environmental Qualification (EQ) Program were identified as potential time-limited aging analyses for VCSNS. Since the EQ analyses meet all six criteria for a TLAA specified in 10 CFR 54.3, EQ is considered a plant specific TLAA at VCSNS.

The EQ Program is based on the requirements of 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.” As part of the EQ evaluation, the electrical equipment in the program is given a quantified value for its service life in a given environment (called the 'qualified' life). This qualified life is often expressed in terms of being greater than 40 years, the original duration of the plant's license. Components in the EQ Program that have calculated lifetimes equal to or greater than 40 years are identified as time-limited aging analyses, and require re-analysis to meet the duration of the new plant operating license (60 years from initial plant start-up).

The EQ documents at VCSNS that are subject to TLAA review include the EQ binders, which are comprehensive reports prepared to support the qualification of each unique type of electrical component for the plant environments in which they are located. Each binder contains or references a vendor test report or an analysis that justifies the qualification of the equipment. Each binder contains or references either a calculation of qualified life or an evaluation to justify a qualified life. Additional documents in the EQ program subject to TLAA review include plant calculations, vendor reports, and the EQ and Regulatory Guide 1.97 design basis document. Many environmental qualification calculations of electrical equipment are identified as time-limited aging analyses for VCSNS. These calculations are considered the technical rationale that the current licensing basis will be maintained during the period of extended operation.

NRC guidance for addressing GSI-168 for license renewal is contained in the June 2, 1998, NRC letter to NEI [**Reference 4.8.2.5**]. In this letter, the NRC states, “With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the Statements of Consideration is to provide a technical rationale demonstrating that the current licensing basis for environmental qualification pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the Statements of Consideration also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time.” Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a renewal application at this time.

The EQ TLAAs for VCSNS are listed in Table 4.4 -1. These documents were selected from the overall list of EQ binders and a review of other associated EQ documents at VCSNS. The EQ binders identified as TLAAs include all components with a qualified life of equal to or greater than 40 years. VCSNS has elected to utilize 10 CFR 54.21(c)(1) - option (iii) to demonstrate that the Environmental Qualification Program will continue to adequately manage the effects of aging of the electrical components for the period of extended operations. Components with qualified lives less than 40 years are not considered to have TLAAs and will not be evaluated further.

Table 4.4-1:
ENVIRONMENTAL QUALIFICATION DOCUMENTS
"HARSH" EQ BINDERS

Document No.	Title	Manufacturer
EQDP-H-CA0-R11-1	Firezone R Power, Control, and Instrument Cable	Rockbestos
EQDP-H-CA1-K03-1	600 Volt Control Cable	Kerite
EQDP-H-CA1-K03-2	Power and Control Cable	Kerite
EQDP-H-CA1-O01	600 Volt Control Cable (Armored)	Okonite
EQDP-H-CA1-R11-1	600 Volt Control Cable	Rockbestos
EQDP-H-CA2-K03	600 Volt Power Cable	Kerite
EQDP-H-CA2-O01	8 kV Power Cable	Okonite
EQDP-H-CA4-B20	Coaxial Twinax Instrument Cable	Brand-Rex
EQDP-H-CA4-O01	Special Instrument Cable	Okonite
EQDP-H-CA4-R05	Switchboard Wire	Raychem
EQDP-H-CA4-R11-1	Switchboard Wire	Rockbestos
EQDP-H-CA4-R11-2	Instrumentation Cable	Rockbestos
EQDP-H-CA4-S02	Instrument Cable	Samuel Moore (Dekoron)
EQDP-H-CA6-C06	CETS and HRCM MI Cable System / Connectors	CE/ERD
EQDP-H-CA6-K03	Thermocouple Extension Cable	Kerite
EQDP-H-CA6-O01	Heat Tracing Thermocouple Cable	Okonite
EQDP-H-CA7-O01	Splice Materials	Okonite
EQDP-H-CA7-R05-1	Raychem Splicing Products	Raychem
EQDP-H-CA7-R05-2	Motor Connection Kits	Raychem
EQDP-H-CA7-R05-3	Nuclear High Voltage Terminations	Raychem
EQDP-H-C01-A05	Electrical Cable Terminals	AMP

Table 4.4-1:
ENVIRONMENTAL QUALIFICATION DOCUMENTS
"HARSH" EQ BINDERS

Document No.	Title	Manufacturer
EQDP-H-C02-S04	Terminal Blocks	States
EQDP-H-CO5-C08	Electrical Penetration Assemblies	Conax Buffalo
EQDP-H-CO5-D01-1	Triax Connectors, M06 Modules	D.G. O'Brien
EQDP-H-CO5-D01-2	Penetration Modules, Plugs, and Hermetic Connectors	D.G. O'Brien
EQDP-H-CO6-C08-1	Nuclear Service Connectors	Conax Buffalo
EQDP-H-CO6-E03-1	Quick Disconnect Connectors	EGS
EQDP-H-CO6-E03-2	Grayboot Connector	EGS
EQDP-H-CO6-L02	Quick Disconnect Multipin T/C Connectors and Socket Plugboard	Litton/Veam
EQDP-H-CO6-N01	Hermetic Connectors and DC734 RTV Thread Sealant	Namco
EQDP-H-HW5-E03-1	Conduit Seal	EGS
EQDP-H-IN1-B05-1	Transmitters Model 764	Barton
EQDP-H-IN1-B05-2	Transmitters Model 763, 763A	Barton
EQDP-H-IN1-B05-3	Transmitters Model 763	Barton
EQDP-H-IN1-B05-4	Transmitters Model 764	Barton
EQDP-H-IN5-P03	Resistance Temperature Detector	Pyco
EQDP-H-IN6-G02-1	Neutron Detector Assembly	Gamma-Metrics
EQDP-H-IN6-G02-2	Neutron Flux Mon. System Amplifier Panel	Gamma-Metrics
EQDP-H-IN6-V05	HRCM, Cable Assembly Hermetic Connector	Victoreen/Hermetic Seal Corp
EQDP-H-IN7-T03	Valve Flow Monitoring System, Transient Shield, Sensor and Charge Converter	Technology for Energy/Endevco

Table 4.4-1:
ENVIRONMENTAL QUALIFICATION DOCUMENTS
"HARSH" EQ BINDERS

Document No.	Title	Manufacturer
EQDP-H-MO1-G03	EFW And RBS Pump Motors	GE
EQDP-H-MO1-R07-1	Class 1E Continuous Duty AC Motors	Reliance Electric
EQDP-H-MO1-W01	CCW Pump Motor	Westinghouse
EQDP-H-MO1-W10A and B	RHR & Charging Pump Motors	Westinghouse
EQDP-H-MO1-R07	Fan Motors	Reliance Electric
EQDP-H-MO2-R07	Fan Motors	Reliance
EQDP-H-SE2-E03-1	Grafoil Paste Thread Sealant	EGS
EQDP-H-SW1-T01	Transfer Switch	Terrac – Gould
EQDP-H-SW4-N01-1	Limit Switches EA-180	Namco
EQDP-H-SW4-N01-2	Limit Switches EA-740	Namco
EQDP-H-SW5-S07-1	Temperature Switch	Static-O-Ring
EQDP-H-VO4-A11	Solenoid Valves	ASCO
EQDP-H-VO4-A13	Solenoid Valves	Allied
EQDP-H-VO4-C16-1	Solenoid Valves	Chicago Fluid Power
EQDP-H-VO4-T02-1	Solenoid Valves	Target Rock Model 83GG-001
EQDP-H-VO4-T02-2	Solenoid Valves	Target Rock Model 97A-001
EQDP-H-VO4-V01	Solenoid Valves	Valcor
EQDP-H-VO5-L01	Motor-Operated Valve Actuators	Limitorque
EQDP-H-VO5-R17	Motor-Operated Valve Operators	Rotork
EQDP-H-WR3-W01	Electric Hydrogen Recombiner	Westinghouse

4.4.1 TLAA EVALUATION

The VCSNS Environmental Qualification Program ensures that the effects of aging will be adequately managed for the period of extended operation. The VCSNS EQ Program is evaluated with respect to the program attributes described in NUREG-1801 [**Reference 4.8.4.2**] in Appendix B of this application. Plant documents that address the qualified lives of the EQ components will be revised to reflect the period of extended operation.

The EQ Program at VCSNS implements the requirements of 10 CFR 50.49 and is an aging management program for license renewal. Re-analysis of an existing aging evaluation, to extend the qualification of electrical components, is performed as part of the EQ Program. These calculations are performed in response to adjusted plant environments, new vendor data, and other input which might impact the conclusions of the EQ binders.

The important attributes of re-analysis include analytical methods, data collection and reduction methods, the underlying assumptions, the acceptance criteria, and any corrective actions to components as a result of prior evaluation. These re-analysis attributes are discussed below.

4.4.1.1 Analytical EQ Re-analysis Methods

The analytical methods used in the re-analysis of an aging evaluation are the same as those applied in the prior analysis. The Arrhenius methodology is an acceptable thermal model for reaction rate and thermal aging evaluation. The analytical method used for radiation aging evaluations is to demonstrate qualification for the total integrated dose via test. The total integrated dose is the normal operational environmental dose plus the projected accident dose. For license renewal, it is acceptable to establish a 60-year normal operational dose by taking the 40-year dose previously established and multiplying this value by 1.5. The result is added to the postulated accident dose to obtain a 60-year total integrated dose. For cyclical aging evaluations, a similar methodology is acceptable. Other methods may be justified on a case-by-case basis.

4.4.1.2 Data Collection And Reduction Methods For EQ

Reducing excess conservatism in electrical component service conditions (i.e., temperature, radiation, number of cycles) used in prior evaluations is the chief method used for re-analysis. Temperature data used in an aging analysis is typically conservative and is based on plant design temperatures. Actual plant operating temperature data is typically less than design values. Actual plant operating data may be obtained from temperature monitors specifically installed for EQ measurements, data taken by operators during rounds, or other temperature monitors in place in the plant.

4.4.1.3 Evaluation Of Underlying EQ Assumptions

EQ component aging evaluations typically contain sufficient conservatism to account for most environmental changes, which occur as a result of plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities, the affected EQ component(s) is (are) evaluated and appropriate corrective actions are taken, which may include revisions to the qualification bases and conclusions.

The re-analysis of an aging evaluation could extend the qualification of a subject component. If the qualification cannot be extended by re-analysis, the component is to be refurbished, replaced or re-qualified prior to the expiration of the current qualification. The timing of the re-analysis must permit sufficient time to refurbish, replace, or re-qualify the component if the re-analysis effort is unsuccessful.

4.4.2 EQ PROGRAM REVIEW

The VCSNS EQ Program, as described in Appendix B of this application, meets the criteria of an acceptable aging management program, as defined in Section X.E1 of NUREG-1801, Volume II [**Reference 4.8.4.2**].

4.4.3 EQ TLAA EVALUATION CONCLUSIONS

The VCSNS Environmental Qualification Program ensures that the effects of aging will be adequately managed for the period of extended operation. The VCSNS EQ Program is evaluated with respect to the program attributes described in NUREG-1801 [**Reference 4.8.4.2**] in Appendix B of this License Renewal Application. Plant documents, which address the qualified lives of the EQ components, will be revised to reflect the period of extended operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (iii) to demonstrate that the Environmental Qualification Program will continue to adequately manage the effects of aging of electrical components for the period of extended operation.

4.5 CONCRETE CONTAINMENT (REACTOR BUILDING) TENDON PRESTRESS ANALYSIS

The VCSNS Reactor Building was prestressed in order to have low-strain linear response at design loads and thus assure integrity of the liner. The exterior wall is post-tensioned in both vertical and hoop directions. Hoop tendons are anchored on three (3) buttresses, each spaced 120° apart along the circumference of the containment wall. On the dome, a three-way post-tensioning system is employed.

General Design Criteria 53, “Provisions for Containment Testing and Inspection,” of Appendix A, requires that the reactor containment be designed to permit periodic inspection and surveillance. The guidelines to perform inservice inspections of the tendons (also referred to as “tendon surveillances”) are established by NRC Regulatory Guide 1.35 [**Reference 4.8.4.5**]. These inspections are required to be performed at 1, 3, and 5-year intervals, following the Structural Acceptance Test and every 5 years thereafter. During these inspections, pre-designated tendons are examined. The examinations include tendon force measurements, visual examination of anchorages and surrounding concrete, tendon wire tensile tests of a limited number of wires, and chemical tests of the corrosion protection medium (grease).

In March 1995, the NRC issued a new rule, 10 CFR 50.55a. This rule invoked new containment inspection requirements, including the requirements of ASME Code, Section XI, Subsections IWE and IWL, 1992 Edition and 1992 Addenda [**Reference 4.8.5.2**]. This rule also imposed new requirements for future tendon surveillances.

The VCSNS tendon surveillance program is based on proposed Revision 3 of Regulatory Guide 1.35 [**Reference 4.8.4.5**]. The Regulatory Guide remained in a proposed status until July 1990 when the finalized Revision 3 was issued. The NRC accepted the VCSNS tendon surveillance program based on the proposed Revision 3 of Regulatory Guide 1.35 [**Reference 4.8.2.3** and **4.8.2.4**].

VCSNS has performed all required tendon surveillances. The Fourth Period (10th year-1990) Tendon Surveillance was used to re-tension the vertical tendons, since the tendon force losses were projected to reach a level where the minimum required force could not be demonstrated by the next surveillance period. This surveillance also indicated the need for potential retensioning of the dome and hoop tendons by the year 2015. The Fifth Period (15th year - 1996) and Sixth Period (20th year - 2000) Tendon Surveillances were completed with acceptable results. Based on trending data and results from previous surveillances, VCSNS does not currently expect the tendons to provide adequate prestress for 60 years without future retensioning of various members.

Chapter X.S1, “Concrete Containment Tendon Prestress,” of NUREG-1801 [**Reference 4.8.4.2**], applies to those facilities that adopt 10 CFR 54.21(c)(1) - Option (iii) for containment tendon prestress. This option credits the Containment Tendon Program (or equivalent) with managing the effects of aging. NUREG-1801, Chapter XI.S2, “ASME Section XI, Subsection IWL” presents a generic Reactor Building Tendon Program.

Programmatic controls are used to ensure that the Reactor Building tendons are capable of performing their design function. Therefore, the Reactor Building tendons are a TLAA, and VCSNS will utilize 10 CFR 54.21(c)(1) - Option (iii) to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.6 CONTAINMENT (REACTOR BUILDING) LINER PLATE, METAL CONTAINMENTS, AND PENETRATION FATIGUE ANALYSIS

4.6.1 CONTAINMENT (REACTOR BUILDING) LINER

The Reactor Building liner provides an essentially leak tight membrane on the inside face of the prestressed concrete Reactor Building that can contain airborne radioactive particles and gases due to postulated accidents such as a LOCA. However, the liner is required to remain within certain strain limits associated with serviceability that are set by the ASME B&PV code for normal operation.

A VCSNS calculation considered Reactor Building liner fatigue for 40 years per ASME Section III, paragraph NE-3131, 1974 with all applicable addenda [**Reference 4.8.5.4**]. This calculation performed comparisons based on 40 years and concluded that the liner (both stainless base and carbon sidewalls) met the criteria of NB 3222.4 (d) for the suitability for cyclic condition and no fatigue analysis was required.

This calculation was revised and concluded that the design criteria for the Reactor Building liner are satisfied for 60 years. Thus VCSNS utilizes 10 CFR 54.21 (c) (1) option (ii) to demonstrate that the Reactor Building liner fatigue is adequately analyzed for the period of extended operation.

4.6.2 METAL CONTAINMENTS - N/A

4.6.3 CONTAINMENT (REACTOR BUILDING) ISOLATION

Time-Limited Aging Analyses issues related to containment isolation are Reactor Building isolation bellows fatigue, fracture toughness of the penetration materials and effects of radiation.

4.6.3.1 Containment (Reactor Building) Isolation Bellows

FSAR Section 6.2.6.2.1.1 [**Reference 4.8.3.2**], “Piping Penetrations and Spares,” states:

“Cold penetrations are sealed by a flat plate, welded to both the sleeve and the process pipe at each end of the penetration sleeve. Since no resilient or flexible seals are used, these penetrations do not require Type B leakage tests.

Hot penetrations are sealed on the inside of the containment by a flat plate in a manner similar to cold penetrations. On the outside, they are sealed by a single bellows, one end of which is attached to the penetration sleeve and the other to the process pipe. Since the containment barrier does not utilize a resilient or flexible seal, these penetrations do not require Type B leakage tests."

Therefore, the credited Reactor Building isolation barrier does not utilize any flexible seals between the process pipe and Reactor Building liner. The bellows are not included in this isolation barrier and have no testing or inspection requirements. The bellows were designed to form a barrier without limiting thermal expansion. The absence of a leak-tight bellows does not affect the thermal expansion.

The pipe penetration bellows do not meet either Criterion 4 or Criterion 5 of 10 CFR 54.3 for a plant specific TLAA because they are not credited in a safety determination and do not perform a containment isolation function. Since the bellows were not required for Reactor Building isolation, they are not within the scope of license renewal (Criterion 1 of 10 CFR 54.3 for a plant specific TLAA). The bellows were not determined to be relevant in making a safety determination since they have no Reactor Building isolation function. The bellows also fail to meet Criterion 5 of 10 CFR 54.3 for a TLAA in that they do not support the capability of the containment isolation system and associated components in performing their design function of Reactor Building isolation.

The fuel transfer tube bellows are not required for Reactor Building isolation. The refueling canal is sealed and tested as required for Reactor Building integrity. This seal involves installing a blind flange that seals the fuel transfer tube, which has a double-gasketed seal. The collar on the transfer tube that mates with the flange is drilled with passageways that allow pressurization between the gaskets. Therefore, the fuel transfer tube bellows are not utilized to provide part of the Reactor Building isolation barrier. The fuel transfer tube is included in the VCSNS 10 CFR 50, Appendix J, Type B Testing program.

The Reactor Building piping penetration bellows and fuel transfer canal bellows do not meet all six criteria for a plant specific TLAA provided in 10 CFR 54.3. Therefore, the isolation bellows and associated analyses do not qualify as a TLAA at VCSNS.

4.6.3.2 Containment (Reactor Building) Isolation - Fracture Toughness And Effects Radiation

The FSAR and the associated NRC Safety Evaluation Report, NUREG-0717, include statements involving the life of the materials used in the RB penetrations and the fracture toughness of the containment pressure boundary.

FSAR Section 6.2.4.5 [**Reference 4.8.3.2**], “Materials”, states that the containment isolation system materials were selected to perform their design function for 40 years based on an integrated radiation dose of 10^8 rads. Materials that contact Reactor Building spray are resistant to corrosion, and isolation system components, which contain zinc and aluminum, are kept to a minimum. No time-limited aging analysis was identified for these components. The statement that components would perform their design function for 40 years was unsupported in that no analyses were identified that specifically calculated the design life.

NUREG-0717 [**Reference 4.8.3.1**] states: “ferritic materials of the containment pressure boundary which were considered in our (NRC) assessment are those which have been used in the fabrication of the equipment hatch, personnel air lock, penetrations and fluid system components, including the valves required to isolate the system.” Taken in the context of NUREG-0717, it is clear that the NRC has determined that the ferritic materials used in the construction of the VCSNS containment meet the appropriate requirements of the ASME Code, comply with GDC-51, and behave in a non-brittle manner. These statements do not present a time-limited aging analysis.

No time-limited aging analyses or calculations were identified that require revision for the effects of radiation on containment isolation components.

4.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

4.7.1 REACTOR COOLANT PUMP FLYWHEEL

The reactor coolant pump (RCP) motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down 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 keyway from stresses due to starting the motor. Therefore, this topic is considered to be a time-limited aging analysis for license renewal.

WCAP-14535A, “Topical Report of Reactor Coolant Pump Flywheel Inspection Elimination,” [**Reference 4.8.1.1**], supports the elimination of RCP flywheel inspections, based on the insignificant increase in probability of failure achieved by inspections over a 60-year service life, the relatively robust nature of flywheels with respect to detectable flaws, and the likelihood that disassembly and re-assembly for continued inspections presented the largest risk of causing flaws in the RCP flywheels. The estimated magnitude of fatigue crack growth during plant life was conservatively calculated based on an assumed initial radial crack length of 10% through the flywheel (from the keyway to the flywheel outer radius). The analysis assumed 6000 cycles of pump starts and stops for a 60-year plant life. Crack growth from postulated flaws in each flywheel was only a few mils. The existing analysis is valid for the period of extended operation. Reaching 6000 starts in 60 years would require a pump start on average every 3.7 days, which is extremely conservative. The findings of the analysis, which have been approved by the NRC, determined that the crack growth for the postulated flaw over 60 years of operation to be acceptable.

The existing analysis in WCAP-14535A is valid for the period of extended operation and demonstrates that the reactor coolant pump flywheels will continue to perform their design functions for the period of extended operation. Thus, Option (i) has been incorporated to satisfy 10 CFR 54.21 (c)(1) for the reactor coolant pump flywheels.

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. The intent of these plant specific analyses is to allow utilities to remove the dynamic effects of primary loop pipe ruptures from the design of reactor coolant loop pipe supports. In a letter dated January 11, 1993 [**Reference 4.8.2.2**], the NRC indicated concurrence with the VCSNS LBB analysis. Since the analysis considerations could be influenced by time, LBB is a TLAA for VCSNS.

The current LBB analysis accounts for the replacement of steam generators, power up-rate, and the RCS A-hot leg repair.

The LBB analysis is based on stainless steel at 40 years and will need to be revised to account for the proposed period of extended operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (ii) to develop a revised leak-before-break analysis for the period of extended operation.

The Thermal Fatigue Management Program will manage the Class 1 component thermal cycle count assumptions that form the foundation of this resolution, thus ensuring the continued validity of the LBB analysis, the management of thermal fatigue for these components, and the continued performance of their intended function(s) for 60 years of operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (iii) for the Thermal Fatigue Program as a whole.

4.7.3 CRANE LOAD CYCLE LIMIT

A potential TLAA issue was identified related to the cranes and associated crane supports that could theoretically affect irradiated fuel during refueling operations. The cranes that meet the criterion are as follows:

- Reactor Building Polar Crane
- Spent Fuel Cask Handling Crane
- Fuel Handling Machine (Spent Fuel Pit Bridge and Hoist)
- Refueling Machine (Reactor Cavity Manipulator Crane)

The Crane Manufacturers Association of America (CMAA) Specification No. 70 (CMAA 70) classifies these cranes as Class “A” cranes [**Reference 4.8.5.6**]. CMAA 70 Class A is defined in paragraph 2.2 as “cranes which may be used in installations such as powerhouses, public utilities, turbine rooms, motor rooms and transformer stations where precise handling of equipment at low speeds with long idle periods between lifts are required. Capacity loads may be handled for initial installation of equipment or infrequent maintenance.”

The current version of CMAA 70, Section 2.8, “Crane Service in Terms of Load Class and Load Cycles” states that Class A Cranes should be designed for 20,000 to 100,000 load cycles. However, the version of CMAA 70 in effect during VCSNS construction lists 20,000 to 200,000 load cycles. Since the cranes listed above were designed to CMAA 70, the load cycle limits apply. Cranes and crane supports are considered a TLAA because they satisfy the six criteria for a TLAA defined in 10 CFR 54.3.

4.7.3.1 Reactor Building Polar Crane

The Reactor Building polar crane, including the bridge girders, end trucks and trolley, were originally designed for construction loads of 360 tons. A seismic analysis was performed for the maximum, non-construction load of 150 tons. Since construction, the polar crane was only used for capacity lifts during the VCSNS steam generator replacement project. The steam generator lifts were rated capacity lifts of 354 tons. Lifts of the lower internals (135 tons), vessel head (125 tons), upper internals (52 tons), reactor coolant pumps, missile shields and other routine refueling operation lifts are commonly done during an outage and do not exceed the seismic load limit of 150 tons. Lifts of 150 tons or less do not qualify as capacity lifts since they are far less than the crane's rated capacity of 360 tons.

The number of lifts was based on one lift for each replaced (old) D-3 steam generator and one for each replacement (new) Delta 75 steam generator, which yields a total of six capacity lifts. Imposing an extremely conservative safety factor of five yields 30 lifts. Assuming a similar number of lifts during initial construction yields an estimate of 60 lifts. In addition, the crane lifted the reactor (330 tons) during construction. This conservative estimate of 61 lifts is exponentially less than the CMAA 70 limit of 200,000 cycles. Therefore, the crane is adequately analyzed and designed for fatigue through the term of extended operation.

4.7.3.2 Spent Fuel Cask Handling Crane

The spent fuel cask handling crane is rated for 125 tons. The projected number of fuel cask lifts is far less than 200,000 over a 60-year period. Assuming that (1) 70 fuel bundles are replaced every 18 months for 60 years, in addition to the original 157 bundles (equals 2957 bundle lifts), (2) each bundle is loaded individually into separate casks, and (3) each cast is lifted twice; the total number of lifts is less than 10,000 lifts. This estimate is extremely conservative and the total number still does not approach the design limit for the crane. Therefore, the spent fuel cask handling crane has been analyzed and shown to be adequate for the period of extended operation.

4.7.3.3 Fuel Handling Machine And Refueling Machine

The fuel handling machines consist of a fuel handling machine (Spent Fuel Pit Bridge and Hoist) and refueling machine (Reactor Cavity Manipulator Crane).

The refueling machine and fuel handling machine lift load consists of the combination of a spent fuel or new fuel assembly and handling tool. The maximum load weighs approximately 2500 pounds [**Reference 4.8.3.3**]. The refueling machine crane is rated for 3000 pounds.

The fuel handling machine hoist is designed with a margin of two for lifts. The hoist capacity is 4000 pounds while the combined weight of the fuel assembly with a rod cluster control assembly and the spent fuel assembly handling tool is approximately 2000 pounds. The fuel handling machine structure is designed to commercial standards and also analyzed to the requirements of Section III, Appendix XVII of the ASME Boiler & Pressure Vessel Code, and has the margins included in the allowable stresses of the Code [**Reference 4.8.5.7**].

Conservatively assuming 400 lifts each refueling cycle for each machine (i.e., loading 70 new fuel assemblies, a full core offload of 157 fuel assemblies, a full core reload of 157 fuel assemblies and 16 miscellaneous fuel assembly shuffles), and 40 refueling cycles in 60 years - results in approximately 16,000 cycles in 60 years.

The fuel handling cranes were analyzed for up to 200,000 cycles of maximum load based on the crane manufacturer's calculations and CMAA Specification No. 70. Since the conservative estimate of load cycles is far less than the design limit in CMAA 70, actual cycle counting is not warranted.

4.7.3.4 Summary

The VCSNS fuel handling machines (Spent Fuel Cask Handling Crane, Fuel Handling Machine, and Refueling Machine) and Reactor Building polar crane are adequately analyzed and designed for fatigue through the term of extended operation. Therefore, VCSNS elects to utilize 10 CFR 54.21(c)(1) - Option (i) to demonstrate that the fuel handling machines and Reactor Building polar crane are adequately analyzed for the period of extended operation.

4.7.4 SERVICE WATER INTAKE STRUCTURE SETTLEMENT

The Service Water Intake Structure (SWIS) is a reinforced concrete rectangular box culvert with two reinforced concrete wing walls at the intake end. The structure is mostly buried within the West Embankment. The portion not covered with soil is submerged within the Service Water Pond. The function of the intake structure is to draw water from the Service Water Pond into the Service Water Pump House (SWPH).

Section 3.7.2, “Seismic System and Subsystem Analysis,” of NUREG-0717 [**Reference 4.8.3.1**] evaluated the VCSNS SWIS. It was noted that excessive non-uniform settlement of the intake structure occurred during construction causing considerable cracking. This settlement was analyzed in a SWPH calculation which was originally based on a plant design life of 40 years. Since this issue meets all six criteria in 10 CFR 54.3, SWIS settlement is considered an actual TLAA for VCSNS.

The VCSNS calculation was revised to account for the period of extended operation (60 years). This revision demonstrates that the expected settlement is acceptable for the period of extended operation. Therefore, VCSNS incorporated Option (ii) to satisfy 10 CFR 54.21(c)(1) for the SWIS settlement.

4.8 REFERENCES

4.8.1 CALCULATIONS AND ANALYSES

4.8.1.1	Westinghouse WCAP-14535a, “Topical Report On Reactor Coolant Pump Flywheel Inspection Elimination,” November 1996.

4.8.2 CORRESPONDENCE

4.8.2.1	Letter, J. L. Skolds, SCE&G, to NRC Document Control Desk, dated June 30, 1992, “Virgil C. Summer Nuclear Station, Docket No. 50/395, Operating License No. NPF-12, Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, (LTR 920001).”
4.8.2.2	Letter, G. F. Wunder, NRC, to J. L. Skolds, SCE&G, dated January 11, 1993, “Safety Evaluation of Request to Use Leak-Before-Break for Reactor Coolant System Piping, Virgil C. Summer Nuclear Station, Unit 1 (TAC No. M83971).”
4.8.2.3	Letter, J. J. Hayes, NRC, to O. S. Bradham, SCE&G, dated April 28, 1989, “Issuance of Amendment No. 76 to Facility Operating License No. NPF-12, Virgil C. Summer Nuclear Station, Unit No. 1, Regarding Containment Structural Integrity (TAC No. 62803).”
4.8.2.4	Letter, O. S. Bradham, SCE&G, to NRC Document Control Desk, dated September 29, 1988, “Virgil C. Summer Nuclear Station, Docket No. 50/395, Operating License No. NPF-12, Containment Structural Integrity Surveillance Requirements.”
4.8.2.5	Letter, C. I. Grimes, NRC, to D. Walters, NEI, dated June 2, 1998, “Guidance on Addressing GSI 168 for License Renewal,” Project 690.
4.8.2.6	Memorandum, A. Thadani, NRC NRR, to W. Travers, NRC Operations, dated December 26, 1999, “Closeout of Generic Safety Issue 190, “Fatigue Evaluations of Metal Components for 60 Year Plant Life”

4.8.3 LICENSING DOCUMENTS AND REPORTS

4.8.3.1	NUREG-0717, "Safety Evaluation Report (SER) Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1," NRC, dated February 1981 (includes Supplements).
4.8.3.2	VCSNS Final Safety Analysis Report (FSAR), through Amendment 02-01.
4.8.3.3	NRC Safety Evaluation Report, EGG-HS-6371, "Control of Heavy Loads at Nuclear Power Plants," Virgil C. Summer Nuclear Station, Unit 1, dated May 23, 1985.

4.8.4 REGULATORY DOCUMENTS

4.8.4.1	NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NRC, April 2001.
4.8.4.2	NUREG-1801, "Generic Aging Lessons Learned Report," Volumes 1 and 2, NRC, April 2001.
4.8.4.3	NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."
4.8.4.4	NRC Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," March 6, 1992.
4.8.4.5	NRC Regulatory Guide 1.35, Proposed Revision 3, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments."

4.8.5 INDUSTRY DOCUMENTS

4.8.5.1	NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Nuclear Energy Institute, Revision 3, March 2001.

4.8.5.2	ASME “American Society of Mechanical Engineers” Boiler and Pressure Vessel Code Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components”, 1992 Edition and 1992 Addenda.
4.8.5.3	EPRI Final Report MRP-47, “Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47)” Revision 0, October 2001.
4.8.5.4	ASME “American Society of Mechanical Engineers” Boiler and Pressure Vessel Code Section III Nuclear Power Plant Components, Division I, (1971 Edition through Summer 1972).
4.8.5.5	American National Standards Institute, ANSI B31.1.0, “Power Piping Code,” 1967 issue with addenda through 1972.
4.8.5.6	Crane Manufacturers Association of America (CMAA) Specification No. 70 (CMAA 70), “Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes,” 1999.
4.8.5.7	ASME “American Society of Mechanical Engineers” Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division I, 1974 Edition and Addenda up to Winter 1975.

APPENDIX A - FSAR CHAPTER 18

INTRODUCTION

South Carolina Electric & Gas Company (SCE&G) has prepared an Application for Renewed Operating License of the Virgil C. Summer Nuclear Station (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 10 CFR 54.29.

Appendix A of the Application contains the FSAR Supplement for the Virgil C. Summer Nuclear Station.

10 CFR 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 10 CFR 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. Section 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 the attached FSAR Supplement.

FSAR SUPPLEMENT

South Carolina Electric & Gas Company (SCE&G) has prepared an Application for Renewed Operating License of the Virgil C. Summer Nuclear Station (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 10 CFR 54.29.

Appendix A of the Application contains the FSAR Supplement for the Virgil C. Summer Nuclear Station required by 10 CFR 54.21(d).

As appropriate, station documents will be revised or established, implemented, and maintained to cover the aging management programs and activities described in Chapter 18.

Insert new FSAR Chapter 18 to read as follows:

18.0 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

18.1 INTRODUCTION

South Carolina Electric & Gas Company prepared an Application for a Renewed Operating License of the Virgil C. Summer Nuclear Station (Application) [**Reference 18.4.1**]. 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 10 CFR 54.29 (Final Safety Evaluation Report - Final SER) [**Reference 18.4.2**]. Pursuant to the requirements of 10 CFR 54.21(d), 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 10 CFR 54.21 (a) and (c), respectively.

Table 18-1 provides a summary listing of the aging management programs and activities required for license renewal. Furthermore, Table 18-2 provides a summary listing of the evaluations of time-limited aging analyses (TLAA) required for license renewal. The first column of Table 18-1 and Table 18-2 provides a listing of aging management programs/activities and TLAA evaluations respectively. The second column of each table indicates where the issue is addressed in the Application. The third column of each table identifies where the description of the program/activity or TLAA is located in the Virgil C. Summer Nuclear Station FSAR.

Section 18.2 contains summary descriptions of the aging management programs and activities that are ongoing through the duration of the operating license of the Virgil C. Summer Nuclear Station, as well as any required one-time inspections.

Section 18.3 contains summary descriptions of the evaluations of time-limited aging analyses that are applicable through the duration of the extended operating license of Virgil C. Summer Nuclear Station.

Table 18-1:
SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES

Programs/Activities	Application Location	FSAR Location
10 CFR 50 Appendix J General Visual Inspection	B.1.11	18.2.1
10 CFR 50 Appendix J Leak Rate Testing	B.1.12	18.2.2
Above Ground Tank Inspection	B.2.1	18.2.3
Alloy 600 Aging Management Program	B.1.1	18.2.4
ASME Section XI ISI Program - IWF	B.1.13	18.2.5
Battery Rack Inspection	B.1.14	18.2.6
Boric Acid Corrosion Surveillances	B.1.2	18.2.7
Bottom-Mounted Instrumentation Inspection	B.1.3	18.2.8
Buried Piping and Tanks Inspection	B.2.10	18.2.9
Chemistry Program	B.1.4	18.2.10
Containment Coating Monitoring and Maintenance Program	B.1.15	18.2.11
Containment ISI Program – IWE/IWL	B.1.16	18.2.12
Diesel Generator Systems Inspection	B.2.2	18.2.13
Environmental Qualification (EQ) Program	B.3.1	18.2.14
Fire Protection Program (including Mechanical, Fire Barriers and Fire Barrier Penetration Seals, and Fire Doors activities)	B.1.5	18.2.15
Flood Barrier Inspection	B.1.17	18.2.16
Flow-Accelerated Corrosion Monitoring Program	B.1.6	18.2.17
Non-EQ Insulated Cables and Connections Inspection Program	B.2.9	18.2.18

Table 18-1:
SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES

Programs/Activities	Application Location	FSAR Location
In-Service Inspection (ISI) Plan	B.1.7	18.2.19
Inspections for Mechanical Components	B.2.11	18.2.20
Liquid Waste System Inspection	B.2.3	18.2.21
Maintenance Rule Structures Program	B.1.18	18.2.22
Material Handling System Inspection Program	B.1.19	18.2.23
Pressure Door Inspection Program	B.1.20	18.2.24
Preventive Maintenance Activities – Ventilation Systems Inspections	B.1.26	18.2.25
Reactor Building Cooling Unit Inspection	B.2.5	18.2.26
Reactor Head Closure Studs Program	B.1.8	18.2.27
Reactor Vessel Internals Inspection	B.2.4	18.2.28
Reactor Vessel Surveillance Program	B.1.24	18.2.29
Service Air System Inspection	B.2.6	18.2.30
Service Water Pond Dam Inspection Program	B.1.21	18.2.31
Service Water Structures Survey Monitoring Program	B.1.22	18.2.32
Service Water System Reliability and In-Service Testing Program	B.1.9	18.2.33
Small Bore Class 1 Piping Inspection	B.2.7	18.2.34
Steam Generator Management Program	B.1.10	18.2.35
Tendon Surveillance Program	B.3.3	18.2.36
Thermal Fatigue Management Program	B.3.2	18.2.37
Underwater Inspection Program (SWIS) and (SWPH)	B.1.23	18.2.38

Table 18-1:
SUMMARY LISTING OF THE AGING MANAGEMENT PROGRAMS AND ACTIVITIES

Programs/Activities	Application Location	FSAR Location
Waste Gas System Inspection	B.2.8	18.2.39
Heat Exchanger Inspections	B.2.12	18.2.40
Preventive Maintenance Activities - Terry Turbine	B.1.25	18.2.41

Table 18-2:
SUMMARY LISTING OF THE TLAA EVALUATIONS FOR LICENSE RENEWAL

Time Limited Aging Analyses	Application Location	FSAR Location
Crane Load Cycle Limit	4.7.3	18.3.6.1
Environmental Qualification (EQ)	4.4	18.3.3
Metal Fatigue - ASME Section III, Class 1	4.3.1	18.3.2.1
Metal Fatigue - Leak-Before-Break Analyses	4.7.2	18.3.2.2
Metal Fatigue - ASME Section III, Class 2 and 3 Piping Fatigue	4.3.2	18.3.2.3
Reactor Building Liner	4.6.1	18.3.5
Reactor Building Tendon Prestress	4.5	18.3.4
Reactor Coolant Pump Flywheel	4.7.1	18.3.6.3
Reactor Vessel Neutron Embrittlement – Upper-Shelf Energy	4.2.1	18.3.1.1
Reactor Vessel Neutron Embrittlement – Pressurized Thermal Shock	4.2.2	18.3.1.2
Reactor Vessel Neutron Embrittlement – Pressure-Temperature (P-T) Limits	4.2.3	18.3.1.3

Table 18-2:
SUMMARY LISTING OF THE TLAA EVALUATIONS FOR LICENSE RENEWAL

Time Limited Aging Analyses	Application Location	FSAR Location
Service Water Intake Structure Settlement	4.7.4	18.3.6.2

18.2 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The programs and activities described in the subsequent sections are credited for the management of aging under all current licensing basis conditions. Evaluation of the programs and activities provides reasonable assurance that subject systems, structures, and components are capable of performing their intended function(s) under all current licensing basis conditions. Aging management is provided through activities such as continued monitoring and assessment of conditions, trending and/or through control of system/structure parameters to preclude degradation. Under certain circumstances, one-time inspections are performed to ascertain plant conditions and/or confirm that degradation is not occurring.

Herein, the names of the programs, or activities are used only in the context of aging management during the period of extended operation and do not necessarily align with existing formalized VCSNS programs and/or procedures. The program and activity names used in this Application are intended to describe the collection of activities that are necessary to effectively manage aging.

18.2.1 10 CFR 50 APPENDIX J GENERAL VISUAL INSPECTION

Prior to conducting a 10 CFR 50 Appendix J Type A Integrated Leak Rate Test (ILRT), a general visual structural examination of the containment system is conducted. The general visual examination satisfies Technical Specification surveillance requirement 4.6.1.6.3. The inspection manages loss of material, cracking of welds, deformed structural attachments, and surface discontinuities associated with the containment liner, deterioration of moisture barriers, and deterioration of the Reactor Building structure.

18.2.2 10 CFR 50 APPENDIX J LEAK RATE TESTING

10 CFR 50 Appendix J Leak Rate Tests are required by Technical Specifications Surveillance Requirement 4.6.1.2. Type A and Type B Leak Rate Tests are performed as described further in **FSAR Section 6.2.6**. The testing program consists of monitoring of leakage rates through containment liner/welds, penetrations, fittings and other access openings for detection of degradation of the containment pressure boundary.

18.2.3 ABOVE GROUND TANK INSPECTION

The Above Ground Tank Inspection is a new one-time inspection activity that will determine if aging management is required for the internal surfaces of certain tanks and associated components (including pipe and valves) during the period of extended operation. The Above Ground Tank Inspection will detect and characterize loss of material due to galvanic and general corrosion in locations with exposure to moist air conditions, loss of material due to general corrosion in locations with exposure to treated water in which dissolved oxygen levels are not controlled, and loss of material and/or cracking due to the corrosive effects of alternate wetting and drying of treated or borated water. The Above Ground Tank Inspection will use suitable examination techniques at the most susceptible (sample) locations.

18.2.4 ALLOY 600 AGING MANAGEMENT PROGRAM

The purpose of the Alloy 600 Aging Management Program is to manage primary water stress corrosion cracking (PWSCC) of nickel-based alloy (Alloy 600 and 82/182) sub-components of the reactor vessel, pressurizer, and steam generators that are exposed to borated water to ensure that the pressure boundary function is maintained during the period of extended operation. The Alloy 600 Aging Management Program includes elements of the Boric Acid Corrosion Surveillances and the ASME Section XI System Pressure Test Program which detect the presence of system leakage and the ASME Section XI Inservice Examination Program which specifies the NDE techniques and acceptance criteria applied to the evaluation of identified cracks.

18.2.5 ASME SECTION XI ISI PROGRAM - IWF

The ASME Section XI Subsection IWF Inservice Inspection (ISI) Program manages loss of material for ASME Class 1, 2, and 3 piping supports (not including shock suppressors) and ASME Class 1, 2, and 3 major equipment supports, as well as cracking of high strength anchorage of ASME Class 1 component supports, for the extended period of operation. The ASME Section XI ISI Program - IWF was developed to implement the applicable requirements of 10 CFR 50.55a. The subsection IWF scope of inspection for supports is based on sampling of the total support population. The inspection program includes periodic volumetric, surface, and/or visual examination of component supports for signs of degradation and provides for corrective actions.

18.2.6 BATTERY RACK INSPECTION

The regulatory basis for inspecting battery racks is found in Technical Specifications Surveillance Requirement 3/4.8.2.1.c for the Electrical DC System and a commitment in the Fire Protection Evaluation Report (FPER) [**Reference 18.4.3**] for the Fire Service System. A visual inspection for loss of material due to corrosion is conducted for the Electrical DC System, in accordance with commitments in **FSAR Section 8.3.2.2.2**. A similar examination is conducted for the Fire Service System.

18.2.7 BORIC ACID CORROSION SURVEILLANCES

The purpose of the Boric Acid Corrosion Surveillances is to manage loss of material due to boric acid corrosion of mechanical and structural components constructed of susceptible materials located in the Reactor Building and in specific areas of the Auxiliary, Intermediate, or Fuel Buildings where borated water leakage is possible. The Boric Acid Corrosion Surveillances also manage boric acid intrusion into electrical equipment located in proximity to borated water systems. Elements of the Boric Acid Corrosion Surveillances include the identification of leakage locations, procedures for locating small leaks, and corrective actions to ensure that boric acid corrosion does not lead to degradation of structures and components that could cause loss of intended function.

18.2.8 BOTTOM-MOUNTED INSTRUMENTATION INSPECTION

The purpose of the Bottom-Mounted Instrumentation Inspection is to identify loss of material due to fretting (wear) in the bottom mounted instrumentation (BMI) thimble tubes prior to leakage in order to preclude a breach of the reactor coolant pressure boundary. Suitable inspection techniques are utilized and trended. The frequency of examination is based on wear rate relationships developed from Westinghouse research data.

18.2.9 BURIED PIPING AND TANKS INSPECTION

The purpose of the Buried Piping and Tanks Inspection is to manage loss of material on the external surfaces of buried components. The conditions of coatings and wrappings are determined by visual inspection whenever buried components are excavated, such as for maintenance. Degraded coatings or wrappings are indicative of potential surface corrosion of the external piping or tank surfaces and require further evaluation.

18.2.10 CHEMISTRY PROGRAM

The Chemistry Program controls the water chemistry in plant systems to minimize contaminant concentrations and adds chemicals, such as corrosion inhibitors and biocides, to manage loss of material, cracking, and fouling. The Chemistry Program is based on Electric Power Research Institute (EPRI) guidelines for primary and secondary water chemistry. The Chemistry Program includes specifications for chemical species, limits, sampling and analysis frequencies, and corrective actions for primary, secondary, and auxiliary (borated or treated) water systems, as well as for oil and fuel oil.

18.2.11 CONTAINMENT COATING MONITORING AND MAINTENANCE PROGRAM

The Containment Coating Monitoring and Maintenance Program provides for maintenance of protective coatings inside the containment. Maintenance of protective coatings manages loss of material due to corrosion. Visual inspections and condition assessments of certain coatings inside containment are periodically conducted as part of the containment structural integrity verification, Maintenance Rule monitoring, general maintenance planning, and during recovery from refueling outages. Containment coatings are visually inspected via walkdowns from accessible floors, platforms or other permanent vantage points. The degree of examination depends on many factors such as accessibility, environmental and radiological conditions, and safety. In cases of inaccessibility, sampling approaches based on plant specific characteristics, industry wide experience and testing history are evaluated in lieu of actual visual inspections. Further discussions relevant to containment coatings is located in **FSAR Appendix 3A** (RG 1.54) and **FSAR Section 6.2.1.6**.

18.2.12 CONTAINMENT ISI PROGRAM - IWE/IWL

10 CFR 50.55a(g)(4) requires a detailed visual examination of the containment system for structural anomalies in accordance with ASME Section XI Subsections IWE, “Requirements for Class MC and Metallic Liners of Class CC Components for Light-Water Cooled Power Plants”, and IWL, “Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants” throughout the service life of nuclear power plants. The inspection program includes periodic volumetric, surface and visual examination of concrete and liner surfaces for signs of degradation and provides for corrective actions.

18.2.13 DIESEL GENERATOR SYSTEMS INSPECTION

The Diesel Generator Systems Inspection is a new one-time inspection activity that will determine if aging management is required for certain carbon steel diesel generator support system components during the period of extended operation. The Diesel Generator Systems Inspection will detect and characterize loss of material due to general corrosion and/or corrosive impacts of alternate wetting and drying in pertinent starting air components. The Diesel Generator Systems Inspection will use suitable examination techniques at the most susceptible (sample) locations.

18.2.14 ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM

The NRC has established environmental qualification (EQ) requirements in 10 CFR 50.49 and Appendix A (Criterion 4) to 10 CFR Part 50. EQ component aging limits are not based on condition or performance monitoring; however, such monitoring programs are an acceptable basis for modifying aging limits. Monitoring or inspection of environmental, condition, or component parameters may be used to ensure that the component is within its qualification, or to provide a basis to modify the qualification analyses. The EQ Program quantifies the plant service conditions (i.e., the operating environments) for defined environmental zones, such that the severity of the aging effects (in comparison to other plant locations) can be determined. Further discussion of the EQ Program is contained in **FSAR Section 3.11** and Appendix 3A (RG 1.89).

18.2.15 FIRE PROTECTION PROGRAM

The Fire Protection Program utilizes the concept of defense-in-depth to achieve a high degree of fire safety as discussed in the Fire Protection Evaluation Report (FPER) [**Reference 18.4.3**]. The Fire Protection Program provides administrative requirements for ensuring the operability of equipment required to ensure safe plant shutdown. The program includes visual inspections, system flushing, and performance tests of fire barriers, fire doors, and fire suppression system components. As described in the FPER, the Fire Protection Program includes the requirements identified in Appendix A of APCSB 9.5.1 and 10 CFR 50 Appendix R, Sections III.G, III.J and III.O. Additional description of the portions of the program pertinent to the management of aging is provided below.

18.2.15.1 Mechanical

The Fire Protection Program includes the performance of flow tests and flushes to ensure that blockage of flow will not occur, performance testing of individual components to ensure they maintain their component intended function, and visual inspections to verify sprinkler and associated component condition. Fire suppression system components (e.g. piping, valves, nozzles, sprinkler heads, hydrants) are included within the scope of the mechanical inspections and tests. The normal Fire Service System pressure is monitored to provide further indication of the ability to maintain system function. Flow tests and flushes are conducted on the main distribution loops via hydrant testing. Performance testing is conducted on selected Fire Service System components based on NFPA recommendations. Visual inspections of Fire Service System components (e.g. sprinklers, hydrants, above ground piping) are periodically conducted to identify corrosion on the exterior surface, physical damage or obstructions that might impede performance of the intended functions.

In addition, disassembly/replacement of representative sprinkler heads in branch lines that do not receive flow during periodic testing is to be conducted in accordance with NFPA standards. Ultrasonic testing of a representative sample of these stagnant section of piping will be conducted at 10 year intervals.

A one-time inspection of the Fire Service System will be performed to determine if aging management is required for brass and cast iron components during the period of extended operation. The inspection activity will detect and characterize loss of material due to selective leaching. This inspection will use suitable hardness measurement techniques at the most susceptible (sample) locations.

18.2.15.2 Fire Barriers And Fire Barrier Penetration Seals

The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, and fire barrier walls, ceilings, and floors to ensure that their operability is maintained.

18.2.15.3 Fire Doors

Fire rated door inspections are performed. Examination guidelines and results of periodic inspections of fire rated doors are provided. Inspections are credited with managing loss of material of doors and door hardware for the period of extended operation.

18.2.16 FLOOD BARRIER INSPECTION

Periodic visual inspections are performed for flood barriers (walls, curbs, equipment pedestals), flood doors, and flood barrier penetration seals. The Flood Barrier Inspection activity is a subset of the Maintenance Rule Structures Program and the Fire Protection Program. The inspections serve to detect cracking and loss of material prior to loss of component intended function.

18.2.17 FLOW-ACCELERATED CORROSION MONITORING PROGRAM

The purpose of the Flow-Accelerated Corrosion Monitoring Program is to manage loss of material for components located in systems within the scope of license renewal which are susceptible to flow-accelerated corrosion (FAC) (also called erosion-corrosion). This program is intended to mitigate FAC by combining the following elements: NUREG guidelines, predictive analysis, inspections, industry experience, station information gathering and communication, engineering judgement, and long-term mitigative strategies to reduce FAC wear rates.

18.2.18 NON-EQ INSULATED CABLES AND CONNECTIONS INSPECTION PROGRAM

The Non-EQ Insulated Cables and Connections Inspection Program provides for visual inspection of instrument as well as power and control cables as a means to identify age-related degradation due to localized ambient thermally and radiologically induced stress prior to significant loss of insulation resistance. The program will be performed at 10-year intervals, with the initial inspection to be performed prior to the period of extended operation. The program will involve a visual inspection of the accessible cables in selected environmental zones, to determine if the cable jackets show any signs of cracking, embrittlement, discoloration, melting, or any other visible evidence of age-related degradation which may indicate loss of insulation resistance. Guidance from the EPRI “Guideline for the Management of Adverse Localized Equipment Environments,” [**Reference 18.4.4**] will be used for these inspections.

18.2.19 IN-SERVICE INSPECTION (ISI) PLAN

The In-Service Inspection (ISI) Plan implements the requirements of 10 CFR 50.55a for Class 1, 2 and 3 components is in accordance with ASME Section XI, Subsections IWB (Class 1), IWC (Class 2), and IWD (Class 3). The program consists of periodic volumetric, surface, and/or visual examination of components for signs of degradation and provides for corrective actions. 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 VCSNS will contain the fourth, fifth, and sixth ten-year inservice inspection intervals. The program is addressed further in **FSAR Section 5.7**.

18.2.20 INSPECTIONS FOR MECHANICAL COMPONENTS

The Inspections for Mechanical Components manage loss of material and cracking for mechanical components constructed of susceptible materials and exposed to ambient conditions. The inspections involve a visual examination of the exposed external surfaces of representative mechanical components. The inspections and associated evaluations also address conditions in locations susceptible to external pitting corrosion due to the presence of insulation materials and the potential for condensation to occur.

18.2.21 LIQUID WASTE SYSTEM INSPECTION

The Liquid Waste System Inspection is a new one-time inspection activity that will determine if aging management is required for certain stainless steel pipe, valves and heat exchanger components during the period of extended operation. The Liquid Waste System Inspection will detect and characterize loss of material due to crevice and pitting corrosion, and cracking due to stress corrosion cracking in systems and components containing unmonitored and uncontrolled water. The Liquid Waste System Inspection will use a combination of volumetric and visual examination techniques at the most susceptible (sample) locations.

18.2.22 MAINTENANCE RULE STRUCTURES PROGRAM

The Maintenance Rule Structures Program for the inspections of structures and structural components meets the regulatory requirements of 10 CFR 50.65, the Maintenance Rule. Visual inspections and condition assessments of structures and structural components are conducted in accordance with the requirements of the Maintenance Rule. The structures and structural components are visually inspected via walkdowns from accessible floors, platforms or other permanent vantage points. In cases of inaccessibility, sampling approaches are evaluated. The Maintenance Rule Structures Program includes chemical analysis of raw water (groundwater, Service Water Pond, reservoir, rainwater) per the Maintenance Rule intervals in support of condition assessment.

18.2.23 MATERIAL HANDLING SYSTEM INSPECTION PROGRAM

The Material Handling System Inspection Program manages loss of material for applicable steel rails and girders. The Material Handling System Inspection Program has been in effect for many years at VCSNS and includes Nuclear Safety Related and Quality Related (seismically restrained) material handling system components. Material handling systems steel support structures (rails, runways, monorails, girders, jib cranes, seismic restraints, and associated connections) are inspected in accordance with guidance provided by ANSI standards. Inspections are implemented in the course of routine maintenance.

18.2.24 PRESSURE DOOR INSPECTION PROGRAM

The Pressure Door Inspection Program provides examination guidelines for periodic inspections of pressure doors. VCSNS pressure doors are Nuclear Safety Related or Quality Related. Most Nuclear Safety Related pressure doors are also fire doors. Pressure door inspection attributes include freedom of movement, function (closed during normal plant operation), structural deterioration, and loss of door/door hardware material. Pressure doors are required to be operable in Plant Operating Modes 1, 2, 3, and 4. The surveillance requirements include monitoring of door position and visual inspection that the door is closed and not impaired.

18.2.25 PREVENTIVE MAINTENANCE ACTIVITIES - VENTILATION SYSTEM INSPECTIONS

The Preventive Maintenance Activities - Ventilation Systems Inspections manage loss of material and fouling in susceptible components. Susceptible components include those components in the Air Handling, Local Ventilation and Component Cooling Systems that are exposed internally to moist air. Routine maintenance inspections are conducted which include detection of age-related degradation.

18.2.26 REACTOR BUILDING COOLING UNIT INSPECTION

The Reactor Building Cooling Unit Inspection is a new one-time inspection activity that will determine if aging management is required for reactor building cooling unit drain lines during the period of extended operation. The Reactor Building Cooling Unit Inspection will detect and characterize loss of material or cracking resulting from exposure to an unmonitored and uncontrolled (alternately wetted/dried) water environment. The Reactor Building Cooling Unit Inspection will use volumetric and/or visual examination techniques at the most susceptible (sample) locations in the reactor building cooling unit drain lines.

18.2.27 REACTOR HEAD CLOSURE STUDS PROGRAM

The purpose of the Reactor Head Closure Studs Program is to manage loss of mechanical closure integrity due to stress relaxation, stress corrosion cracking, and wear for the alloy steel components of the reactor vessel closure stud assembly. The In-Service Inspection (ISI) Plan portion of the Reactor Head Closure Studs Program detects cracking and loss of material through the use of surface examination (magnetic particle or liquid penetrant) and/or ultrasonic examination. Reactor vessel studs may be examined in place (under tension), when the connection is disassembled, or when the studs are removed. Further discussions of the inspection and protection of reactor vessel closure stud assemblies is contained in **FSAR Section 5.4**.

18.2.28 REACTOR VESSEL INTERNALS INSPECTION

The Reactor Vessel Internals Inspection supplements the In-Service Inspection (ISI) Plan to assess the condition of reactor vessel internals in order to ensure that the intended functions are maintained during the period of extended operation. The inspection provides examination techniques and engineering evaluations to address the aging effects listed below:

- Changes in dimensions due to irradiation creep and void swelling
- Cracking due to irradiation-assisted stress corrosion cracking
- Cracking due to primary water stress corrosion cracking (PWSCC) for nickel-based materials
- Loss of material due to wear
- Loss of preload due to stress relaxation
- Reduction of fracture toughness due to irradiation embrittlement and void swelling

For those components that are accessible or can be rendered accessible by the removal of the core and other internals for examination, a visual inspection is performed to detect the presence and extent of cracking and loss of material. For bolts and for inaccessible components, a volumetric inspection is performed to detect the presence and extent of changes in dimensions, cracking, loss of preload, and reduction of fracture toughness. With respect to changes in dimensions due to void swelling, industry activities (including WOG and EPRI) are under way to better characterize the effect and, if necessary, to develop and qualify methods for detection and management. These activities will be monitored by VCSNS and implemented, as applicable.

18.2.29 REACTOR VESSEL SURVEILLANCE PROGRAM

The purpose of the Reactor Vessel Surveillance Program is to manage reduction of fracture toughness due to irradiation embrittlement of reactor vessel beltline materials to assure that the pressure boundary function is maintained for the period of extended operation. The program includes an evaluation of radiation damage based on testing of Charpy V-notch and tensile specimens. The Reactor Vessel Surveillance Program conforms to 10 CFR 50, Appendix H. Further discussions of this program are contained in **FSAR Section 5.4.3**.

Additionally, a one-time demonstration is a necessary part of this program to ensure that materials in the upper shell/nozzle course of the reactor vessel do not become limiting during the period of extended operation, with respect to radiation damage.

18.2.30 SERVICE AIR SYSTEM INSPECTION

The Service Air System Inspection is a new one-time inspection activity that will determine if aging management is required for certain carbon steel pipe and valves during the period of extended operation. The Service Air System Inspection will detect and characterize loss of material due to general corrosion on internal surfaces of subject components resulting from exposure to moist air. The Service Air System Inspection will use a combination of volumetric and visual examination techniques at the most susceptible (sample) locations.

18.2.31 SERVICE WATER POND DAM INSPECTION PROGRAM

The purpose of the Service Water Dam Inspection Program is to assess the condition of the Service Water Pond Dams and the West Embankment with respect to loss of material (erosion), alignment (movement), surface cracking and seepage that could result in loss of component intended function. Additionally, the submerged slope stability of the West Embankment in the vicinity of the Intake Structure is monitored as specified by VCSNS Operating License Number NPF-12, Condition 2.C.5. SCE&G conducts annual walkdowns of the Service Water Pond Dams scheduled in the course of routine maintenance.

18.2.32 SERVICE WATER STRUCTURES SURVEY MONITORING PROGRAM

Survey monitoring is required for structures that are supported by earthen fill material and which have exhibited the potential for settlement. Settlement is not considered to be adverse unless it imposes stresses on a structure that could exceed the design values. Initial settlement of the Service Water Pump House and Service Water Intake Structure was much more than the original pre-construction estimates. As a result, survey monitoring of the Service Water Pumphouse, Service Water Intake Structure, Electrical Duct Banks, and Service Water Intake Line “A” is conducted to satisfy the requirements specified by Operating License Condition 2.C.5 and **FSAR Section 2.5.4.10.6.2**. The purpose of the surveys is to monitor and evaluate any differential in vertical and horizontal displacement in order to identify settlement issues prior to their resulting in significant degradation or loss of function.

18.2.33 SERVICE WATER SYSTEM RELIABILITY AND IN-SERVICE TESTING PROGRAM

The purpose of the Service Water System Reliability and In Service Testing Program is to manage cracking, fouling, and loss of material for susceptible materials located in systems that contain raw water from the Service Water Pond. The Service Water System Reliability and In Service Testing Program is intended to detect the presence of and assess the extent of cracking, fouling and loss of material. The program also serves to mitigate aging effects through the use of chemical additives in order to minimize fouling. Visual inspections of Service Water System piping and components are conducted on a periodic basis to monitor the extent of cracking, fouling, and loss of material. The heat transfer capabilities of heat exchangers serviced by the Service Water System are evaluated to detect the presence of fouling.

18.2.34 SMALL BORE CLASS 1 PIPING INSPECTION

The Small Bore Class 1 Piping Inspection is a new one-time inspection activity that will determine if aging management is required for cracking due to flaw growth and stress corrosion cracking for Reactor Coolant System stainless steel piping and fittings less than four inches NPS. The Small Bore Class 1 Piping Inspection will serve to increase confidence in the current condition of small bore Reactor Coolant System piping which does not receive a volumetric examination per the ASME Code. The Small Bore Class 1 Piping Inspection will use suitable examination techniques at the most susceptible (sample) locations.

18.2.35 STEAM GENERATOR MANAGEMENT PROGRAM

The purpose of the Steam Generator Management Program is to perform examinations of nickel-based alloy steam generator tubes and tube plugs to ensure that cracking and loss of material are identified and corrected prior to exceeding allowable limits. The program implements the requirements of Technical Specification 4.4.5. The program follows the recommendations provided by NEI and EPRI guidelines.

18.2.36 TENDON SURVEILLANCE PROGRAM

The Tendon Surveillance Program meets the requirements of the ASME Code, Section XI, Subsection IWL, as supplemented by the requirements of 10 CFR 55.55a(b)(2)(viii). The tendon lift-off forces are evaluated to ensure that the rate of tendon force loss is within predicted limits and that a minimum required tendon force level exists in the Reactor Building. Degradation of the Reactor Building post tensioning system is detected by periodic inspections of randomly selected tendons. Further discussion pertinent to tendon stresses is contained in **FSAR Appendix 3A** (1.35).

18.2.37 THERMAL FATIGUE MANAGEMENT PROGRAM

As defined for license renewal, the Thermal Fatigue Management Program seeks to preclude cracking due to low-cycle thermal fatigue by managing the thermal fatigue basis. Management of the fatigue basis is accomplished by continually showing that the severity and number of occurrences of the actual transients are enveloped (bounded) by the severity and number of occurrences of the analyzed transients. The program documents and evaluates plant operational transients/cycles using a specially designed computer program. The evaluation process includes comparison of the number of thermal cycles incurred against the design transient limit and/or calculation of cumulative usage factors (CUF) and verification that adequate margin exists. The Thermal Fatigue Management Program will incorporate applicable industry guidance once it is finalized to account for environmental effects of the reactor coolant fluid.

18.2.38 UNDERWATER INSPECTION PROGRAM (SWIS AND SWPH)

Underwater inspections of the Service Water Intake Structure are conducted and the conditions assessed in accordance with the requirements of VCSNS Operating License NPF-12, Condition 2.C.5.d. The purpose of the inspection is to monitor the condition of existing cracks in the Service Water Intake Structure that originated during construction due to greater than expected settlement of the structure. Underwater inspections of the Service Water Intake Structure and the Service Water Pump House also serve to monitor corrosion and fouling within the Service Water System.

18.2.39 WASTE GAS SYSTEM INSPECTION

The Waste Gas System Inspection is a new one-time inspection activity that will determine if aging management is required for certain stainless steel components of the Gaseous Waste Processing System during the period of extended operation. The Waste Gas System Inspection will detect and characterize loss of material due to crevice and pitting corrosion in portions of the system exposed to unmonitored and uncontrolled treated water, and cracking due to stress corrosion cracking in portions of the system containing gas. The Waste Gas System Inspection will use a combination of volumetric and visual examination techniques at the most susceptible (sample) locations.

18.2.40 HEAT EXCHANGER INSPECTIONS

The Heat Exchanger Inspections are a new one-time inspection activity that will determine if aging management is required for certain malleable heat exchanger components during the period of extended operation. The Heat Exchanger Inspections will detect and characterize loss of material due to selective leaching and erosion-corrosion, as well as particulate fouling. The Heat Exchanger Inspections will use a combination of volumetric and visual examination and hardness measurement techniques at the most susceptible (sample) locations.

18.2.41 PREVENTIVE MAINTENANCE ACTIVITIES - TERRY TURBINE

The purpose of the Preventive Maintenance Activities - Terry Turbine is to manage loss of material in carbon steel due to crevice, general, or pitting corrosion of the turbine casing and associated components that are normally exposed to ambient conditions with periodic exposure to steam allowing moisture to accumulate. The Preventive Maintenance Activities - Terry Turbine is a condition monitoring program composed of controlled plant procedures. Routine maintenance inspections are conducted which include detection of age-related degradation and initiation of corrective actions as necessary.

18.3 TIME-LIMITED AGING ANALYSES (TLAA) EVALUATIONS

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) be provided for the period of extended operation. The following TLAAs have been identified and evaluated to meet this requirement.

18.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is subjected to neutron irradiation from the core. This irradiation results in the embrittlement of the reactor vessel materials. Analyses have been performed that address the following:

- Upper-Shelf Energy
- Pressurized Thermal Shock
- Pressure-Temperature (P-T) Limits

18.3.1.1 Upper-Shelf Energy

The requirements on reactor vessel Charpy Upper-Shelf Energy are included in 10 CFR 50, Appendix G. The Charpy Upper-Shelf Energy must be maintained at no less than 50 ft.-lb. throughout the life of the reactor vessel.

Charpy Upper-Shelf Energy values have been calculated to the end of current license (32 EFPY) to be 67.5 ft.-lb. for the limiting component of the beltline region materials. Additional analyses are required to calculate Charpy Upper-Shelf Energy for the end of the period of extended operation. The remaining capsules must incur additional exposure to neutron fluence in order to provide data that correlates to estimated fluence on the vessel at the end of the period of extended operation. Following adequate capsule exposure, a capsule will be withdrawn and analyzed. The Charpy Upper-Shelf Energy will be recalculated for additional fast neutron fluence corresponding to the end of the extended operating period. Therefore, the Charpy Upper-Shelf Energy analyses will be projected for the end of the period of extended operation.

18.3.1.2 Pressurized Thermal Shock

The requirements of 10 CFR 50.61 provide for protection against pressurized thermal shock events for pressurized water reactors. The screening criteria established by 10 CFR 50.61 are 270°F for plates, forgings, and axial weld materials and 300°F for circumferential weld materials. According to 10 CFR 50.61, if the calculated reference temperature (RT_{PTS}) for the limiting beltline materials is less than the specified screening criteria, then the vessel is acceptable with regard to the risk of vessel failure during postulated pressurized thermal shock transients. The regulations require updating the pressurized thermal shock assessment upon a request for a change in the expiration date of the facility operating license.

The RT_{PTS} values for VCSNS have been calculated for the end of the current 40-year license term. The RT_{PTS} values require revision in order to project the values to the end of the period of extended operation. This cannot be accomplished until the surveillance capsules are exposed to sufficient neutron fluence to provide data corresponding to the estimated fluence on the vessel wall at the end of 60 years. The analyses will be revised to project the RT_{PTS} to the end of the period of extended operation using the methods provided in 10 CFR 50.61.

18.3.1.3 Pressure-Temperature (P-T) Limits

Appendix G to 10 CFR 50 requires that heatup and cooldown of the reactor pressure vessel shall be accomplished within established pressure-temperature limits. The pressure-temperature limits are established by calculations that utilize the materials and fluence data obtained through the site 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 term.

Pressure-temperature limit curves have been calculated for 32 EFPY that correspond to the end of the current license term but not the period of extended operation. The pressure-temperature limit curves will be recalculated following the removal of one of the remaining surveillance capsules from the vessel. The surveillance capsule will be removed when the calculated fast neutron fluence on the capsule meets or exceeds the calculated fast neutron fluence on the vessel wall at the end of the period of extended operation. The Technical Specifications will be updated as required by 10 CFR 50. Therefore, the pressure-temperature limit analyses will be projected for the period of extended operation.

18.3.2 METAL FATIGUE

The thermal fatigue analyses of the station's mechanical components have been identified as time-limited aging analyses.

18.3.2.1 ASME Section III, Class 1

The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that the initiation of fatigue cracks is precluded.

Experience has shown that the transients used to analyze the ASME III requirements are often very conservative. The magnitude and frequency of the design transients are more severe than those occurring during plant operation. The magnitude and number of actual transients are monitored. This monitoring assures that the existing frequency and magnitude of transients are conservative and bounding for the period of extended operation, and that the existing ASME III equipment will perform its intended functions for the period of extended operation. A program for thermal transient cycle counting and analysis is in place and provides reasonable assurance that the actual transients are smaller in magnitude and within the number of the transients used in the design.

18.3.2.2 Leak-Before-Break Analyses

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.

The leak-before-break analyses are currently valid for 40 years. The analyses require revision in order to demonstrate that the design is adequate for the extended period of operation.

18.3.2.3 ASME Section III, Class 2 and 3 Piping Fatigue

Piping systems, designed in accordance with ASME Section III, Class 2 and 3 or ANSI B31.1, utilize allowable stress values based on a stress reduction factor to account for thermal cycles during normal operation. Adequate margin is available to account for 60 years of plant operation in the current analyses for the majority of the plant systems reviewed. Restrictions on sampling activities, or reanalysis, is required on the “B” Reactor Coolant System loop sampling line in order to account for 60 years of plant operation.

Metal fatigue for the ASME Section III, Class 2 and 3 and ANSI B31.1 piping systems has been determined to be valid for the period of extended operation.

18.3.3 ENVIRONMENTAL QUALIFICATION (EQ)

The qualification analyses for some electrical equipment included in the Environmental Qualification (EQ) Program have been identified as time-limited aging analyses for license renewal. The qualification analyses for electrical equipment with a 40-year or greater qualified life have been determined to be time-limited aging analyses.

Equipment included in the VCSNS EQ Program will be evaluated to determine if existing environmental qualification analyses can be projected to the end of the period of extended operation by reanalysis or additional analysis. Qualification into the license renewal period is treated the same as for equipment currently qualified at VCSNS for 40 years or less. When aging analyses cannot justify a qualified life to the end of the period of extended operation then the components or parts will be replaced prior to exceeding their qualified lives in accordance with the EQ Program.

The existing EQ process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation.

18.3.4 REACTOR BUILDING TENDON PRESTRESS

The Reactor Building was prestressed in order to have low-strain linear response at design loads and thus assure integrity of the liner. The exterior wall is post-tensioned in both vertical and hoop directions. On the dome a three-way post-tensioning system is employed. The pre-stress of the containment tendons decreases over the life of the plant due to elastic deformation, creep, anchorage seating losses, tendon wire friction, stress relaxation and corrosion. Periodic inspections include examination of selected tendon parameters and provide data for prestress analyses. Tendon prestress analyses are used to determine if addition retensioning is required before the next scheduled inspection based on the state of the tendon stress. Therefore reactor building tendon prestress is a time-limited aging analysis.

The existing Tendon Surveillance Program will ensure that the Reactor Building tendons are analyzed for the period of extended operation.

18.3.5 REACTOR BUILDING LINER

The Reactor Building is lined on the inside face with a steel plate that provides an essentially leak-tight barrier. The liner is designed to remain within strain limits associated with serviceability in accordance with the ASME B&PV Code for normal operation.

The reactor building liner calculations evaluate liner fatigue for a 40 year period and conclude that the liner meets the criteria of ASME NB 3222.4 (d) for the suitability for cyclic condition and no fatigue analysis is required. The reactor building liner analyses has been revised for the period of extended operation.

18.3.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

18.3.6.1 Crane Load Cycle Limit

The crane load cycle limit was identified as a time-limited aging analysis for the cranes within the scope of license renewal. The cranes within the scope of license renewal are listed below.

- Reactor Building Polar Crane
- Fuel Handling Machine (Spent Fuel Pit Bridge and Hoist)
- Refueling Machine (Reactor Cavity Manipulator Crane)
- Spent Fuel Cask Handling Crane

The cranes listed above are classified as Class “A” cranes by the Crane Manufacturers Association of America Specification No. 70 (CMAA 70) which specifies a design limit for the number of load cycles for the life of a crane. The load cycles for these cranes have been evaluated for the period of extended operation. For each crane, the actual usage over the projected life through the period of extended operation will be far less than the analyzed quantity of cycles. All the cranes in the scope of license renewal will continue to perform their intended function through the period of extended operation.

Therefore, the analyses associated with crane design, including fatigue, are valid for the period of extended operation.

18.3.6.2 Service Water Intake Structure Settlement

The Service Water Intake Structure is a reinforced concrete rectangular box culvert with two reinforced concrete wing walls at the intake end. The structure is mostly buried within the West Embankment. The portion not covered with soil is submerged within the Service Water Pond. The function of the Service Water Intake Structure is to draw water from the Service Water Pond into the Service Water Pump House. Excessive non-uniform settlement of the intake structure occurred during construction resulting in cracking of the structure. The settlement of the structure was analyzed based on a plant design life of 40 years. Therefore, Service Water Intake Structure settlement is a time-limited aging analysis for VCSNS.

The Service Water Intake Structure settlement calculation has been revised to evaluate the settlement of the structure for the period of extended operation.

18.3.6.3 Reactor Coolant Pump Flywheel

The reactor coolant pump motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down and assuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway from stresses due to starting the motor. An analysis has been performed to estimate the magnitude of fatigue crack growth during the plant life. The analysis assumes 6,000 cycles of pump starts and stops for a 60-year plant life.

The analysis associated with the reactor coolant pump flywheel has been evaluated and determined to remain valid for the period of extended operation.

18.4 REFERENCES

18.4.1	LRA (later)
18.4.2	SER (later)
18.4.3	VCSNS Fire Protection Evaluation Report (FPER), Amendment 02-01.
18.4.4	EPRI Report TR-109619, “Guideline for the Management of Adverse Localized Equipment Environments,” June 1999

APPENDIX B - AGING MANAGEMENT PROGRAMS AND ACTIVITIES

INTRODUCTION

For those structures and components that are identified as being subject to an aging management review, 10 CFR 54.21(a)(3) requires demonstration that the effects of aging will be adequately managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

The NRC and the industry identified ten (10) program elements that would be useful in describing an aging management program and then demonstrating its effectiveness. These program elements are described in Appendix A.1, Section A.1.2.3 of NUREG-1800 [**Reference B-1**]. NUREG-1801, Generic Aging Lessons Learned (GALL) Report [**Reference B-2**], applies these program elements to evaluate the adequacy of generic aging management programs in managing certain aging effects.

VCSNS applied the program elements in the review of VCSNS programs and activities to demonstrate that the effects of aging will be adequately managed in accordance with the License Renewal Rule. There are two types of VCSNS programs that are evaluated in this manner. The first is a VCSNS program that is evaluated against a program in NUREG-1801. The second is a VCSNS plant-specific program.

For VCSNS aging management programs being evaluated against a program in NUREG-1801, the criteria or activities delineated in each of the 10 elements are reviewed. A conclusion is reached concerning consistency for the VCSNS program with each NUREG-1801 recommended program elements and a demonstration of overall program effectiveness is made. Any program enhancements that are required are documented. Clarification is provided for instances where the VCSNS program does not match specific details of a NUREG-1801 program element but is still determined to be consistent. Finally, an overall determination is made as to consistency with the program description in NUREG-1801.

For plant-specific aging management programs, an evaluation is performed to document how VCSNS meets the 10 generic program elements in NUREG-1800. A determination of overall program effectiveness and any required program enhancements are identified.

The VCSNS Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG-1800, "Standard Review Plan for Review of License Renewal." A description of the VCSNS Quality Assurance Program is contained in **FSAR Section 17.2**. The Quality Assurance Program addresses three of the aging management program elements: 1) Corrective Action, 2) Confirmation Process (which is an integral part of the Corrective Action), and 3) Administrative Controls. The Quality Assurance Program applies these program elements via existing corrective action and document control programs. VCSNS will employ the corrective action and document control programs to address the program elements of corrective action, confirmation process, and administrative (document) controls for both safety-related and non-safety-related structures and components that require aging management during the period of extended operation.

The VCSNS aging management programs described herein are credited for managing the effects of aging. These programs include existing programs, existing programs that have been enhanced to deal with specific aging effects, and new programs not currently defined in VCSNS administrative controls. The programs provide reasonable assurance that the effects of aging will be adequately managed so that the structures and components subject to aging management will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. The demonstrations, along with the program and activity descriptions, meet the requirements of 10 CFR 54.21(a)(3). Along with the technical information contained in the body of this application, this appendix is intended to allow the NRC to make the finding required by 10 CFR 54.29(a)(1).

The results of VCSNS aging management program evaluations are documented in the following sections. Each aging management program presented in this Appendix is characterized as one of the following:

> **Existing Aging Management Program (Section B.1.0)**: A current program or activity that will continue to be implemented during the extended period of operation to manage aging. Any required enhancements to the program or activity will be implemented prior to the period of extended operation.
>
> **New Aging Management Program (Section B.2.0)**: A program or activity that does not currently exist, which will manage aging during the extended period of operation.
>
> **TLAA Support Program (Section B.3.0)**: A program or activity that supports the basis for a time-limited aging analysis during the period of extended operation.

Table B-1 presents the correlation between the programs evaluated in NUREG-1801 and the VCSNS programs credited with aging management.

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
Chapter X			
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Thermal Fatigue Management Program	B.3.2
X.E1	Environmental Qualification (EQ) of Electric Components	Environmental Qualification (EQ) Program	B.3.1
X.S1	Concrete Containment Tendon Prestress	Tendon Surveillance Program	B.3.3
Chapter XI			
XI.M1	ASME Section XI Inservice Inspection, Subsection IWB, IWC, IWD	In-Service Inspection (ISI) Plan	B.1.7
XI.M2	Water Chemistry	Chemistry Program	B.1.4
XI.M3	Reactor Head Closure Studs	Reactor Head Closure Studs Program	B.1.8
XI.M4	BWR Vessel ID Attachment Welds	Not applicable, VCSNS is a PWR.	N/A
XI.M5	BWR Feedwater Nozzle	Not applicable, VCSNS is a PWR.	N/A
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not applicable, VCSNS is a PWR.	N/A
XI.M7	BWR Stress Corrosion Cracking	Not applicable, VCSNS is a PWR.	N/A
XI.M8	BWR Penetrations	Not applicable, VCSNS is a PWR.	N/A
XI.M9	BWR Vessel Internals	Not applicable, VCSNS is a PWR.	N/A
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Surveillances	B.1.2

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.M11	Nickel-Alloy Nozzles and Penetrations	Alloy 600 Aging Management Program	B.1.1
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not credited for aging management.	N/A
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not credited for aging management.	N/A
XI.M14	Loose Part Monitoring	Not credited for aging management.	N/A
XI.M15	Neutron Noise Monitoring	Not credited for aging management.	N/A
XI.M16	PWR Vessel Internals	Reactor Vessel Internals Inspection	B.2.4
XI.M17	Flow-Accelerated Corrosion	Flow - Accelerated Corrosion Monitoring Program	B.1.6
XI.M18	Bolting Integrity	Not credited for aging management.	N/A
XI.M19	Steam Generator Tube Integrity	Steam Generator Management Program	B.1.10
XI.M20	Open-Cycle Cooling Water System	Service Water System Reliability and In-Service Testing Program	B.1.9
XI.M21	Closed-Cycle Cooling Water System	Not credited for aging management.	N/A
XI.M22	Boraflex Monitoring	Not credited for aging management.	N/A

Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Material Handling System Inspection Program	B.1.19
XI.M24	Compressed Air Monitoring	Not credited for aging management.	N/A
XI.M25	BWR Reactor Water Cleanup System	Not applicable, VCSNS is a PWR.	N/A
XI.M26	Fire Protection	Fire Protection Program	B.1.5
XI.M27	Fire Water System	Fire Protection Program	B.1.5
XI.M28	Buried Piping and Tanks Surveillance	Not credited for aging management.	N/A
XI.M29	Above ground Carbon Steel Tanks	Not credited for aging management.	N/A
XI.M30	Fuel Oil Chemistry	Chemistry Program	B.1.4
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance Program	B.1.24
XI.M32	One-Time Inspection	Above Ground Tank Inspection Diesel Generator Systems Inspection Liquid Waste System Inspection Reactor Building Cooling Unit Inspection Service Air System Inspection Small Bore Class 1 Piping Inspection Waste Gas System Inspection Heat Exchanger Inspections	B.2.1 B.2.2 B.2.3 B.2.5 B.2.6 B.2.7 B.2.8 B.2.12

**Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS**

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.M33	Selective Leaching of Materials	Fire Protection Program Heat Exchanger Inspections	B.1.5 B.2.12
XI.M34	Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection	B.2.10
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Non-EQ Insulated Cables and Connections Inspection Program	B.2.9
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Non-EQ Insulated Cables and Connections Inspection Program	B.2.9
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Not credited with aging management.	N/A
XI.S1	ASME Section XI, Subsection IWE	Containment ISI Program - IWE/IWL	B.1.16
XI.S2	ASME Section XI, Subsection IWL	Containment ISI Program - IWE/IWL	B.1.16
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI ISI Program – IWF	B.1.13
XI.S4	10 CFR 50, Appendix J	10 CFR 50 Appendix J General Visual Inspection 10 CFR 50 Appendix J Leak Rate Testing	B.1.11 B.1.12
XI.S5	Masonry Wall Program	Maintenance Rule Structures Program	B.1.18

**Table B-1:
CORRELATION OF NUREG 1801 AND VCSNS PROGRAMS**

GALL Program ID	NUREG 1801 Program Name	VCSNS Program Name	Appendix B Section
XI.S6	Structures Monitoring Program	Maintenance Rule Structures Program	B.1.18
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Service Water Pond Dam Inspection Program	B.1.21
XI.S8	Protective Coating Monitoring and Maintenance Program	Containment Coating Monitoring and Maintenance Program	B.1.15
N/A	None	Bottom-Mounted Instrumentation Inspection	B.1.3
N/A	None	Battery Rack Inspection	B.1.14
N/A	None	Flood Barrier Inspection	B.1.17
N/A	None	Pressure Door Inspection Program	B.1.20
N/A	None	Service Water Structures Survey Monitoring Program	B.1.22
N/A	None	Underwater Inspection Program (SWIS and SWPH)	B.1.23
N/A	None	Inspections for Mechanical Components	B.2.11
N/A	None	Preventive Maintenance Activities - Ventilation Systems Inspections	B.1.26
N/A	None	Preventive Maintenance Activities - Terry Turbine	B.1.25

B.1.0 EXISTING AGING MANAGEMENT ACTIVITIES

B.1.1 ALLOY 600 AGING MANAGEMENT PROGRAM

The Alloy 600 Aging Management Program is consistent with XI.M11 *Nickel - Alloy Nozzles and Penetrations*, as identified in NUREG-1801, the enhancements specified in the following table, and with the following clarification:

- Detection of Aging Effects: The Alloy 600 Aging Management Program will not rely on an enhanced leakage detection system for detection of small leaks caused by primary water stress corrosion cracking (PWSCC) during plant operation as suggested by XI.M11. Industry operating experience indicates that PWSCC cracks can be detected by means of inspecting for signs of boric acid leakage during outages and by monitoring primary coolant leakage per Technical Specifications during plant operation prior to the structural integrity of the pressure boundary being compromised.

The following enhancements will be incorporated into the Alloy 600 Aging Management Program prior to the period of extended operation.

NUREG-1801 Program	Attributes	Enhancements
XI.M11 Nickel - Alloy Nozzles and Penetrations	3. Parameters Monitored or Inspected 4. Detection of Aging Effects 5. Monitoring and Trending 6. Acceptance Criteria	Changes indicated by emerging regulatory requirements and developed by the industry groups will be implemented for the applicable Attributes.

B.1.1.1 Operating Experience

Recent industry inspection experience documented in NRC information notices confirms that Alloy 600 PWSCC cracks may initiate and grow through-wall in vessel head penetrations with susceptible materials. NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, addressed the occurrence of circumferential cracking of reactor pressure vessel head penetration (VHP) nozzles. The SCE&G response to Bulletin 2001-01 [**Reference B-3**] concluded that VCSNS falls into the NRC category of plants considered to have low susceptibility to PWSCC of the reactor pressure vessel (RPV) top head nozzles. The August 2001 VCSNS response stated that VCSNS performed VT-3 inspections of the interior surface of the reactor vessel head in April 1999 and found no recordable indications. NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, addressed further experience with PWSCC at Davis-Besse in February of 2002. The SCE&G 15-day response to Bulletin 2002-01 [**Reference B-4**] documented the types of inspections that are performed for primary coolant leakage. SCE&G stated that based on reviews of maintenance and repair history at VCSNS that there is no boric acid build up under the insulation on the RPV head surface. Inspections of the RPV head surfaces and VHP nozzles during the April 2002 refueling outage revealed no boric acid build up other than the trace amounts of boric acid residue left behind from a previously repaired instrument penetration (conoseal). Industry experience confirms that indications of Alloy 600 PWSCC crack formation by means of observed primary coolant leakage, and/or surveillance of boric acid residue in the vicinity of affected vessel head penetrations, provide adequate opportunity to detect Alloy 600 PWSCC before cracks reach critical length.

The NRC closure letter [**Reference B-14**] for SCE&G response to Generic Letter 97-01, Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations, requested that three issues be addressed in an application for license renewal of VCSNS. (1) As stated above, VCSNS falls into the NRC category of plants considered to have low susceptibility to PWSCC of the vessel head penetrations. (2) The vessel head penetrations are included within the scope of the boric acid corrosion inspection program, as confirmed by the SCE&G 15-day response to NRC Bulletin 2002-01. (3) The results of inspections completed on the vessel head penetrations are provided above, and is documented in the SCE&G response to NRC Bulletin 2001-01.

VCSNS discovered a crack on the 'A' hot leg nozzle at the beginning of Refuel Outage 12 (October 2000), when boric acid was found on the floor of the containment building. The crack in the nozzle was located in the weld between the RCS piping and the vessel nozzle, on the nozzle side of the weld. The RCS piping is SA-376, type 304 stainless steel. The vessel nozzle is SA-508 material clad with austenitic stainless steel. The weld used Inconel 182 butter (safe end) between the weld piece and the vessel nozzle. The VCSNS crack was the subject of NRC Information Notice 2000-17, Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V. C. Summer.

The investigation into the crack concluded that the cause was indirectly attributed to PWSCC. The weld at the nozzle was determined to be subjected to high tensile stresses as a result of extensive weld repairs performed during the original construction. A number of smaller PWSCC cracks were subsequently identified when the weld was cut out and destructively tested. A spool piece was used to replace the affected weld and was installed utilizing Inconel 52 and 152 weld materials, in effect removing the susceptible material. The welding was performed in a manner which minimized residual stresses. Further inspections of the other RCS nozzle safe end-to-pipe welds detected minor indications of cracking. An inspection of the B and C hot leg nozzles was performed in the April 2002 refueling outage. Results were provided in a letter from Stephen A. Byrne to the Document Control Desk (TAC No. MB3839) dated May 4, 2002. Future trending of these indications will be performed to assess the need for any further repair activities.

B.1.1.2 Conclusion

The Alloy 600 Aging Management Program has been demonstrated to be capable of detecting and managing cracking due to PWSCC prior to loss of component intended function based on indications of leakage. The Alloy 600 Aging Management Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.2 BORIC ACID CORROSION SURVEILLANCES

The Boric Acid Corrosion Surveillances are consistent with XI.M10, *Boric Acid Corrosion*, as identified in NUREG-1801. These surveillances additionally address boric acid corrosion of electrical connector contacts which may be exposed to borated water leakage mentioned in Chapter VI, Item A.2.1 of NUREG-1801.

B.1.2.1 Operating Experience

The Boric Acid Corrosion Surveillances were originally implemented as a result of NRC Generic Letter 88-05. In October 1999, a series of discussions involving plant personnel were initiated that resulted in procedure changes to coordinate inspection activities for boric acid leakage inside containment. The revised procedures coordinated the Generic Letter 88-05, health physics general area and ASME Section XI inspections.

The Boric Acid Corrosion Surveillances have been successful in managing loss of material due to boric acid induced corrosion. It has provided for timely identification of leakage and implementation of corrective actions. Since establishing the surveillances, there have been no instances of boric acid corrosion that have impacted components, structures, or systems from performing their intended functions. The following example illustrates the capability of the surveillances:

While performing visual inspections of the Reactor Building during RF-12 in October 2000, a significant quantity of boric acid deposits was discovered on the floor coming from the boot of the loop “A” RCS hot leg penetration at the bio-shield wall. Investigation revealed that the deposits originated from the leaking of reactor coolant at the welded joint between the reactor vessel nozzle and the loop “A” hot leg reactor coolant pipe.

As a result of this incident, the Boric Acid Corrosion Surveillances have been enhanced to ensure that all dissimilar metal welds are included in the population of components that are visually inspected at refueling outages or when appropriate plant conditions permit access.

The task sheets for the procedures credited for the Boric Acid Corrosion Surveillances were reviewed for the results of the past five years. The majority of the leaks were identified as small or showing signs of previous leakage. All leaks were documented, cleaned, visually examined, and evaluated by engineering for continued service, repair, or replacement. No significant loss of material has been found on leaking components or on adjacent structures or components in the area of any leak.

B.1.2.2 Conclusion

The Boric Acid Corrosion Surveillances have been demonstrated to be capable of identifying leaks from borated water systems, and subsequently managing the effects of boric acid corrosion. The Boric Acid Corrosion Surveillances provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.3 BOTTOM-MOUNTED INSTRUMENTATION INSPECTION

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The purpose of the Bottom Mounted Instrumentation Inspection is to identify loss of material due to fretting 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 (BMI) Inspection is a condition monitoring program.

(1) Scope - The Bottom Mounted Instrumentation Inspection is applicable to all BMI thimble tubes installed in the reactor vessel.

(2) Preventive Actions - No actions are taken as part of the Bottom Mounted Instrumentation Inspection to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored or Inspected - The Bottom Mounted Instrumentation Inspection monitors tube wall degradation (loss of material due to fretting) of the BMI thimble tubes. Failure of the thimble tubes would result in a breach of the Reactor Coolant pressure boundary.

(4) Detection of Aging Effects - In accordance with information provided in Monitoring and Trending below, the Bottom Mounted Instrumentation Inspection will detect loss of material due to fretting prior to loss of component intended function.

(5) Monitoring and Trending - Inspection of the BMI thimble tubes is performed using eddy current testing (ECT). 100% of the thimble tubes are inspected. The frequency of examination is based on an analysis of the data obtained using wear rate relationships predicted based on Westinghouse research. The ECT results are trended, wear rates are calculated, and inspections are planned prior to the refueling outage in which thimble tube wear is predicted to exceed the acceptance criteria. This ensures that the thimble tubes continue to perform their pressure boundary intended function.

(6) Acceptance Criteria - The acceptance criteria for the BMI thimble tubes is in the form of an “earliest projected date”. Using a wear rate formula, a calculation is performed to determine the earliest projected date for which wear on each BMI thimble tube wear exceed 75% loss of initial wall thickness.

(7) Corrective Actions - Thimble tubes must be capped or repositioned if projected through wall wear will exceed 75% prior to the next scheduled ECT. If measured or projected thimble tube wear exceeds 80%, then the thimble tube must be capped or replaced per the acceptance criteria. A condition evaluation report (CER) is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) Confirmation Process - Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) Administrative Controls – The Bottom Mounted Instrumentation Inspection is implemented in accordance with station procedures and work processes.

B.1.3.1 Operating Experience

Operating Experience - Flux thimble wear was first identified as an issue when three flux thimbles developed through wall leakage in a three month period at the Salem plant in 1981. Since that time, numerous plants have detected thimble wear in varying degrees. 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 because they serve as a portion of the reactor coolant system pressure boundary.

IE Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors” was issued in July 1988. The NRC requested the implementation of inspection programs for thimble tubes. Since the issuance of IE Bulletin 88-09, two inspections have been performed (RF-4 and RF-5) on thimble tubes at VCSNS. Several thimble tubes were repositioned in RF-5, but no thimble tubes have been capped or required replacement. Analysis of the wear rate data determined that ECT is not required on the thimble tubes again until RF-14 based on calculations performed in association with the inspections.

B.1.3.2 Conclusion

The Bottom Mounted Instrumentation Inspection has been demonstrated to be capable of detecting and managing loss of material in the thimble tubes. The Bottom Mounted Instrumentation Inspection provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.4 CHEMISTRY PROGRAM

The Chemistry Program is consistent with XI.M2, *Water Chemistry*, and the chemistry-related portions of XI.M30, *Fuel Oil Chemistry*, as identified in NUREG-1801 with the following clarifications:

- Detection of Aging Effects: The Chemistry Program is a mitigation program and no aging effects are detected as part of this program. The plant operating experience provides confirmation of the effectiveness of the program for managing aging during the period of extended operation. Based on this experience, VCSNS does not commit to performing one-time inspections to verify the effectiveness of the Chemistry Program as suggested by NUREG-1801 under XI.M2.

B.1.4.1 Operating Experience

The VCSNS Chemistry Program is an ongoing program that incorporates the best practices of industry organizations, vendors, utilities, and water treatment experts. The program provides assurance that the fluid environment to which the surfaces of components are exposed will minimize corrosion. The Chemistry Program incorporates EPRI and Institute of Nuclear Power Operations (INPO) guideline documents as well as the “lessons learned” from South Carolina Electric and Gas (SCE&G) and external industry operating experience. The program has been subject to periodic internal and external assessment activities that help to maintain highly effective chemistry control, and facilitate continuous improvement. The overall effectiveness of the chemistry program is supported by the excellent operating experience for systems, structures, and components that are influenced by the program.

A review of operating experience did not reveal a loss of component intended function of components that are exposed to borated water, closed cooling water, or treated water that could be attributed to an inadequacy of the Chemistry Program. This operating experience confirms the effectiveness of the Chemistry Program for borated, closed cooling (treated), and treated water to manage aging effects when continued into the period of extended operations.

Analyzing and trending the water chemistry specifications has been in effect since the initial implementation of the facility operating license at VCSNS and is considered acceptable based on industry operating experience. A review of the Chemistry Program confirms the reasonableness and acceptability of the sampling frequency.

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 an inadequacy of the Chemistry Program. There have been no readings found out of specification during testing of the fuel oil in the storage tanks for the Class 1E Diesel Generators. This can be credited to the fact that fuel delivered to the site is sampled/analyzed before any additions are made to the tanks.

B.1.4.2 Conclusion

The Chemistry Program has been demonstrated to be capable of managing loss of material, cracking, and fouling of components exposed to borated water, closed cooling water, treated water, or fuel oil environments. The Chemistry Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.5 FIRE PROTECTION PROGRAM

The Fire Protection Program is consistent with XI.M26, *Fire Protection*, and XI.M27, *Fire Water System*, as well as XI.M33, *Selective Leaching of Materials*, as identified in NUREG - 1801 with of the enhancements specified in the following table and with the following clarifications:

- Parameters Monitored/Inspected: VCSNS fire rated door inspections monitor holes or breaks in the door surface at a frequency of every 6 months rather than the bimonthly frequency recommended. Based on VCSNS and industry operating experience the 6 month inspection frequency provides reasonable assurance that degradation of a door is detected prior to loss of function. [XI.M26]
- Parameters Monitored/Inspected: Aging management of the fuel supply line for the diesel-driven fire pump at VCSNS is credited to the Chemistry Program and is not managed by the Fire Protection Program. [XI.M26]
- Parameters Monitored/Inspected: VCSNS maintains proper clearances (gap) between door, frame, and threshold in accordance with station procedures. However, VCSNS does not consider maintaining the clearances to be an aging effect for license renewal. [XI.M26]
- Detection of Aging Effects: VCSNS intends to perform ultrasonic testing of selected fire protection piping to detect aging effects in lieu of disassembly of fire protection piping for inspection or full-flow testing of stagnant portions of fire protection piping. (See program enhancements.) [XI.M27]

The following enhancements will be incorporated into the Fire Protection Program prior to the period of extended operations.

NUREG-1801 Program	Attributes	Enhancements
XI.M27, Fire Water System	4. Detection of Aging Effects	Sprinklers will either be replaced or representative samples will be submitted to a recognized laboratory for field service testing in accordance with NFPA code 25. Subsequent replacement or field service testing of representative samples will occur at 10-year intervals. Ultrasonic testing of representative portions of above ground fire protection piping that are exposed to water but do not normally experience flow will be performed before the end of the current operating term. Ultrasonic testing will occur at 10-year intervals thereafter.
XI.M33, Selective Leaching of Materials	4. Detection of Aging Effects	A new one-time inspection, to be performed just prior to the end of the current operating term, will be added to the Fire Protection Program. The inspection will include a Brinnell Hardness Test or equivalent test in order to detect and characterize a reduction of material hardness (loss of material) due to selective leaching for a representative sample of susceptible brass and cast iron components in the Fire Service System.

B.1.5.1 Operating Experience

B.1.5.1.1 Mechanical

Monthly surveillances are conducted on the fire protection system consisting of flow tests and pump start tests. Flow tests and flushes of the main distribution loops have been conducted to ensure functionality and have all met acceptance criteria with the exception described below. Working pressure and flow pressure are measured during these tests. This will indicate fouling to an unacceptable level and hence manage this aging effect. Fire hydrants and sprinklers are visually inspected for aging effects. This visual inspection looks for painted, corroded, damaged, or dirty sprinkler heads, obstruction of sprinkler heads, and proper orientation of sprinkler heads. The fire hydrants are inspected for corrosion on the exterior surface that might impede operation and standing water in the hydrant barrel that might indicate valve leakage or fouling.

A non-conformance notice (NCN) was generated in January of 1994 in association with low flow during flow testing of the main distribution loop. As part of the resolution the piping was hydrolazed to remove accumulated deposits. Additionally, engineering evaluation determined that a reduction and redistribution of sprinkler heads was permissible and would restore the required pressure at the sprinkler heads to ensure full spray pattern. The results of flow testing of the fire protection piping since this occurrence have been acceptable.

B.1.5.1.2 Fire Barriers And Fire Barriers Penetration Seals

Fire barrier and fire barrier penetration seal inspections in the past five years do not indicate any fire barrier or fire barrier penetration seal that is in non-conformance with the acceptance criteria.

NRC Inspection Report 50-395 / 98-01 [**Reference B-5**] concludes that the surveillance procedures are excellent and satisfy the requirements of Generic Letter 86-10 “Implementation of Fire Protection Requirements”.

A review of past licensee event reports (LERs) and NRC Inspection Reports associated with fire barriers indicated design deficiencies such as lack of qualifying documentation, or installation deficiencies such as not sealing core drills or damaging Kaowool wrap and seals during maintenance. Other LERs indicate fire watches not initiated after a fire barrier is inoperable, and effects of electrical storms on the fire protection system. Based on this review, none of the LERs or NRC Inspection Reports are related to aging.

Non-conforming conditions were noted during surveillance of fire barrier penetration seals that were aging related cracks and separations. Conditions were repaired in accordance with station procedures.

No condition evaluation reports (CERs) were initiated for fire barriers or fire barrier penetration seals relevant to aging.

B.1.5.1.3 Fire Doors

VCSNS has no failures or adverse trends for fire doors. Surveillance inspections in the last five years have not identified any non-conformance relative to the acceptance criteria.

Frequency of inspections performed since the initial implementation of the Technical Specifications requirements is considered acceptable based on industry operating experience. Industry experience indicates that degradation of a door will be detected prior to loss of function.

LERs for fire doors were compiled using the licensing database. An LER was generated for a combination of fire/pressure door, but not the fire barrier function, and reported a design deficiency which is not related to aging. Two LERs were identified for missed weekly surveillances. Two LERs were identified for fire doors not fully closed.

No non-conformance notices (NCNs) or condition evaluation reports (CERs) were initiated for fire doors relevant to aging.

B.1.5.2 Conclusion

The Fire Protection Program has been demonstrated to be capable of detecting and managing aging effects for the fire water system, for fire barriers and fire barrier penetrations seals, and for fire doors. The Fire Protection Plan provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.6 FLOW-ACCELERATED CORROSION MONITORING PROGRAM

The Flow-Accelerated Corrosion Monitoring Program is consistent with XI.M17, *Flow-Accelerated Corrosion*, as identified in NUREG-1801.

B.1.6.1 Operating Experience

After industry experience indicating that feedwater heaters may be subject to flow-accelerated corrosion (FAC), VCSNS conducted an inspection of feedwater heaters that revealed some degradation of the pressure boundary. A repair of the subject feedwater heater was accomplished in accordance with the requirements of the applicable code. Some degradation of the feedwater piping was found downstream of the feedwater regulating valves and the piping was replaced. The need for inspections is determined by a calculation performed in accordance with engineering procedures. If components exhibit high wear during a cycle they are replaced with more FAC-resistant material. The change to MPA (Methoxypropylamine) chemistry control after RF-9 (1996) has aided in the control of FAC at the station. Refueling summary reports for each refuel outage since RF-8 in 1994 were examined. The results demonstrate a mature well functioning FAC program at VCSNS.

B.1.6.2 Conclusion

The Flow-Accelerated Corrosion Monitoring Program has been demonstrated to be capable of detecting and managing loss of material for components susceptible to flow-accelerated corrosion. The Flow - Accelerated Corrosion Monitoring Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.7 IN-SERVICE INSPECTION (ISI) PLAN

The In-Service Inspection (ISI) Plan is consistent with XI.M1, *ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD*, as identified in NUREG-1801 with the following clarification:

- VCSNS is committed to the 1989 Edition of ASME Section XI with no addenda for the second ten-year inspection interval. VCSNS has adopted the 1995 Edition of ASME Section XI with 1996 Addenda for ultrasonic examination requirements.

B.1.7.1 Operating Experience

VCSNS has operated since October 1982, and has performed inservice inspections in accordance with relevant portions of approved editions of ASME Code Section XI throughout that period. Two specific examples of VCSNS operating experience in which the In-Service Inspection (ISI) Plan (including repair and replacement) played a role follow.

In the Reactor Coolant System, primary water stress corrosion cracking contributed to leakage that developed at the reactor vessel “A” hot leg nozzle (discovered in 2000 while Refueling Outage 12). This leakage was detected by virtue of boric acid residue, and confirmed by volumetric examination. The crack was inspected, evaluated and repaired in accordance with ASME Section XI criteria.

In the Steam Generators, inservice inspections of tubes are performed in accordance with station surveillance procedures. Degradation of steam generator tubes was noted during the first ten-year inservice inspection interval. Indications of magnetite formation and tube denting were noted as early as 1990, and damping cables were installed to reduce high-cycle vibration of steam generator tubes during RF-5. Subsequently, the steam generators were replaced in 1994.

B.1.7.2 Conclusion

The In-Service Inspection (ISI) Plan has been demonstrated to be capable of detecting and managing aging effects of ASME code components in the Reactor Coolant System. The In-Service Inspection (ISI) Plan provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.8 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program is consistent with XI.M3, *Reactor Head Closure Studs*, as identified in NUREG-1801.

B.1.8.1 Operating Experience

The current aging management program for reactor head closure stud bolting is largely dependent upon the In-Service Inspection (ISI) Plan. That program has provisions regarding inspection techniques and evaluations, and other aspects relevant to monitoring the condition of reactor head closure stud bolting. The ASME Code has been shown to be effective in managing aging effects in Class 1 components, including the reactor closure head stud bolting.

VCSNS has operated since October 1982, and has performed inservice inspections in accordance with relevant portions of approved editions of ASME Code Section XI throughout that period. During this period of time no damage to the reactor head closure stud bolting materials has been detected.

B.1.8.2 Conclusion

The Reactor Head Closure Studs Program has been demonstrated to be capable of detecting and managing loss of mechanical closure integrity for the closure stud bolting. The Reactor Head Closure Studs Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.9 SERVICE WATER SYSTEM RELIABILITY AND IN-SERVICE TESTING PROGRAM

The Service Water System Reliability and In-Service Testing Program is consistent with XI.M20 *Open-Cycle Cooling Water System*, as identified in NUREG-1801.

B.1.9.1 Operating Experience

The application of measured corrosion rates has been demonstrated to provide adequate information on the rate of loss of material to predict when replacement of components might be necessary. Operating experience demonstrates that performance testing on raw water heat exchangers provides adequate predictive modeling for fouling of heat transfer surfaces to prevent loss of intended function.

Based on operating experience, the Service Water System Reliability and In-Service Testing Program is capable of detecting pin hole leaks in the Service Water System prior to loss of function. A combination of chemicals are injected into the system to form protective coatings on the internal surface of the piping, induce gradual flaking off of tubercles, and to act as a silt dispersant. VCSNS has adjusted the rate of addition of corrosion inhibitor, deposit control, and silt dispersant chemicals based on industry experience.

B.1.9.2 Conclusion

The Service Water System Reliability and In-Service Testing Program has been demonstrated to be capable of managing the effects of aging for components in raw water environments. The Service Water System Reliability and In-Service Testing Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.10 STEAM GENERATOR MANAGEMENT PROGRAM

The Steam Generator Management Program is consistent with XI.M19 *Steam Generator Tube Integrity,* as identified in NUREG-1801.

B.1.10.1 Operating Experience

Industry operating experience in the mid-1970's showed that there was a high rate of tube plugging with the dominant damage caused by tube wastage. Chemistry controls were adjusted to correct the tube wastage but led to conditions conducive to corrosion of the carbon steel support plates, which led to tubing deformation as a result of denting and cracking and an unacceptable rate of tube plugging. The industry, working through EPRI, implemented steam generator programs with aggressive improvements in control of secondary-side water chemistry and upgrades in secondary-side equipment, thus essentially eliminating wastage and denting. The industry incorporated these successful programmatic strategies in the EPRI Secondary Water Chemistry Guidelines and associated supporting documents. VCSNS meets the industry guidelines for a steam generator program and secondary water chemistry;.

All three steam generators were replaced at VCSNS during RF-8 (1994). Since then, at the recommendation of the vendor, four tubes have been plugged because they were not expanded into the tube sheet during manufacturing. A partial eddy current inspection (Steam Generators A and B) and moisture carryover modification was conducted during RF-9. Partial eddy current inspections were conducted during RF-10 (Steam Generator C) and RF-11 (Steam Generators A and B). A 100% eddy current inspection of Steam Generators A, B, and C was conducted during RF-12. Also, during RF-12, sampling at top tube sheet (TTS) and low row U bends, a full secondary side inspection, and sludge lancing of tube sheets were performed.

No significant degradation was found during these inspections.

B.1.10.2 Conclusion

The Steam Generator Management Program has been demonstrated to be capable of detecting and managing the effects of aging for the steam generator tubes and tube plugs. The Steam Generator Management Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.11 10 CFR 50 APPENDIX J GENERAL VISUAL INSPECTION

The 10 CFR 50 Appendix J General Visual Inspection is consistent with XI.S4, *10 CFR 50 Appendix J*, as identified in NUREG-1801.

B.1.11.1 Operating Experience

The most recent Type A ILRT was completed in March 1993 during RF-8, with a general visual structural examination of the containment system also implemented during RF-8. General visual structural examinations of the containment system were also implemented during RF-10 and RF-12 to satisfy the additional requirements for general visual structural examination of the containment system.

No licensee event reports (LERs) were initiated subsequent to any general visual structural examination of the containment system. There were no non-conformances (NCNs) or condition evaluation reports (CERs) identified that resulted from conditions related to aging mechanisms. NRC Inspection Report 50-395 / 93-09 [**Reference B-6**] reviewed the station surveillance procedure and concluded that the procedure is acceptable to implement the general containment visual inspection prior to a Type A ILRT.

B.1.11.2 Conclusion

The 10 CFR 50 Appendix J General Visual Inspection has been demonstrated to be capable of managing the effects of aging for the containment liner, associated moisture barriers, and the Reactor Building structure. The 10 CFR 50 Appendix J General Visual Inspection provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.12 10 CFR 50 APPENDIX J LEAK RATE TESTING

The 10 CFR 50 Appendix J Leak Rate Testing is consistent with XI.S4, *10 CFR 50 Appendix J,* as identified in NUREG-1801.

B.1.12.1 Operating Experience

NRC Inspection Report 50-395 / 93-09 [**Reference B-6**] concluded that the station surveillance procedure provides proper guidance and satisfies regulatory requirements to perform a Type A ILRT.

Over three refueling cycles (most recently RF-10, RF-11, and RF-12), Type B penetrations delineated in the station surveillance procedure were tested with satisfactory results.

A non-conformance (NCN) was documented for rust found on the Reactor Building liner plate adjacent to the moisture barrier and a degraded moisture barrier. The disposition was to clean-up the rust on the Reactor Building liner plate adjacent to the moisture barrier and to replace affected portions of the moisture barrier. Visual examination and ultrasonic tests demonstrated that the liner plate had not degraded. The evaluation concluded that the condition was normal surface life exposure and was not aging related.

B.1.12.2 Conclusion

The 10 CFR 50 Appendix J Leak Rate Testing Program has been demonstrated to be capable of detecting and managing the effects of aging for the components forming the containment pressure boundary. The 10 CFR 50 Appendix J Leak Rate Testing Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.13 ASME SECTION XI ISI PROGRAM - IWF

The ASME Section XI ISI Program - IWF is consistent with XI.S3, *ASME Section XI, Subsection IWF*, as identified in NUREG-1801 with the following clarification:

- VCSNS uses 1989 edition of ASME Section XI with no Addenda

B.1.13.1 Operating Experience

A review of ASME Class 1, 2, and 3 component support inspections for the past five years identified one case where acceptance criteria was not met. The gap at the top of a pipe support exceeded the acceptance criteria, however this is not aging related. Operating experience at other nuclear facilities shows that improperly heat-treated anchor bolts are susceptible to stress corrosion cracking. At VCSNS, ASTM A490 anchor bolt material is properly heat-treated by conforming to ASTM Specification A490 through a Certified Material Test Report in accordance with station specifications. Section 3.3.1.1 of the Safety Evaluation Report for WCAP-14422 [**Reference B-7**] states “In the absence of a high level of sustained tensile stress, stress corrosion cracking is not likely to occur”. ASTM A490 anchor bolts are not highly pre-loaded at VCSNS and do not have a high level of sustained tensile loads due to lower LOCA applied loads as a result of the elimination of the dynamic effects of postulated High Energy Line Break (HELB) of the Reactor Coolant System - Primary Coolant Piping. Therefore stress corrosion cracking is not a significant aging effect for ASTM A490 anchor bolts for major equipment supports. Class 1 pipe supports use Hilti Kwik bolts as anchors. Hilti Kwik bolts are not susceptible to stress corrosion cracking since the tensile strength is specified to be 125,000 psi, which is below the threshold tensile strength level of 150,000 psi where stress corrosion cracking is a concern.

IWF sampling inspections are effective in managing aging effects for ASME Class 1, 2, and 3 supports. There is reasonable assurance that the IWF inspection program will be effective through the period of extended operation.

Two non-conformance notices (NCNs) were identified for instances of minor surface corrosion on supports and anchor bolting. The intended functions were not affected and corrective actions were performed in accordance with site procedures.

No condition evaluation reports (CERs) were initiated subsequent to ASME Class 1, 2, and 3 component support inspections.

B.1.13.2 Conclusion

The ASME Section XI ISI Program - IWF has been demonstrated to be capable of detecting and managing the effects of aging for ASME code supports. The ASME Section XI ISI Program - IWF for Class 1, 2, and 3 component supports, and support anchorage provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.14 BATTERY RACK INSPECTION

There is no NUREG-1801 item addressing this program. This is a plant specific program.

(1) Scope - The scope of the Battery Rack Inspection includes the battery racks for the following systems:

- Electrical DC (ED) System (Vital Batteries)
- Fire Service (FS) System (Diesel Fire Service Pump Battery)

The regulatory basis for inspecting battery racks for the ED System is found in the VCSNS Technical Specifications Surveillance Requirement 3.8.2.1, while the regulatory basis for inspecting battery racks for the FS System is the commitment in the fire protection procedure.

(2) Preventive Actions - No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The Battery Rack Inspection is a conditioning monitoring program.

(3) Parameters Monitored or Inspected - The ED System and FS System battery racks specific examination guidelines are provided in IEEE-450. For the ED System and FS System, battery racks are inspected for loss of material due to corrosion. Although not credited for license renewal, the battery racks are also inspected for physical damage.

(4) Detection of Aging Effects - The Battery Rack Inspection Program detects structural damage or degradation (including loss of material due to corrosion) prior to loss of structure intended function.

(5) Monitoring and Trending - For the ED System, a visual examination is performed every 18 months in accordance with commitments in **FSAR Section 8.3.2.2.2** and Technical Specifications Surveillance Requirement 4.8.2.1.c.

For the FS System, visual examination is performed every 18 months in accordance with a commitment in the fire protection procedure.

Results of 18 month battery rack inspections are retained in sufficient detail to permit adequate confirmation of the inspection program. In particular these records identify inspectors, results of the inspections, note discrepancies with the cause, and prescribe corrective action. No actions are taken as part of this program to trend inspection or test results.

(6) Acceptance Criteria - For the ED System, the acceptance criterion is “no visual indication of loss of material due to corrosion” as stated in a surveillance test procedure. For the FS System, acceptance criterion is “no visual indication of loss of material due to corrosion” as stated in a surveillance test procedure.

(7) Corrective Actions - For the ED and FS Systems, the surveillance procedure provides guidance when abnormalities are observed. Repair / replacement of unacceptable batteries or racks is in accordance with an electrical maintenance procedure. A condition evaluation report (CER) is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) Confirmation Process - Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) Administrative Controls - For the ED and FS Systems, visual examination for structural integrity of racks, that includes an attribute for loss of material due to corrosion, is implemented in accordance with a surveillance procedure. Visual examination of the ED System is performed to comply with commitments in **FSAR Section 8.3.2.2.2** and Technical Specifications Surveillance Requirement 4.8.2.1.c. Visual examination of the FS System is performed to comply with the commitment in the fire protection procedure.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedures.

B.1.14.1 Operating Experience

Visual inspection of the battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade performance. The presence of physical damage or deterioration does not necessarily represent a failure, provided an evaluation determines that the physical damage or deterioration does not affect the ability of the battery rack to perform its function. Review of work orders for the past five years did not identify any instance where abnormal deterioration of battery racks occurred.

Licensee event reports (LERs) associated with batteries were reviewed. LERs document missed weekly surveillances or electrical test deficiencies that were not aging related. Inspections were performed punctually and were satisfactory. No non-conformance notices

(NCNs) or condition evaluation reports (CERs) were initiated subsequent to inspections for battery racks.

B.1.14.2 Conclusion

The Battery Rack Inspection Program has been demonstrated to be capable of managing loss of material for steel battery racks. The Battery Rack Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.15 CONTAINMENT COATING MONITORING AND MAINTENANCE PROGRAM

The Containment Coating Monitoring and Maintenance Program is consistent with XI.S8, *Protective Coating Monitoring and Maintenance,* as identified in NUREG-1801with the following clarifications:

- Scope of Program: The VCSNS aging management program is based on RG 1.54 Revision 0 which is acceptable per the XI.S8 program description discussion.
- Scope of Program: For the Westinghouse scope of supply (NSSS components and equipment) an alternative methodology for meeting the requirements of RG 1.54 was employed and was accepted by the NRC, as documented in Section 3A of the VCSNS FSAR.

B.1.15.1 Operating Experience

The ASME Section XI, Subsections IWE and IWL inspections conducted in 2000 are considered as the baseline examination. All previous inspections conducted for other programs (e.g., Maintenance Rule and Appendix J) had not identified any areas inside containment with surface areas likely to experience accelerated degradation or aging. Therefore, there were no areas designated for augmented examination prior to this baseline inspection.

The IWE inspection of the containment liner conducted during the 2000 Containment Inservice Inspection, revealed several areas of containment liner coating degradation. Minor flaking and/or split separation of the liner top coat in the vicinity of the spray rings was noted. One indication of a small split in the top coat (approximately 6 inches long) and several indications of partial delamination (flaking) of the top coat were observed with no exposure of the primer coat. These conditions were documented in the non-conformance (NCN) program.

The IWE inspection conditions and other areas of containment coating degradation were also documented in the report for maintenance rule inspections performed in 2000. None of the degraded conditions have an immediate adverse effect on the ability of the coatings to perform their intended function(s). Requirements were established for engineering to evaluate the identified conditions for corrective actions. These conditions were documented in the condition evaluation report (CER) program.

In accordance with the dispositions of the NCN and CER documents, most of the areas with identified conditions were reworked per appropriate civil maintenance procedures during RF-12. Conditions that have not been repaired are considered minimal at this time and the coating is judged to be capable of performing its intended function of protection of the liner without significant failures. These areas will be monitored by periodic (each outage) civil maintenance walk-downs and by augmented ASME Section XI IWE inspections for any changes in conditions that may suggest a loss of integrity or function. Augmented inspections were conducted during the April 2002 refueling outage with no observable changes in the condition of the identified areas.

Additionally, GSI-191, Assessment of Debris Accumulation on PWR Sump Performance, addresses operating experience which identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings, that could potentially impact the performance of the emergency core cooling system. However, degradation of coatings is an issue under the CLB and is not specifically related to the 40-year term of the current operating license, and therefore is not a TLAA. As such, the issue is not specifically a license renewal concern, but has been accounted for during the period of extended operation as described below.

Generic Letter 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, describes the industry experience pertaining to coatings degradation inside containment and the consequential clogging of sump strainers. Monitoring and maintenance of service Level I coatings (conducted in accordance with RG 1.54) are effective programs for managing degradation, and therefore an effective means to manage loss of material due to corrosion of carbon steel inside containment.

The SCE&G response to Generic Letter 98-04 [**Reference B-8**] states that the plant has implemented controls for the procurement, application, and maintenance of service Level I protective coatings used inside the containment in a manner that is consistent with the licensing basis and regulatory requirements.

B.1.15.2 Conclusion

The Containment Coating Monitoring and Maintenance Program has been demonstrated to be capable of maintaining the integrity of the protective coatings inside the Reactor Building. The Containment Coating Monitoring and Maintenance Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.16 CONTAINMENT ISI PROGRAM - IWE/IWL

The Containment ISI Program - IWE/IWL is consistent with XI.S1 *ASME Section XI, Subsection IWE* and XI.S2 *ASME Section XI, Subsection IWL,* as identified in NUREG-1801with the following clarification:

- VCSNS uses the 1992 Edition of ASME XI with 1992 Addenda

B.1.16.1 Operating Experience

Examinations for the first period of first interval were performed during RF-12 with satisfactory results. There were no licensee event reports (LERs) based on these examinations.

Non-conformance notices (NCNs) and/or condition evaluation reports (CERs) were originated and dispositioned for the following conditions identified during these examinations:

- Containment Liner Coating Degradation (NCN) - Several areas of top coat were identified as degraded; however, the primer coat was intact with no signs of deterioration. The affected areas were cleaned and re-coated. Two areas of top coat in the dome were identified with initial signs of degradation. These areas have been identified for augmented inspections during future refueling outages.
- RHR and Spray Guard Pipe (CER) - Groundwater leakage identified at penetrations in the Auxiliary Building resulted in degradation (corrosion) of guard pipes. Subsequent evaluations determined that the guard pipe wall thickness remained acceptable. These areas have been identified for augmented inspections during future refueling outages.
- Concrete Leaching (CER) - Concrete leaching in the Tendon Access Gallery has been attributed to groundwater seepage through cracks and construction joints within the surrounding fill concrete. One specific location was also identified with a minor corrosion build-up on the outer wall. Chemical analysis has determined that the groundwater is not aggressive. These areas have been identified for augmented inspections during future refueling outages.
- Moisture Barrier (CER) - Minor cracking and separation of the moisture barrier was identified at a few locations. These areas were repaired an/or replaced.

Augmented inspections were conducted during the April 2002 refueling outage for the above conditions. Additional CERs were originated for follow-up repair and/or replacement.

B.1.16.2 Conclusion

The Containment ISI Program - IWE/IWL has been demonstrated to be capable of detecting and managing the effects of aging for the liner, associated moisture barriers, and the Reactor Building structure. The Containment ISI Program - IWE/IWL provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.17 FLOOD BARRIER INSPECTION

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The VCSNS Flood Barrier Inspection Program is identified for completeness since it contains individual components that have the unique function of mitigating the effects of internal flooding. All flood barrier components are managed by either the Fire Protection Program or Maintenance Rule Structures Program.

(1) Scope – Nuclear safety-related flood barriers are credited with mitigating the effects of internal flood. Nuclear safety-related flood barriers include curbs at entrances to cubicles housing safety grade equipment as stated in **FSAR Section 6.3.2.2.7**.

Designated flood doors (watertight doors) are identified in plant specifications and are listed on architectural drawings. Ten doors are designated as flood doors (watertight doors).

Penetrations requiring Nuclear Safety Related flood seals are specified in design basis documents. Penetrations requiring Nuclear Safety Related flood seals are shown on engineering drawings for the Intermediate Building, the Control Building, and the Diesel Generator Building. Eleven penetrations require Nuclear Safety Related flood seals.

(2) Preventive Actions – No actions are taken as part of this program to prevent aging effects or mitigate aging degradation. The Flood Barrier Inspection is a condition monitoring program.

(3) Parameters Monitored or Inspected – Aging effects for flood barriers are cracks, exposed reinforcing steel, corrosion, scaling, popouts, surface pitting, and spalling. Aging effects are listed in an engineering procedure. Aging effects are the same for Nuclear Safety Related or Quality Related flood barriers.

The aging effects for flood barrier penetration seals are similar to aging effects for fire barrier penetration seals and include cracking, fraying, separation from penetration, and through-wall holes.

(4) Detection of Aging Effects – The Flood Barrier Inspection Program detects aging effects prior to loss of intended function.

(5) Monitoring and Trending – Visual examination of concrete structures is performed as stated in an engineering procedure. Visual examination of the flood barrier penetration seals that are also fire barrier penetration seals is performed as stated in Surveillance Test Procedures. Visual examination of Nuclear Safety Related flood barrier penetration seals that are also fire barrier penetration seals is performed once every 18 months as stated in Surveillance Test Procedures. No actions are taken as part of this program to trend inspection or test results.

(6) **Acceptance Criteria** – Flood barrier and flood barrier penetration seal examination acceptance criteria are provided in an engineering services procedure for flood barriers that are not fire barriers. Acceptance criteria are no cracks, no exposed reinforcing steel, no corrosion, no scaling, no popouts, no surface pitting, and no spalling. Acceptance criteria are the same for Nuclear Safety Related or Quality Related flood barriers.

Flood barrier penetration seal examination acceptance criteria are the same as for fire barrier penetration seals and are provided in a technical requirements package. Acceptance criteria are provided for indication of cracking, separation between surfaces at penetration, and no through-wall holes.

(7) **Corrective Actions** – Maintenance work requests are initiated to repair abnormalities. The non-conformance (NCN) process is initiated for flood barrier penetration seals that do not meet the acceptance criteria in compliance with the Fire Protection Evaluation Report. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls** – The flood barriers inspections are implemented through an engineering procedure and surveillance test procedures described above. Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with Station Administrative Procedure.

B.1.17.1 Operating Experience

Since the majority of flood barriers are also fire barriers, inspection attributes delineated in surveillance test procedures are the same for fire barriers and flood barriers and thus satisfy the same acceptance criteria. Therefore if fire barrier and fire barrier penetration seal inspections are satisfactory, then flood barrier and flood barrier penetration seal inspections are satisfactory. All flood doors are also fire doors. Fire door inspection attributes are the same as flood door inspection attributes. Flood doors are adequate if the fire door inspections are adequate.

No licensee event reports (LERs), non-conformance notices (NCNs) or condition evaluation reports (CERs) were initiated for flood barriers (walls, curbs, equipment pedestals), flood doors, and flood barrier penetration seals relevant to aging.

B.1.17.2 Conclusion

The Flood Barrier Inspection Program has been demonstrated to be capable of detecting and managing the effects of aging for flood barrier components. The Flood Barrier Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.18 MAINTENANCE RULE STRUCTURES PROGRAM

The Maintenance Rule Structures Program is consistent with XI.S6, *Structures Monitoring Program*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Maintenance Rule Structures Program prior to the period of extended operation:

NUREG-1801 Program	Attributes	Enhancement
XI.S6, Structures Monitoring Program	1.) Scope	Future inspections will add: North Berm, Electrical Manhole EMH-2 interior inspection, Inaccessible Areas when exposed by excavation, Flood Barrier Seals for Control and Diesel Generator Buildings, Portions of the power path from the power circuit breaker (PCB) in the substation to the safety-related buses, and Groundwater chemical analyses.
	3.) Parameters Monitored / Inspected	Groundwater chemical analyses will include: pH, Sulfates and Chlorides.
	5.) Monitoring and Trending	Groundwater chemical analyses will be used to monitor changes in aggressiveness of the below grade environment.

B.1.18.1 Operating Experience

An initial baseline position was established at VCSNS for the acceptability of the maintenance rule structures to remain capable of providing their maintenance rule functions over the life of the plant in accordance with the Maintenance Rule. This baseline position documented numerous periodic inspections and surveillances that were performed on certain maintenance rule structures in accordance with existing regulatory or licensing commitments at VCSNS during the period of 1993 to 1996.

The 1996 baseline assessment concluded that the maintenance rule structures and structural components were acceptable and were free of deficiencies or degradation, which could lead to possible failure. Therefore, these structures were determined to be capable of performing their structural functions, including the protection and support of 0 systems and components.

The maintenance rule inspection report completed in 2000 noted that in general, most of the maintenance rule structures and structural components were evaluated to be “acceptable” with regards to continued function. However, nine items/areas were identified as “Acceptable with Deficiencies” that exhibited a trend of aging. These conditions mostly deal with rust/corrosion due to weathering, water in-leakage and ponding. None of the conditions have an immediate adverse effect on the ability of the structures or components to perform their intended function(s). These items were entered into the plant corrective action program for resolution. The next inspection is scheduled in 2005.

B.1.18.2 Conclusion

The Maintenance Rule Structures Program has been demonstrated to be capable of detecting and managing the effects of aging for structures and structural components. The Maintenance Rule Structures Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.19 MATERIAL HANDLING SYSTEM INSPECTION PROGRAM

The Material Handling System Inspection Program is consistent with XI.M23, *Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems*, for the bridge and trolley structural members and rails as identified in NUREG-1801 with the following clarifications:

- **Scope:** The polar crane girders and brackets are addressed under the Maintenance Rule Structures Program.
- **Scope:** Wear on the crane rails has been determined to not require aging management for the VCSNS cranes.
- **Parameters Monitored and Inspected:** The number and magnitude of lifts made by the cranes are addressed as a TLAA under **Section 4.7.2** of this Application.

B.1.19.1 Operating Experience

Through monitoring effectiveness of maintenance at nuclear power plants there has been no corrosion-related degradation that has impaired cranes. Cranes have not operated beyond their design lifetime so there are no significant fatigue-related structural failures.

NRC Inspection Report 50-395 / 82-28 [**Reference B-9**] documents review and approval of the VCSNS material handling system maintenance procedures and concludes that the procedures comply with NUREG-0612 requirements.

Prior to 1996, a condition was identified of overstressed bolted connections in the trolley of the polar crane. The vendor (Whiting) identified this non-conformance as a design deficiency. The overstressed bolts were replaced with high strength ASTM A325 bolts during refueling outage RF-7.

While evaluating the polar crane for handling the replacement steam generators, the vendor (Whiting) identified a design deficiency on overstressed areas of the trolley and bridge girders. Modifications were completed prior to refueling outage RF-8.

Industry operating experience for the Material Handling System was reviewed in association with the maintenance rule. A total of 15 events were identified of which eight are relevant to VCSNS. VCSNS concluded that system design and operating procedures would anticipate these events.

In 2000 a non-conformance (NCN) was identified for catastrophic failure of the spent fuel bridge crane roller guide bearing due to age related stress corrosion cracking. The probable cause for catastrophic failure was determined to be a long period of inactivity in a humid environment. The NCN evaluation determined that this condition does not affect structural integrity or function of the spent fuel bridge crane. Therefore, this condition is not an aging effect since the intended function of the crane is maintained.

B.1.19.2 Conclusion

The Material Handling System Inspection Program has been demonstrated to be capable of managing loss of material for crane rails, rail supports, and structural supports. The Material Handling System Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.20 PRESSURE DOOR INSPECTION PROGRAM

There is no NUREG-1801 item addressing this program. This is a plant specific program.

Pressure doors at VCSNS are used to separate critical equipment from high energy pipe breaks; and are designed, procured and installed to specific specifications.

(1) Scope – The need to maintain pressure barriers (which also serve as fire barriers) is required by VCSNS Technical Specification 4.7.6.e.3. There are 34 doors that are Nuclear Safety Related pressure resistant doors. Thirteen (13) doors are Quality Related pressure doors. There are 47 doors that are rated as pressure resistant.

(2) Preventive Actions – No actions are taken as part of this program to prevent aging effects or to mitigate aging degradation. The Pressure Door Inspection Program is a condition monitoring program.

(3) Parameters Monitored or Inspected – Parameters monitored for Nuclear Safety Related pressure doors are loss of material of doors and door hardware. Parameters monitored for Quality Related pressure doors are loss of material of doors and door hardware. Excessive wear for door appurtenances such as latches, gaskets, hinges, sills, and closing devices are additional attributes in the technical requirements package, but are not credited for license renewal.

(4) Detection of Aging Effects – The pressure door inspection program detects structural damage or degradation, including loss of material due to corrosion prior to loss of intended function.

(5) Monitoring and Trending – Aging effects for Quality Related pressure doors are detected by a visual examination of the door and frame and functional testing for closure. Aging effects for Nuclear Safety Related pressure doors are detected by visual examination. No actions are taken as part of this program to trend inspections or test results.

(6) Acceptance Criteria – Quality Related pressure door acceptance criteria is provided in technical requirement packages. Nuclear Safety Related pressure door acceptance criteria is provided in surveillance test procedures. Acceptance criteria for self-closing doors are that hinges are intact with all screws tight, pins in good condition, and the door closes. Acceptance criteria for double self-closing doors are that bolts are in good condition, the astragal (metal molding strip) is in good condition, and the door closes. Automatic closing doors are checked to be in good operating condition and the door closes. Acceptance criteria for hollow pressure doors are no holes and no damage in the skin of the door or the frame.

(7) Corrective Actions – Minor abnormalities (loose knobs, latches or other appurtenances) are repaired using guidance provided by the vendor. The condition evaluation report (CER) process is initiated for pressure doors that do not meet the acceptance criteria. A CER is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) Administrative Controls – The need to maintain pressure barriers (which also serve as fire barriers) is required by VCSNS Technical Specification 4.7.6.e.3. The surveillance requirements are established in fire protection procedures. Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

B.1.20.1 Operating Experience

VCSNS has no failures or adverse trends for Nuclear Safety Related or Quality Related pressure doors. Inspections in the last five years do not identify any non-conformances (NCNs) relative to the acceptance criteria.

An occurrence of steam propagation into sensitive rooms through fire doors was identified. One door was replaced with a Quality Related pressure resistant / fire door, while a Quality Related pressure resistant door was added at another location.

No non-conformance notices (NCNs) or condition evaluation reports (CERs) were initiated for pressure doors relevant to aging.

The frequency of inspections performed since the implementation of the Technical Specifications requirements in 1984 is acceptable based on industry operating experience. A review of pressure door inspections confirms the reasonableness and acceptability of this inspection frequency such that any degradation of a door is detected prior to loss of function.

If the results of the visual inspection indicate that repairs are required, then specific repairs are made in accordance with plant procedures. The pressure door inspections are implemented by plant procedures and controlled by the SCE&G Quality Assurance Program.

B.1.20.2 Conclusion

The Pressure Door Inspection Program has been demonstrated to be capable of detecting and managing the effects of aging for pressure doors. The Pressure Door Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.21 SERVICE WATER POND DAM INSPECTION PROGRAM

The Service Water Pond Dam Inspection Program is consistent with XI.S7, *RG 1.127 Inspection of Water-Control Structures Associated With Nuclear Power Plants*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Service Water Pond Dam Inspection Program prior to the period of extended operation.

NUREG-1801 Program	Attributes	Enhancement
XI.S7, RG 1.127 Inspection of Water Control Structures Associated with Nuclear Power Plants	1.) Scope	North Dam piezometers will be added.
	3.) Parameters Monitored / Inspected	Water level.
	5.) Monitoring and Trending	Inspections will be made every 5-years concurrent with the RG 1.127 inspections.
	6.) Acceptance Criteria	Nominal elevation of adjacent Service Water Pond and Monticello Reservoir.

B.1.21.1 Operating Experience

During each inspection of the Service Water Pond Dams and West Embankment a review of the previous inspection's observations/recommendations is performed and the current status (such as repairs implemented or continued monitoring) is documented. Previous abutment erosion control modifications completed in 1989 significantly reduced earlier erosion problems overall, as noted by inspections performed in 1990 and 1995. Additional grading of diversion trenches/berms to direct rainwater away from the dams has further controlled erosion. There are currently no erosion areas that have a direct impact on any of the earthen structures. Weed, brush and sapling growth are controlled via cutting or spraying of herbicides conducted in accordance with plant procedures.

Structural calculations document the results of the survey monitoring data for the SWP North, South Dam and West Embankment. The calculations provide a review of the vertical and horizontal displacements of the Service Water Pond North Dam and South Dam since

1977. The calculations also provide a review of the vertical displacement of the West Embankment since 1978 and the horizontal displacement of the West Embankment since 1983. For the 2000 survey, all vertical and horizontal displacements were within the acceptance criteria as compared to the previous survey and found to be acceptable. Structural calculations also provide a review of the slope survey of the West Embankment since 1983. For the 2000 survey, all of the measurements were within the acceptance criteria as compared to the previous survey and found to be acceptable. No further evaluations were required and no unusual trends were noted.

In addition to the five year inspection of the Service Water Pond Dams required by the NRC, FERC conducted inspections of the Service Water Pond Dams in February 1997, July 1999, and July 2001. The conclusions reached by these inspections were that no significant conditions were observed that were considered detrimental to the safety of the Dams.

The 1997 FERC Dam safety inspection report [**Reference B-13**] recommended that SCE&G visually inspect the Service Water Pond Dams and West Embankment annually and test the accessible piezometers. The annual visual inspection is scheduled for the fall of each year. The first annual visual inspection and testing of the accessible piezometers was conducted in November 1999. Three accessible piezometers located along the crest of the North Dam were tested and found to be functional with acceptable results.

B.1.21.2 Conclusion

The Service Water Pond Dam Inspection Program has been demonstrated to be capable of detecting and managing trends in movement and the effects of aging for the service water dams. The Service Water Pond Dam Inspection Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.22 SERVICE WATER STRUCTURES SURVEY MONITORING PROGRAM

There is no NUREG-1801 item addressing this program. This is a plant specific program.

Survey monitoring is required for structures that are supported by earthen fill material and that have exhibited the potential for settlement. Settlement is not considered adverse unless it imposes stresses on structures that may exceed their design capacities. Initial settlement of the Service Water Pump House (SWPH) and the Service Water Intake Structure (SWIS) was much more than the original pre-construction estimates. As a result, survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” is conducted to monitor any differential in vertical and horizontal displacement. This monitoring is conducted to satisfy the requirements specified by Operating License condition 2.C.5 and **FSAR Section 2.5.4.10.6.2**.

(1) **Scope** – Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” is conducted in accordance with plant procedures.

(2) **Preventive Actions** – No actions are taken, as part of the survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A”, which prevent aging effects or mitigate aging degradation. Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” is a condition monitoring program.

(3) **Parameters Monitored or Inspected** – Survey monitoring is conducted to detect any vertical and/or horizontal movement associated with settlement of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A”. The survey monitoring data is reviewed by Design Engineering to ensure that settlements remain within established criteria.

In addition to survey monitoring, the structures are visually inspected in accordance with engineering services procedures for the following:

SWPH	movement, alignment or sloughing, cracking, settlement, and structural degradation
SWIS	cracking (per underwater diver’s inspection)
SW Electrical Duct Bank	differential movement and integrity of the expansion joint material
SW Intake Line “A”	ground above is inspected for settlement, sloughing, surface cracking, and erosion

(4) **Detection of Aging Effects** – Attributes associated with aging for the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” are detected by the survey monitoring. The survey results are reviewed and evaluated for trends in movement associated with settlement that exceeds the acceptance criteria. This review and the visual inspection of the structures will detect any adverse horizontal or vertical displacements prior to the loss of structure intended function(s).

(5) **Monitoring and Trending** – Aging effects associated with settlement for the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” are detected by survey monitoring in accordance with Current Licensing Basis (CLB) requirements. Survey monitoring data is retained in sufficient detail to permit adequate confirmation of the inspection program. The survey data reports and reviews/evaluations are filed in structural calculations. In particular these records identify the person(s) performing the survey, the structure/component and points surveyed, the person(s) reviewing/evaluating the survey data, whether or not the results are acceptable, discrepancies and their causes, and any corrective action(s) taken as a result. Trending is accomplished by comparing the current survey data to the previous survey data and evaluating for trends in movement that exceed the acceptance criteria.

(6) **Acceptance Criteria** – The acceptance criteria and guidelines for reviewing the survey settlement data for the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” are specified in design engineering guidelines. Survey results are evaluated for adverse trends in vertical displacement. The measurements are compared to the previous survey results and to acceptance criteria defined in engineering guidelines. The SWIS is also monitored for differential displacement between the middle and ends of the tunnel. If the acceptance criterion for the differential displacement is reached or exceeded, further engineering evaluations are required.

(7) **Corrective Actions** – Any settlement of structures or components that exceeds the established acceptance criteria is evaluated for adverse trends to determine whether or not there is a potential problem. Corrective actions may include increased frequency of inspection or further engineering evaluations to ascertain an exact cause for the movement. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls** – Periodic survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” is performed in accordance with Current Licensing Basis (CLB) requirements. Survey data is collected and documented. Design Engineering reviews the survey data in accordance with procedures and documents the results of the evaluation in structural calculation series. Survey monitoring of the SWPH, SWIS, Electrical Duct Banks, and Service Water Intake Line “A” are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

B.1.22.1 Operating Experience

Initial settlement of the SWPH and SWIS was much more than the original pre-construction estimates. The degree and manner of settlement caused cracking to occur in the SWIS, which was subsequently repaired (grouted). A special settlement study was performed for the SWPH and SWIS. There has been no significant settlement of the SWPH or SWIS since December 1978, subsequent to filling the Service Water Pond in February 1978.

Since 1991, there have been two instances where movement of the SWPH exceeded the acceptance criteria. The first instance was in February 1991, a re-survey was conducted in March 1991 and it was determined that the initial survey data was in error. In the second instance (July 1994), the acceptance criterion was minimally exceeded. Considering survey process inaccuracy and seasonal fluctuations affecting data collection, the total differential was not considered significant enough to warrant further evaluation. Survey results from 1977 to the present are documented in structural calculations.

Survey monitoring for differential settlement (middle to ends) of the SWIS has been conducted since February 1985. Between the July 1985 survey and February 1986 survey of the SWIS there was a sudden increase in the recorded differential displacement for which no ready explanation could be found. As a result of this sudden change, the survey monitoring frequency was increased to monthly for a period of eight months and the results showed the differential movement remained steady. Consequently, the frequency of monitoring was returned back to semi-annually. No further significant increase in differential movement has been recorded since February 1986 and the total settlement to date is within the acceptance limit.

No significant differential settlement was expected between the SWPH and incoming buried services as these were intentionally laid and connected to the SWPH after the major initial settlement during construction and the effects of filling the Service Water Pond in February 1978 had ceased. However, semi-annual survey data is recorded and evaluated.

Settlement of the Electrical Duct Banks is measured from inside the SWPH where the duct banks terminate on the inside face of the west wall of the SWPH. Historically, gap measurements have not undergone any significant changes since monitoring began, with any differential measurements well within the established acceptance criteria. Survey results are documented in structural calculations.

Service Water Intake Line “A” settlement has been monitored since January 1983. Since then there has been no appreciable movement or trend based on data reviews. However, there have been three occasions, one each in 1996, 1999 and 2000, when the acceptance criteria was minimally exceeded. These conditions are considered acceptable since the overall measurements remain within the general bounds of the long-term trend of data. These minor fluctuations may well be attributed to survey process imprecision, seasonal changes between summer and winter surveys, or ground water fluctuations. Survey results are documented in structural calculations.

B.1.22.2 Conclusion

The Service Water Structures Survey Monitoring Program has been demonstrated to be capable of detecting and managing trends in movement associated with settlement of the service water structures. The Service Water Structures Survey Monitoring Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.23 UNDERWATER INSPECTION PROGRAM (SWIS AND SWPH)

There is no NUREG-1801 item addressing this program. This is a plant specific program.

(1) Scope – The scope of the SWIS underwater inspection, conducted in accordance with engineering services procedures, includes visual inspection of the interior length of the intake tunnel, survey monitoring masts, trash racks, access ladder and east end wing walls. The scope of the SWPH underwater inspection, conducted in accordance with engineering services procedures, includes a visual inspection of the intake tunnel, traveling screens/bays and service water pump bays.

(2) Preventive Actions – No actions are taken, as part of the SWIS and SWPH underwater inspections, which prevent aging effects or mitigate aging degradation. The Underwater Inspection Program (SWIS and SWPH) is a condition monitoring program.

(3) Parameters Monitored or Inspected – Guidelines for the underwater inspection of the SWIS and SWPH are specified in engineering services procedures. The main reason for inspecting the SWIS is to measure/monitor cracks (old and new) in the concrete structure that originated due to earlier settlement. Additionally, a general inspection of the structure is made to document the as-found condition, noting any unusual observations. The specific areas that are inspected (and for which the condition is documented) are the access ladder, trash racks, survey monitoring masts, and concrete wing walls at the intake end of the SWIS.

Underwater inspections of the SWIS and SWPH monitor corrosion and fouling within the service water system. The SWIS and the SWPH forebay area, traveling screen bays and service water pump bays are inspected for fouling (clam and silt) accumulations. The density of the accumulation is documented and subsequently removed. The submerged trash racks, traveling screen components, service water pump components, and other structural components are inspected for corrosion. Any corrosion observed is documented in the inspection report.

(4) Detection of Aging Effects – Attributes associated with aging for the SWIS and SWPH are detected by the underwater inspections. Additionally, survey monitoring of the SWIS and SWPH will detect any horizontal or vertical movement associated with settlement.

(5) Monitoring and Trending – The underwater inspection reports are retained in sufficient detail to permit adequate confirmation of the inspection programs. The SWIS inspection documentation and reviews/evaluations are filed in structural calculations. In particular these records include the subcontractor’s underwater inspection report, Design Engineering review and evaluation of the results, comparison with previous inspection results, and whether or not the results are acceptable. Discrepancies and their cause and any corrective action resulting from these inspections are also documented in the calculations.

(6) **Acceptance Criteria** – The acceptance criteria for the underwater inspection of the SWIS is that the inspection data is reviewed by engineering.

Cracks (old and new) are documented and mapped on an engineering procedure attachment. Crack width is measured using wire gauges on a “Go – No/Go” basis by inserting the wire into the crack.

(7) **Corrective Actions** – Any problems or concerns observed during the underwater inspections of the SWIS or SWPH that exhibit attributes associated with aging are evaluated by engineering for continued service or repair as required and documented. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls** – The SWIS crack inspections are governed by VCSNS Operating License, Condition 2.C.5.d and implemented through an engineering procedure. The inspection of the SWIS and SWPH to monitor and control corrosion and fouling within the service water system is governed by the SCE&G response to Generic Letter 89-13 [**Reference B-10**] and implemented through another engineering procedure. Inspections are scheduled by Preventive Maintenance Task Sheets (PMTS) initiated in accordance with station administrative procedures.

B.1.23.1 Operating Experience

VCSNS Operating License Condition 2.C.5.d requires SCE&G to perform an inspection of the SWIS every five years to monitor and measure the cracks in the reinforced concrete tunnel which originated due to settlement problems during construction.

Cracks in the SWIS (tunnel) which were identified during construction were grouted with a high strength epoxy grout in 1978 prior to filling the Service Water Pond. Underwater inspections were initiated in 1983 and have been performed every five years. The inspections of

1983 and 1988 identified very little change in the existing grouted and ungrouted cracks along with a few new hairline cracks. An improved method of marking old cracks was implemented during 1988, with additional improvements made during the 1993 inspection, which allows better distinction of old versus new ungrouted cracks.

The 1993 inspection also identified nine existing cracks that had widened and four cracks with a maximum width greater than the minimum criteria. These cracks were evaluated and documented in a structural calculation. The cracks were grouted in 1994 with a flexible urethane grout in order to eliminate/reduce the potential for corrosion of the reinforcing steel.

No new cracks were identified during the 1998 inspection and all cracks that had any visible gap were measured to be less than the minimum criteria. The 1998 inspection data for each crack was compared to the results of the 1993 inspection to ensure consistency and no significant differences were noted between the two inspection reports.

After filling the Service Water Pond, visual inspection and cleaning of the SWIS and SWPH was performed once each refueling cycle within the preventive maintenance program. In response to Generic Letter 89-13, a new engineering procedure was developed to direct the SWIS and SWPH inspections. A review of the inspection data for the past five years shows that no corrosion has been discovered on the trash racks, foot section of each traveling screen, endbell of each service water pump and/or other submerged structural components. The location and density of fouling accumulations (e.g., silt and clams) is recorded and subsequently removed by divers using an eductor.

B.1.23.2 Conclusion

The Underwater Inspection Program (SWIS and SWPH) has been demonstrated to be capable of detecting and managing the effects of aging for concrete components in fluid environments. The Underwater Inspection Program of the Service Water Intake Structure (SWIS) and Service Water Pump House (SWPH) provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.24 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is consistent with XI.M31, *Reactor Vessel Surveillance*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Reactor Vessel Surveillance Program prior to the period of extended operation.

NUREG-1801 Program	Attributes	Enhancement
XI.M31, Reactor Vessel Surveillance	1.) Scope	Perform a one-time analysis to demonstrate that the materials in the inlet and outlet nozzles and upper shell course will not become controlling during the period of extended operations. Successful demonstration will preclude the addition of such materials to the material surveillance program for the period of extended operation.
	4.) Detection of Aging Effects	Remove both remaining surveillance capsules during RF-14, have one capsule analyzed and place the other capsule in storage, in accordance with the recommendations of Item 6 of the December 3, 1999 Christopher Grimes (NRC) letter to Douglas Walters(NEI) [**Reference B-11**].

B.1.24.1 Operating Experience

Industry experience to date supports the conclusion that a Reactor Vessel Surveillance Program compliant with regulatory requirements provides assurance that fracture toughness requirements are met.

Direct measurement of material properties of the irradiated samples recovered from the surveillance capsules provides objective evidence of the effects of radiation embrittlement on reactor vessel materials. Analysis of the VCSNS surveillance capsules removed to date demonstrates that changes to the material properties of the vessel beltline materials are well known and will not result in brittle failure. The VCSNS vessel beltline materials are currently demonstrated by Westinghouse analysis to be below the RTPTS screening criteria for 48 effective full power years (EFPY). Further evaluation to 54 EFPY will be performed prior to the period of extended operations.

The NRC Staff response to License Renewal Issue 98-0085 [**Reference B-11**] confirms that existing radiation surveillance programs provide a suitable means of monitoring vessel fracture toughness to neutron fluence levels associated with the end of the extended operating period, provided that surveillance capsules are available or can be reconstituted.

Industry experience with Inservice Inspections suggests that such programs are capable of detecting flaws before those flaws grow larger than one quarter of vessel wall thickness.

The fuel loading program was revised to implement a low-leakage pattern that reduced the fast neutron flux escaping the core. This change effectively reduced the neutron flux on, and resulting embrittlement of the reactor vessel.

B.1.24.2 Conclusion

The Reactor Vessel Surveillance Program has been demonstrated to be capable of managing reduction of fracture toughness for the reactor vessel beltline materials. The Reactor Vessel Surveillance Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.25 PREVENTIVE MAINTENANCE ACTIVITIES - TERRY TURBINE

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The Preventive Maintenance Activities - Terry Turbine is a condition monitoring program that manages loss of material due to general corrosion of carbon steel. This program is composed of controlled plant procedures.

(1) **Scope -** The Preventive Maintenance Activities – Terry Turbine is applicable to the turbine casing and components exposed to an air environment with periodic exposure to steam.

(2) **Preventive Actions -** No actions are taken as part of the Preventive Maintenance Activities – Terry Turbine to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected -** Parameters monitored or inspected as part of the Preventive Maintenance Activities - Terry Turbine include visible evidence of corrosion on internal surfaces to indicate potential loss of material.

(4) **Detection of Aging Effects -** The Preventive Maintenance Activities - Terry Turbine will detect the presence and extent of aging effects on internal surfaces by visual inspection prior to a loss of component intended function. These effects are loss of material due to general corrosion.

(5) **Monitoring and Trending -** Routine periodic visual inspections are conducted as part of the Preventive Maintenance Activities - Terry Turbine in order to detect age-related degradation and to initiate corrective actions as necessary.

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

(6) **Acceptance Criteria -** The acceptance criteria for the Preventive Maintenance Activities - Terry Turbine is no unacceptable visible indication of loss of material. Indications of loss of material are evaluated by engineering to determine if the condition could result in a loss of the component intended function(s).

(7) **Corrective Actions -** If the results of an inspection are not acceptable an engineering evaluation(s) is performed to assess the material condition and to determine whether the component intended function is affected. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded

conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Preventive Maintenance Activities - Terry Turbine are implemented in accordance with controlled station procedures and work processes.

B.1.25.1 Operating Experience

A review of work histories, including non-conformance notices (NCNs), condition evaluation reports (CERs), and problem reports, for the past ten years reveals that no age-related degradation has been detected for the subject components. The inspection activities incorporate vendor recommendations. Based on vendor recommendations and inspection results, continuation of the periodic inspections will identify any age-related degradation prior to a loss of component intended function.

B.1.25.2 Conclusion

The Preventive Maintenance Activities - Terry Turbine has been demonstrated to be capable of detecting and managing loss of material in carbon steel components of the Terry Turbine. The Preventive Maintenance Activities - Terry Turbine provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.1.26 Preventive Maintenance Activities - Ventilation Systems Inspections

There is no NUREG-1801 item addressing this program. This is a plant specific program.

The Preventive Maintenance Activities - Ventilation Systems Inspections is a condition monitoring program that will manage loss of material due to boric acid corrosion, galvanic corrosion and general corrosion in carbon steel, galvanized steel, and copper components, and fouling due to particulates in aluminum, copper, and copper-nickel heat exchanger components.

(1) Scope - The Preventive Maintenance Activities – Ventilation Systems Inspections is applicable to the following systems and components exposed to a ventilation environment:

Air Handling (HVAC) System:
- Carbon steel air handling units, air plenums, and fan and plenum housings.
- Galvanized steel air handling units and heat exchanger tubesheets.
- Copper heat exchanger fins and tubes.

Component Cooling System (pump motor coolers):
- Aluminum heat exchanger fins.
- Copper-nickel heat exchanger tubes.
- Carbon steel heat exchanger tubesheets.

Local Ventilation and Cooling System:
- Galvanized steel air handling units.
- Copper heat exchanger fins and tubes.

(2) Preventive Actions - No actions are taken as part of the Preventive Maintenance Activities – Ventilation Systems Inspections to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored or Inspected - Parameters monitored or inspected as part of the Preventive Maintenance Activities – Ventilation Systems Inspections include visible evidence of corrosion, including pitting and discoloration, to indicate possible loss of material, and accumulation of dust and particulates on fins and tubes to indicate possible fouling. For those components located in the Reactor Building, visible evidence of boron precipitation may indicate loss of material due to boric acid corrosion.

(4) Detection of Aging Effects - In accordance with the information provided in Monitoring and Trending below, the Preventive Maintenance Activities – Ventilation

Systems Inspections will detect the presence and extent of the following aging effects prior to a loss of component intended function:

- Loss of material due to boric acid corrosion, galvanic corrosion and general corrosion.
- Fouling due to particulates (in the ventilation environment).

(5) Monitoring and Trending - Routine periodic inspections are conducted as part of the Preventive Maintenance Activities – Ventilation Systems Inspections in order to detect age-related degradation and to initiate corrective actions, as necessary.

Except for the Reactor Building Cooling Units (RBCUs), no actions are taken as part of this program to trend inspection results. For the RBCUs, an engineering procedure requires recording of temperature monitoring data annually or at least once per refueling cycle.

(6) Acceptance Criteria - The acceptance criteria for the Preventive Maintenance Activities – Ventilation Systems Inspections is no unacceptable loss of material or fouling of subject components that could result in a loss of the component intended function(s), as determined by engineering evaluation. The engineering procedure contains specific acceptance criteria for the RBCUs.

(7) Corrective Actions - If the results of a inspection are not acceptable, an engineering evaluation is performed to assess material condition and to determine whether the component intended function is affected. A condition evaluation report (CER) is generated to provide a thorough description of the problem along with a disposition specifying the corrective action(s). The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) Confirmation Process - Engineering reviews the inspection results for completeness and acceptability, The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) Administrative Controls - The Preventive Maintenance Activities - Ventilation Systems Inspections are implemented in accordance with controlled station procedures and work processes.

B.1.26.1 Operating Experience

A review of work histories, including non-conformance notices (NCNs), condition evaluation reports (CERs), and problem reports, for the past ten years reveals that no age-related degradation has been detected for the subject components.

B.1.26.2 Conclusion

The Preventive Maintenance Activities - Ventilation Systems Inspections has been demonstrated to be capable of detecting and managing the effects of aging for components exposed to a ventilation environment. The Preventive Maintenance Activities - Ventilation Systems Inspections provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.2.0 NEW AGING MANAGEMENT ACTIVTIES

B.2.1 ABOVE GROUND TANK INSPECTION

The Above Ground Tank Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Above Ground Tank Inspection is a new one-time inspection that will detect and characterize loss of material due to galvanic and general corrosion in an internal air space environment, and loss of material and cracking to the due to the corrosive effects of alternate wetting and drying in treated or borated water environments. The Above Ground Tank Inspection will also detect and characterize loss of material due to general corrosion in a treated water (with uncontrolled oxygen levels) environment. The Above Ground Tank Inspection will be performed prior to the period of extended operation.

(1) Scope - The Above Ground Tank Inspection is applicable to the internal surfaces of the following components:

- Carbon steel tanks exposed to an internal air space environment in the Condensate, Component Cooling, and Chilled Water Systems;
- Carbon steel pipe and valves exposed to an internal air space environment in the Component Cooling System;
- Carbon steel and stainless steel tanks exposed to a treated water environment in the Condensate, Component Cooling, Reactor Makeup Water Supply and Chilled Water Systems;
- Carbon steel tanks, pipe and valves exposed to treated water having uncontrolled oxygen levels in the Sodium Hydroxide Storage Tank in the Reactor Building Spray System;
- Stainless steel tanks exposed to a borated water environment in the Refueling Water System (Refueling Water Storage Tank).

(2) Preventive Actions - No actions are taken as part of the Above Ground Tank Inspection to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored/Inspected - The parameters inspected by the Above Ground Tank Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking or other age-related degradation.

(4) Detection of Aging Effects - The Above Ground Tank Inspection will use a combination of proven volumetric and visual examination techniques on a sample population of subject components, to be determined by engineering evaluation.The results will be applied to the remainder of the components within the scope of the inspection activity. For components exposed to borated and treated water environments, the sample population should include locations near the air-water interface within the stainless steel Refueling Water Storage Tank (RWST), and near the air-water interface within one of the following carbon steel tanks: the Condensate Storage Tank, the Component Cooling Surge Tank, or one of the Chilled Water Expansion Tanks. It is expected that an engineering evaluation will confirm that the borated water environment of the RWST is more likely to concentrate contaminants at the air-water interface than the treated water environment of the Reactor Makeup Water Supply Tank.

For components exposed to treated water with uncontrolled oxygen levels, the sample population should include the submerged portions of the Sodium Hydroxide Tank.

For components exposed to an internal air space environment, the sample population should include locations within the air space of one of the following carbon steel tanks: the Condensate Storage Tank, the Component Cooling Surge Tank, or one of the Chilled Water Expansion Tanks. If possible, to simplify the inspection, the same tank chosen to inspect for corrosive impacts of alternate wetting and drying should be selected here.

The Above Ground Tank Inspection will detect the presence and extent of loss of material and cracking on internal surfaces prior to a loss of component intended function.

(5) Monitoring and Trending - No actions are taken as part of the Above Ground Tank Inspection to trend inspection results. This is a new one-time inspection used to determine if further actions are required.

(6) Acceptance Criteria - The acceptance criteria for the Above Ground Tank Inspection is no unacceptable loss of material or cracking of subject components that could result in a loss of the component intended function(s) as determined by engineering evaluation.

(7) Corrective Actions - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the 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 the engineering evaluation determines that additional information is required to more fully characterize 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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Above Ground Tank Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.1.1 Operating Experience

The Above Ground Tank Inspection is a new one-time inspection activity for which there is no operating experience.

B.2.1.2 Conclusion

Implementation of the Above Ground Tank Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.2 DIESEL GENERATOR SYSTEMS INSPECTION

The Diesel Generator Systems Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Diesel Generator Systems Inspection is a new one-time inspection that will detect and characterize loss of material due to general corrosion and the corrosive impacts of alternate wetting and drying in air-gas environments. The Diesel Generator Systems Inspection will be performed prior to the period of extended operation.

(1) **Scope** - The Diesel Generator Systems Inspection is applicable to the following components in the Diesel Generator Services System:
- Carbon steel expansion joints normally exposed to moist air, and exposed to exhaust air during engine operation, both of which are air-gas internal environments.
- Carbon steel tanks and associated tubing components exposed to starting and control air, an air-gas internal environment.

(2) **Preventive Actions -** No actions are taken as part of the Diesel Generator Systems Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected -** The parameters inspected as part of the Diesel Generator Systems Inspection include wall thickness and/or visible evidence of corrosion, including pitting and discoloration, to indicate possible loss of material for the carbon steel components.

(4) **Detection of Aging Effects -** The Diesel Generator Systems Inspection will use a combination of proven volumetric and/or visual examination techniques on a sample population of subject components, to be determined by engineering evaluation. The results of the inspection will be applied to the remainder of the components within the scope of the inspection activity.

The Diesel Generator Systems Inspection will detect the presence and extent of any loss of material due to general corrosion and the corrosive impacts of alternate wetting and drying for the subject components prior to a loss of component intended function.

(5) **Monitoring and Trending -** No actions are taken as part of the Diesel Generator Systems Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria -** The acceptance criteria for the Diesel Generator Systems Inspection is no unacceptable loss of material of subject components that could

result in a loss of the component intended function(s) as determined by engineering evaluation.

(7) **Corrective Actions -** 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 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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls** - The Diesel Generator Systems Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.2.1 Operating Experience

The Diesel Generator Systems Inspection is a new one-time inspection program for which there is no operating experience.

B.2.2.2 Conclusion

Implementation of the Diesel Generator Systems Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.3 LIQUID WASTE SYSTEM INSPECTION

The Liquid Waste System Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Liquid Waste System Inspection is a new one-time inspection that will detect and characterize loss of material due to crevice and pitting corrosion, and cracking due to stress corrosion cracking (SCC) in unmonitored and uncontrolled borated water environments. The Liquid Waste System Inspection will be performed prior to the period of extended operation.

(1) Scope - The Liquid Waste System Inspection is applicable to stainless steel components exposed to unmonitored and uncontrolled borated water in the following systems:

- Nuclear Plant Drains (ND) – pipe and valve bodies.
- Liquid Waste Processing (WL) – pipe, valve bodies and heat exchanger components.

The unmonitored and uncontrolled borated water environment consists of the following:

- Contents of the Reactor Building or Incore Instrumentation Sumps being discharged through a containment penetration (ND System).
- Contents of the Reactor Coolant Drain Tank (RCDT) on the tube-side of the RCDT Heat Exchanger and passing through a containment penetration (WL System).
- Concentrates from the Waste Evaporator Package on the tube-side of the Concentrates Sample Cooler (WL System).

(2) Preventive Actions - No actions are taken as part of the Liquid Waste System Inspection to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored or Inspected - The parameters inspected by the Liquid Waste System Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking, or other age-related degradation.

(4) Detection of Aging Effects - The Liquid Waste System Inspection will use a combination of proven volumetric and visual examination techniques on a sample population of subject components, to be determined by engineering evaluation. The results of the inspection will be applied to the remainder of the components within the scope of the inspection activity. The sample population will consist of susceptible locations within the boundaries of either of the two affected containment penetrations, as well as the internal tube surfaces of the affected heat exchangers.

The Liquid Waste System Inspection is a new one-time inspection that will detect the presence and extent of any loss of material and cracking prior to a loss of component intended function.

(5) Monitoring and Trending - No actions are taken as part of the Liquid Waste System Inspection to trend inspection results. This is a new one-time inspection used to determine if further actions are required.

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

(7) Corrective Actions - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the 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 the engineering evaluation determines that additional information is required to more fully characterize 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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) Confirmation Process - Engineering reviews the inspection results for completeness and acceptability, The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) Administrative Controls - The Liquid Waste System Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.3.1 Operating Experience

The Liquid Waste System Inspection is a new one-time inspection for which there is no operating experience.

B.2.3.2 Conclusion

Implementation of the Liquid Waste System Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.4 REACTOR VESSEL INTERNALS INSPECTION

The Reactor Vessel Internals Inspection will be consistent with XI.M16, *PWR Vessel Internals,* as identified in NUREG-1801 prior to the period of extended operation, with the following clarification:

- **Detection of Aging Effects:** The VCSNS resolution criterion for the enhanced VT-1 examination is expected to be less than specified in the GALL program.

The reactor vessel internals inspection is a new inspection, supplementing the In-Service Inspection (ISI) Plan, that will assess the condition of reactor vessel internals in order to assure that the applicable aging effects will not result in loss of the intended functions during the period of extended operation.

(1) Scope - The Reactor Vessel Internals Inspection is applicable to the following stainless steel reactor vessel internals components, that are exposed to the borated water environment of the Reactor Coolant System:

- Baffle and Former Assembly (and Bolts)
- Core Barrel (including Flange and Outer Nozzle)
- Fuel Alignment Pins
- Guide Tubes (including Bolts and Support Pins)
- Head and Vessel Alignment Pins
- Hold-down Springs
- Lower Core Plate
- Lower Support Columns (and Bolts)
- Lower Support Plate
- Neutron Panels
- Radial Keys
- Secondary Core Support
- Spray Nozzles
- Upper Core Plate (and Alignment Pins)
- Upper Instrumentation Column
- Upper Support Column (and Bolts)
- Upper Support Plate Assembly
-

The Reactor Vessel Internals Inspection is also applicable to the following nickel-based reactor vessel internals component that are exposed to the borated water environment of the Reactor Coolant System:

- Clevis Inserts and Clevis Insert Bolts

(2) **Preventive Actions -** No actions are taken as part of the Reactor Vessel Internals Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored/Inspected -** The parameters inspected by the Reactor Vessel Internals Inspection include visual evidence of cracking (enhanced by reduction of fracture toughness) and loss of material, and volumetric measurements to indicate possible changes in dimensions, cracking, reduction of fracture toughness and loss of preload.

(4) **Detection of Aging Effects -** In accordance with the information provided in Monitoring and Trending below, the Reactor Vessel Internal Inspection will detect the following aging effects prior to a loss of component intended function:

- Changes in dimensions due to irradiation creep and void swelling
- Cracking due to irradiation-assisted stress corrosion cracking (IASCC)
- Cracking due to primary water stress corrosion cracking (PWSCC) in nickel-based materials
- Loss of material due to wear
- Loss of preload due to stress relaxation
- Reduction of fracture toughness due to irradiation embrittlement and void swelling

(5) **Monitoring and Trending -** Effective and proven volumetric and visual examination techniques will be selected for use in performing the inspection.

The Reactor Vessel Internals Inspection includes the following inspection activities, which will be conducted on a sample of the most susceptible components, as determined by engineering evaluation.

For those components that are accessible or can be rendered accessible by the removal of the core and/or other internals for examination, a visual inspection will be performed to detect the presence and extent of cracking due to IASCC (and enhanced by reduction of fracture toughness due to irradiation embrittlement) and loss of material due to wear.

For bolts and other inaccessible components, a volumetric inspection will be performed to detect the presence and extent of changes in dimensions due to irradiation creep and void swelling, cracking due to IASCC, loss of preload due to stress relaxation, and reduction of fracture toughness due to irradiation embrittlement and void swelling.

With respect to changes in dimensions due to void swelling, industry activities are under way to determine whether this is an aging effect requiring management for license renewal, and, if necessary, to develop and qualify methods for detection and management. These activities will be monitored by VCSNS, and will be performed if necessary.

(6) **Acceptance Criteria -** The Reactor Vessel Internals Inspection includes the following acceptance criteria:

For all subject components, critical crack size will be determined by analysis prior to the inspection.

For bolts, any detectable crack indication is unacceptable for a particular bolt. However, the intended function(s) of reactor vessel internals can be maintained with fewer than 100% of the bolts intact. That quantity, and specific bolt locations, will be determined by analysis prior to the inspection.

Specific acceptance criteria for changes in dimensions due to void swelling, loss of preload due to stress relaxation, and loss of material due to wear will be determined by analysis as part of the inspection plan.

Inspection results may also be compared with the acceptance standards of ASME Section XI, Subsections IWB-3400 and IWB-3500.

(7) **Corrective Actions -** If the results of the Reactor Vessel Internals Inspection are not acceptable, based on acceptance criteria to be determined by analysis, then actions will be taken to repair or replace the affected item, or to determine by analysis the acceptability of the item. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Reactor Vessel Internals Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.4.1 Operating Experience

The Reactor Vessel Internals Inspection is a new inspection for which there is no operating experience.

B.2.4.2 Conclusion

The Reactor Vessel Internals Inspection will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.2.5 REACTOR BUILDING COOLING UNIT INSPECTION

The Reactor Building Cooling Unit Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operation.

The Reactor Building Cooling Unit Inspection is a new one-time inspection that will detect and characterize loss of material due to crevice and pitting corrosion and cracking due to stress corrosion cracking resulting from exposure to an unmonitored and uncontrolled borated water environment. The borated water environment results from condensation out of the Reactor Building atmosphere, onto Reactor Building Cooling Unit cooling coils, and then into the associated drain lines. The Reactor Building Cooling Unit Inspection will be performed prior to the period of extended operation.

(1) **Scope -** The Reactor Building Cooling Unit Inspection is applicable to stainless steel pipe exposed to an unmonitored borated water environment in the reactor building cooling unit drain lines that are part of the VCSNS Roof Drain System.

(2) **Preventive Actions -** No actions are taken as part of the Reactor Building Cooling Unit Inspection to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected -** The parameters inspected by the Reactor Building Cooling Unit Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking or other age-related degradation.

(4) **Detection of Aging Effects -** The Reactor Building Cooling Unit Inspection will use a combination of proven volumetric and visual examination techniques at sample locations in the drain lines determined by engineering evaluation to be most susceptible to the applicable aging effects. If no parameters are known that would distinguish the susceptible locations, sample locations will be selected based on accessibility and radiological concerns, and the results will be applied to the associated piping.

The Reactor Building Cooling Unit Inspection will detect the presence and extent of any loss of material and cracking prior to a loss of component intended function.

(5) **Monitoring and Trending -** No actions are taken as part of the Reactor Building Cooling Unit Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria -** The acceptance criteria for the Reactor Building Cooling Unit Inspection is no unacceptable loss of material or cracking of subject compo-

nents that could result in a loss of the component intended function(s), as determined by engineering evaluation.

(7) Corrective Actions - If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the 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 the engineering evaluation determines that additional information is required to more fully characterize 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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) Confirmation Process - Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) Administrative Controls - The Reactor Building Cooling Unit Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.5.1 Operating Experience

The Reactor Building Cooling Unit Inspection is a new one-time inspection for which there is no operating experience.

B.2.5.2 Conclusion

Implementation of the Reactor Building Cooling Unit Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.6 SERVICE AIR SYSTEM INSPECTION

The Service Air System Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801 prior to the period of extended operations.

The Service Air System Inspection is a new one-time inspection that will detect and characterize loss of material due to general corrosion resulting from exposure to an internal moist air environment. The Service Air System Inspection will be performed prior to the period of extended operation.

(1) Scope - The Service Air System Inspection is applicable to carbon steel pipe, tubing, and valve body components exposed to internal moist air environment that perform the function of maintaining pressure boundary for containment integrity i the following locations:

- Service Air System components in the supply line to the Reactor Building where the line penetrates the containment.
- Service Air and Building Services Systems components used for the leak testing of the Personnel Hatch, Equipment Hatch, and Emergency Personnel Hatch seals.
- Building Services System components used to supply emergency air to the Personnel Hatch and Emergency Personnel Hatch.
- Instrument Air System components in the air intake piping upstream of the instrument air dryers where the piping penetrates the containment.

(2) Preventive Actions - No actions are taken as part of the Service Air System Inspection to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored or Inspected - The parameters inspected by the Service Air System Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material or other age-related degradation

(4) Detection of Aging Effects - The Service Air System Inspection will use a combination of proven volumetric and visual examination techniques to inspect for general corrosion at selected sample locations to be determined by engineering evaluation.

The Service Air System Inspection will detect the presence and extent of any loss of material prior to a loss of component intended function.

(5) Monitoring and Trending - No actions are taken as part of the Service Air System Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria -** The acceptance criteria for the Service Air System Inspection is no unacceptable loss of material in subject components that could result in a loss of the component intended function, as determined by engineering evaluation.

(7) **Corrective Actions -** If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function under the 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 the engineering evaluation determines that additional information is required to more fully characterize 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 under current licensing basis design conditions for the period of extended operation, then programmatic oversight will be defined. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Service Air System Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.6.1 Operating Experience

The Service Air System Inspection is a new one-time inspection for which there is no operating experience.

B.2.6.2 Conclusion

Implementation of the Service Air System Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.7 SMALL BORE CLASS 1 PIPING INSPECTION

The Small Bore Class 1 Piping Inspection will be consistent with XI.M32, *One-Time Inspection*, as identified in NUREG-1801.

The Small Bore Class 1 Piping Inspection is a new one-time inspection that will increase confidence in the current condition of small bore Reactor Coolant System piping that does not receive a volumetric examination. The Small Bore Class 1 Piping Inspection will be scheduled at or near the end of the second period of the fourth ISI interval.

(1) Scope - The Small Bore Class 1 Piping Inspection is applicable to Reactor Coolant System piping and fittings less than 4 inches NPS which are subject to cracking due to flaw growth and stress corrosion cracking (SCC). This inspection is elective in that a sample of piping less than 4 inches NPS that is not required to be examined volumetrically per ASME Section XI code requirements will be examined.

(2) Preventive Actions - No actions are taken as part of the Small Bore Class 1 Piping Inspection to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored or Inspected - The parameter inspected by the Small Bore Class 1 Piping Inspection is evidence of cracking as determined by visual and surface examination that indicates age-related degradation.

(4) Detection of Aging Effects - Table IWB 2500-1 of ASME Section XI contains the requirements for in-service inspection of Class 1 piping. The examination categories for piping require volumetric and surface examinations for piping greater than or equal to 4 inches NPS and surface examinations for piping greater than 1 inch NPS but less than 4 inches NPS. Piping less than or equal to 1 inch NPS is exempt from volumetric and surface examinations and receives visual examination during pressure testing.

Cracking due to flaw growth and SCC are aging mechanisms that originate from the inside diameter of the piping. Volumetric examination of small diameter piping by current ultrasonic or radiographic methods is often ineffective and provides poor results. Engineered removal and replacement of representative sections of small bore piping for destructive testing will provide more reliable inspection results. Destructive examination of the sample permits examination of the inside of the piping and allows for visual and surface examination of the interior surfaces without outage schedule pressure.

Inspection locations will be selected by engineering using risk-based approaches. Locations most susceptible to cracking will be identified based on engineering evaluation, operating experience, current code requirements, and industry initiatives. Actual inspection locations will be selected based on physical accessibility, exposure levels, and scheduling requirements as well as the results of a review of failure consequences. One of the sample locations should include a butt-weld.

The Small Bore Class 1 Piping Inspection will detect cracking due to flaw growth and SCC in stainless steel components prior to loss of component intended function.

(5) **Monitoring and Trending -** No actions are taken as part of the Small Bore Class 1Piping Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria -** No established acceptance criteria exists for destructive examinations or for volumetric examination of small bore Class 1 piping. Acceptance standards for a destructive examination will be determined by VCSNS if not established by industry initiative prior to the time of the inspection.

(7) **Corrective Actions -** Since destructive examination is the currently preferred technique (in the absence of reliable NDE methods) and since piping will be replaced in accordance with ASME Section XI, no corrective actions need be defined prior to the inspection. The results of the inspection may be used as a baseline inspection if results indicate a longer term monitoring program is warranted. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Small Bore Class 1 Piping Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.7.1 Operating Experience

The Small Bore Class 1 Piping Inspection is a new inspection for which there is no operating experience. NRC Information Notice 97-46, Unisolable Crack in High-Pressure Injection Piping, contains industry experience related to cracking of small bore piping.

B.2.7.2 Conclusion

Implementation of the Small Bore Class 1 Piping Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.8 WASTE GAS SYSTEM INSPECTION

The Waste Gas System Inspection will be consistent with XI.M32, *One-Time Inspection,* as identified in NUREG-1801 prior to the period of extended operation.

The Waste Gas System Inspection is a new one-time inspection that will detect and characterize loss of material due to crevice and pitting corrosion and cracking due to stress corrosion cracking in unmonitored and uncontrolled treated water, and cracking due to stress corrosion cracking in gas environments. The Waste Gas System Inspection will be performed prior to the period of extended operation.

(1) Scope - The Waste Gas System Inspection is applicable to the following Gaseous Waste Processing System components:

- Stainless steel pipe and valve bodies exposed to an unmonitored and uncontrolled treated water environment.
- Stainless steel tube coils and manifolds in the Hydrogen Recombiner Cooler Condenser exposed to a gas environment.

The unmonitored and uncontrolled treated water environment consists of condensation that forms within the Waste Gas Decay Tanks and is periodically pumped to the Volume Control Tank in the Chemical and Volume Control System.

The gas environment is a mostly nitrogen, with trace amounts of hydrogen, oxygen, and fission product gases, and water vapor from the recombination of hydrogen and oxygen.

(2) Preventive Actions - No actions are taken as part of the Waste Gas System Inspection to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored or Inspected - The parameters inspected by the Waste Gas System Inspection are wall thickness as a measure of loss of material, and visual evidence of loss of material, cracking or other age-related degradation.

(4) Detection of Aging Effects - The Waste Gas System Inspection will use a combination of proven volumetric and visual examination techniques on a sample population of subject components, to be determined by engineering evaluation. The results of the inspection will be applied to the remainder of the components within the scope of the inspection activity. The sample population should consist of at least one susceptible location in the stainless steel Waste Gas Decay Tank drain piping (preferably at a low point), and at least one susceptible location in the stainless steel tube-side inlet piping to the Hydrogen Recombiner Cooler Condenser.

The Waste Gas System Inspection will detect the presence and extent of any loss of material or cracking prior to a loss of component intended function.

(5) **Monitoring and Trending -** No actions are taken as part of the Waste Gas System Inspection to trend inspection results. This is a one-time inspection used to determine if further actions are required.

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

(7) **Corrective Actions -** If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the 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 the engineering evaluation determines that additional information is required to more fully characterize 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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Waste Gas System Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.8.1 Operating Experience

The Waste Gas System Inspection is a new one-time inspection activity for which there is no operating experience.

B.2.8.2 Conclusion

Implementation of the Waste Gas System Inspection will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.2.9 NON-EQ INSULATED CABLES AND CONNECTIONS INSPECTION PROGRAM

The Non-EQ Insulated Cables and Connections Inspection Program will be consistent with XI.E1, *Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements*, as identified in NUREG-1801 prior to the period of extended operation.

This same program will be applied to XI.E2, *Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits*, and will have the following clarification:

- The calibration of instrument circuits as a means to detect age-related degradation of cable insulation as identified in XI.E2 is not included in the VCSNS program. The visual inspection of instrument as well as power and control cables is considered a better means to identify age-related degradation due to localized ambient thermally and radiologically induced stress prior to loss of intended function. The cables addressed by XI.E2 are therefore bounded by the XI.E1 cable aging management program.

This is a new inspection program that will assess the condition of non-EQ insulated cables and connections to provide assurance that the aging effects of concern will not result in loss of the intended functions during the period of extended operation.

(1) Scope - The specific non-EQ insulated cables and connections that will be included in the aging management program for VCSNS include accessible non-EQ insulated cables and connections, including splices and terminal blocks, that are subject to degradation in the more adverse thermal and radiological areas of the plant. Selection of the areas to be inspected shall include considerations for circuits with potentially significant ohmic heating. While certain areas of the Intermediate and Auxiliary Buildings will be the focus, there will be flexibility to inspect cables and connections in a variety of Environmental Zones, as determined by the responsible electrical engineering group at VCSNS. The technical basis for the location selected will be documented. The program will involve a visual inspection of the accessible cables in these zones, to determine if the cable jackets or connections show any signs of cracking, embrittlement, discoloration, melting, or any other visible evidence of age-related degradation, which may lead to loss of the intended function.

(2) Preventive Actions – No actions are taken as a part of the Non-EQ Insulated Cables and Connections Inspection Program to prevent aging effects or to mitigate aging degradation. The program provides for component inspection only.

Issues of component degradation will be addressed within other program attributes.

(3) Parameters Monitored or Inspected – The parameters inspected as a part of the Non-EQ Insulated Cables and Connections Inspection Program include visual evidence of cable jacket or connection embrittlement, cracking, swelling, discoloration, melting, loss of dielectric strength leading to reduced insulation resistance and/or electrical failure.

(4) Detection of Aging Effects – The Non-EQ Insulated Cables and Connections Inspection Program, conducted in the more thermally and radiologically severe areas of the plant containing in-scope cables and connections will provide input into Monitoring and Trending (see below). These inspections will serve to detect degradation of cable and connections, which could ultimately lead to electrical failure. During each inspection, and visual evidence of embrittlement, cracking, swelling, discoloration, melting, degradation of organics, radiation-induced oxidation, and moisture intrusion will be evaluated.

(5) Monitoring and Trending - The Non-EQ Insulated Cables and Connections Inspection Program will be initially performed prior to the period of extended operation and then at 10-year intervals thereafter. Documentation of these inspections will be available in subsequent inspections for comparison, review, and evaluation such that an increase in the degradation of the cable and connections seen by visual means may be monitored, trended as appropriate, and evaluated as input to make a determination of remaining service life.

(6) Acceptance Criteria - The Non-EQ Insulated Cables and Connections Inspection Program consists of visual inspections for degradation of cable jackets and connections due to aging. Acceptance criteria are based on the cable and connection insulation service life. The service life evaluation of the insulation material includes consideration of the material's mechanical and electrical properties and their performance in ambient environments under plant operational conditions of temperature, radiation, and humidity as well as ohmic heating effects. The results of the Non-EQ Insulated Cables and Connections Inspection Program will serve as input into the service life evaluation as well as the rate of visually detectable degradation through monitoring and trending from the baseline taken prior to the period of extended operation.

(7) Corrective Actions – The VCSNS Corrective Action Program is utilized as applicable to provide specific corrective and confirmatory actions.

(8) Confirmation Process – Engineering reviews the inspections for completeness and acceptability and the corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls –** The Non-EQ Insulated Cables and Connections Inspection Program will be implemented in accordance with controlled station and work processes.

B.2.9.1 Operating Experience

The Non-EQ Insulated Cables and Connections Inspection Program is a new inspection activity for which there is no operating experience. Effective and proven visual inspection techniques will be selected for use in performing the inspections. Lessons learned during the performance of the inspections, experience gained and shared by other utilities, and other inspection techniques developed in the industry will be considered as proposed enhancements to the program so that the effects of aging will continue to be adequately managed.

B.2.9.2 Conclusion

The Non-EQ Insulated Cables and Connections Inspection Program will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.2.10 BURIED PIPING AND TANKS INSPECTION

The Buried Piping and Tanks Inspection will be consistent with XI.M34, *Buried Piping and Tanks Inspection*, as identified in NUREG –1801 prior to the period of extended operation.

The Buried Piping and Tanks Inspection is a new inspection activity that will manage loss of material due to crevice, galvanic, general, pitting, and microbiologically influenced corrosion (MIC) on the external surfaces of components exposed to an underground environment. The Buried Piping and Tanks Inspection contains elements of a condition-monitoring program and a prevention program.

(1) Scope - The Buried Piping and Tanks Inspection is applicable to carbon steel, cast iron and ductile iron components exposed to an underground environment in the following systems:

- Diesel Generator Services – carbon steel fuel oil pipe and tanks
- Emergency Feedwater - carbon steel pipe
- Fire Service – ductile iron pipe, and cast iron hydrants and valve bodies
- Service Water – carbon steel pipe and couplings

(2) Preventive Actions – In accordance with standard industry practice, underground components were coated and wrapped during installation to prevent them from directly contacting the soil environment. Otherwise, no actions are taken as part of the Buried Piping and Tanks Inspection to prevent aging effects or mitigate aging degradation.

(3) Parameters Monitored or Inspected – As part of the Buried Piping and Tanks Inspection, the condition of coatings and wrappings will be determined by visual inspection whenever buried components are excavated for maintenance or for other reasons.

(4) Detection of Aging Effects – The results of previous inspections, as discussed below under Operating Experience, indicate a very slow (or negligible) rate of wall thinning. Since the process of excavation itself can damage protective coatings and wrappings, a specific inspection frequency for buried components is not warranted. If buried components are excavated for maintenance or for other reasons, the integrity of their coatings and wrappings will be evaluated. If the coatings or wrappings are damaged of removed as part of the maintenance activity the underlying metal will be visually inspected for degradation.

The Buried Piping and Tanks Inspection will detect loss of material prior to a loss of component intended function.

(5) **Monitoring and Trending** - No actions are taken as part of the Buried Piping and Tanks Inspection to trend inspection results. However, the results of an inspection may indicate the need for additional inspections to be performed.

(6) **Acceptance Criteria** - The acceptance criteria for the Buried Piping and Tanks Inspection is no unacceptable degradation of coatings and wrappings that could result in loss of material and therefore a loss of component intended function, as determined by engineering evaluation.

(7) **Corrective Actions** – If the results of the Buried Piping and Tanks Inspection are not acceptable, as determined by engineering evaluation, then corrective actions are taken to repair or replace the affected items. The corrective action includes a determination of whether additional inspections or programmatic oversights are required. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process** – Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls –** The Buried Piping and Tanks Inspection will be implemented in accordance with controlled station procedures and work processes.

B.2.10.1 Operating Experience

The Buried Piping and Tanks Inspection is a new inspection activity. There is some operating experience with buried piping and tanks at VCSNS. During the EDSFI evaluation, the Cathodic Protection System was found to provide inadequate protection to the diesel generator fuel oil storage tanks and associated underground piping. As a result, an ultrasonic examination of the fuel oil storage tanks and associated piping was performed. Each tank was inspected at 102 locations, evenly distributed over the entire surface area, and no significant corrosion or age-related degradation was found. The tank inspection indicated a very slow (or negligible) rate of wall thinning. Approximately 35 feet of fuel oil piping was inspected and found to be in good condition with no corrosion identified.

B.2.10.2 Conclusion

The Buried Piping and Tanks Inspection will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.2.11 INSPECTIONS FOR MECHANICAL COMPONENTS

There is no NUREG-1801 item addressing this program. This is a plant specific program.

Inspections for Mechanical Components is a new inspection activity that will manage loss of material due to galvanic, general, and pitting corrosion and cracking due to radiation and thermal embrittlement for the external surfaces of mechanical components within the scope of license renewal that are exposed to ambient conditions. Inspections for Mechanical Components is a condition monitoring program.

(1) Scope - Inspections for Mechanical Components will manage the relevant aging effects for mechanical components constructed of carbon steel, low alloy steel, and other susceptible materials. All or portions of the following mechanical systems contain components/component types subject to the aging effects managed by this program:

- Air Handling (HVAC)
- Auxiliary Boiler Steam and Feedwater
- Auxiliary Coolant (Closed Loop) / CRDM Cooling Water
- Boron Recycle
- Building Services
- Chemical and Volume Control
- Chilled Water
- Component Cooling
- Condensate
- Demineralized Water – Nuclear Service
- Diesel Generator Services
- Emergency Feedwater
- Extraction Steam
- Feedwater
- Fire Service
- Gaseous Waste Processing
- Gland Sealing Steam
- Hydrogen Removal
- Instrument Air Supply
- Liquid Waste Processing
- Local Ventilation and Cooling
- Main Steam
- Main Steam Dump
- Nitrogen Blanketing
- Nuclear Sampling
- Radiation Monitoring
- Reactor Building Leak Rate Testing
- Reactor Building Spray
- Reactor Coolant
- Reactor Makeup Water Supply
- Residual Heat Removal
- Safety Injection
- Service Water
- Spent Fuel Cooling
- Station Service Air
- Steam Generator Blowdown
- Thermal Regeneration

(2) Preventive Actions - No actions are taken as part of Inspections for Mechanical Components to prevent aging effects or to mitigate aging degradation.

(3) **Parameters Monitored or Inspected -** Inspections for Mechanical Components involves a visual examination of the exposed external surfaces of mechanical components for loss of material or cracking.

(4) **Detection of Aging Effects -** In accordance with information provided in Monitoring and Trending below, Inspections for Mechanical Components will detect loss of material and cracking prior to loss of component intended function. Pitting is a concern in locations where components are insulated and internal system fluid temperatures are below the ambient temperature conditions.

(5) **Monitoring and Trending -** Inspections of material surface condition will be performed and documented in accordance with station procedures. Following a baseline inspection, the frequency of the inspections will be determined based on inspection results and industry experience.

(6) **Acceptance Criteria -** The acceptance criteria for Inspections for Mechanical Components are no unacceptable visible indication of loss of material or cracking. An indication of a rate of deterioration due to loss of material or cracking that could cause the component to fail its intended function prior to its next scheduled inspection, as determined by engineering evaluation, is considered unacceptable.

(7) **Corrective Actions -** If the results of the inspections for Mechanical Components are not acceptable, as determined by engineering evaluation, then corrective actions are taken to repair or replace the effective components. The corrective action includes determination of whether additional inspections or programmatic oversights are required.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Inspections for Mechanical Components will be implemented in accordance with controlled station procedures and work processes.

B.2.11.1 Operating Experience

The Inspections for Mechanical Components is a new inspection activity. There is VCSNS operating experience with the detection of aging effects on exposed external surfaces of components. An instance of pitting below insulation in the Chilled Water System was identified and repaired. Several instances of leakage in the Chilled Water System have been identified by surveillance procedures. The leakage was evaluated and repaired under the Corrective Action Program.

B.2.11.2 Conclusion

Inspections for Mechanical Components will provide reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

B.2.12 HEAT EXCHANGER INSPECTIONS

The Heat Exchanger Inspections will be consistent with XI.M32, *One-Time Inspection*, and XI.M33, Selective Leaching of Materials, as identified in NUREG-1801 prior to the period of extended operation.

The Heat Exchanger Inspections is a new one-time inspection activity that will detect and characterize loss of material due to selective leaching and erosion-corrosion as well as heat exchanger fouling due to particulates for heat exchanger components in a treated water environment. The Heat Exchanger Inspections will be performed prior to the period of extended operation.

(1) Scope - The Heat Exchanger Inspections is applicable to copper, copper-nickel, and brass heat exchanger components (as well as brass thermowells) exposed to a treated water environment in various systems. These systems are the Air Handling System, the Component Cooling System, the Chemical and Volume Control System, the Diesel Generator System, the Emergency Feedwater System, the Chilled Water System, and the Local Ventilation and Cooling System.

(2) Preventive Actions - No actions are taken as part of the Heat Exchanger Inspections to prevent aging effects or to mitigate aging degradation.

(3) Parameters Monitored or Inspected - The parameters inspected by the Heat Exchanger Inspections are wall thickness as a measure of loss of material, material hardness as a measure of selective leaching, and visual evidence of loss of material, heat exchanger fouling or other age-related degradation.

(4) Detection of Aging Effects - The Heat Exchanger Inspections will use a combination of proven volumetric and visual examination techniques at sample locations in the various heat exchangers determined by engineering evaluation to be most susceptible to the applicable aging effects. If no parameters are known that would distinguish the susceptible locations, sample locations will be selected based on accessibility and radiological concerns, and the results will be applied to the associated components. The inspection will include a Brinnell Hardness Test or an equivalent test on a sample of susceptible components in order to characterize a reduction of material hardness (loss of material) due to selective leaching.

The Heat Exchanger Inspections will detect the presence and extent of any loss of material and heat exchanger fouling prior to a loss of component intended function.

Inspection locations for heat exchange fouling should focus on heat exchanger components having an intended function of heat transfer and which are normally in a standby condition with no flow.

(5) **Monitoring and Trending -** No actions are taken as part of the Heat Exchanger Inspections to trend inspection results. This is a one-time inspection used to determine if further actions are required.

(6) **Acceptance Criteria -** The acceptance criteria for the Heat Exchanger Inspections is no unacceptable loss of material or heat exchanger fouling of subject components that could result in a loss of the component intended function(s) as determined by engineering evaluation.

(7) **Corrective Actions -** If engineering evaluation determines that continuation of the aging effects will not cause a loss of component intended function(s) under the 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 the engineering evaluation determines that additional information is required to more fully characterize 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. The VCSNS Corrective Action Program is utilized to provide specific corrective and confirmatory actions.

(8) **Confirmation Process -** Engineering reviews the inspection results for completeness and acceptability. The corrective action processes ensure that degraded conditions are tracked and corrected in a timely manner. Engineering reviews the previous inspection reports to ensure implementation of recommended corrective actions and to determine their effectiveness.

(9) **Administrative Controls -** The Heat Exchanger Inspections will be implemented in accordance with controlled station procedures and work processes.

B.2.12.1 Operating Experience

The Heat Exchanger Inspections is a new one-time inspection for which there is no operating experience.

B.2.12.2 Conclusion

Implementation of the Heat Exchanger Inspections will either verify that there are no aging effects requiring management for the subject components or appropriate corrective actions will be taken so that the component intended functions will be ensured for the period of extended operations.

B.3.0 TLAA SUPPORT ACTIVITIES

B.3.1 ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM

The Environmental Qualification (EQ) Program is consistent with X.E1, *Environmental Qualification (EQ) of Electric Components*, as identified in NUREG-1801.

Prior to the period of extended operation, the equipment subject to the provisions of 10 CFR 50.49 will be re-evaluated for 60 years of installation. Parameters to be evaluated include:

- Thermal Considerations - The component qualification temperatures will be evaluated for 60 years using the Arrhenius method. Components not meeting a 60 year qualified life will be replaced prior to expiration of qualified life.
- Radiation Considerations - The bounding 60 year radiation dose qualification values for all EQ components will be compared to the dose values typically established by manufacturers in qualification testing. Any EQ components qualification dose level not enveloping the 60 year radiation environment will be evaluated by available means, shielded, or replaced prior to expiration of its radiation qualified life.
- Wear Cycle Aging Considerations - Wear cycle aging is a factor for some equipment in the VCSNS EQ Program. In cases where wear cycle aging is a credible aging mechanism, wear cycles will be evaluated and/or controlled through the end of the period of extended operation.

B.3.1.1 Operating Experience

The VCSNS EQ Program includes consideration of operating experience, both at VCSNS and in the industry, to modify the qualification bases and conclusions. This includes data on specific components and materials, data on aging limits, and new test data from manufacturers, industry groups, or the NRC. VCSNS is actively involved in the nuclear industry technical groups on EQ, such as NEI, NUGEQ, and various EPRI programs, including the License Renewal Electrical Working Group (under NEI).

B.3.1.2 Conclusion

The EQ Program has been demonstrated to be capable of maintaining the qualification of components within the scope of 10 CFR 50.49. The EQ Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.2 THERMAL FATIGUE MANAGEMENT PROGRAM

The Thermal Fatigue Management Program is consistent with X.M1, *Metal Fatigue of Reactor Coolant Pressure Boundary*, as identified in NUREG-1801 with the enhancements specified in the following table.

The following enhancements will be incorporated into the Thermal Fatigue Management Program prior to the period of extended operation:

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NUREG-1801 Program	Attributes	Enhancement
X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary	1.) Scope	Incorporate the new guidance in EPRI Report MRP-47 [Reference B-12] for the environmental effects of the reactor coolant environment on fatigue into the VCSNS program.
	6.) Acceptance Criteria	Revise the acceptance criteria to account for the environmental effects on fatigue.

B.3.2.1 Operating Experience

The Thermal Fatigue Management Program includes consideration of operating experience, both at VCSNS and in the industry. Thermal fatigue transients have been tracked since operation began at VCSNS. Operating experience has demonstrated that the program continues to monitor plant transients and track the accumulation of those transients consistent with the requirements in VCSNS Technical Specification 5.7. Industry issues identified after plant start-up have been incorporated into the Thermal Fatigue Management Program.

B.3.2.2 Conclusion

The Thermal Fatigue Management Program has been demonstrated to be capable of managing the thermal fatigue basis to preclude cracking due to thermal fatigue. The Thermal Fatigue Management Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.3.3 TENDON SURVEILLANCE PROGRAM

The Tendon Surveillance Program is consistent with X.S1, *Concrete Containment Tendon Prestress*, as identified in NUREG-1801.

B.3.3.1 Operating Experience

A brief history of the Tendon Surveillance Program is as follows:

- Reactor Building concrete placement was completed In August 1976.
- Vertical tendon stressing was completed In March 1979.
- The Structural Integrity Test (SIT) was performed in January 1981.
- The first tendon surveillance was performed during March and April 1982.
- The plant operating license was issued in August 1982.
- The second tendon surveillance was performed during October – December 1983.
- The third tendon surveillance was performed during November and December 1985.
- The test results from the first three surveillance's indicated that the wire relaxation force losses in the tendon system were greater than that which were predicted during design. Consequently in June 1988, the predicted wire relaxation force losses were increased from 8.5% to 12.8%.
- The fourth period (10^{th} year) tendon surveillance was performed during January – April 1990. In addition, the vertical tendons were retensioned because the previous surveillance data indicated that the vertical tendon forces would be below the Technical Specifications minimum prior to the fifth period surveillance.
- The fifth period (15^{th} year) tendon surveillance was performed during March – April 1996.
- The sixth period (20^{th} year) tendon surveillance was performed during September – November 2000.

A review of the non-conformances (NCNs) written to address programmatic and problematic deficiencies with the Tendon Surveillance Program indicates that there have been no adverse trends associated with aging that are not inherent to this type of post tensioning system.

A non-conformance (NCN) was identified to address the collection of water due to in-leakage into the Auxiliary Building tendon sump area to a depth that submerged a tendon end cap. The water level in the pit was reduced to a level below the tendon end cap. During RF-12 the tendon end cap was removed for inspection and no free water was found. Grease samples (analyzed for entrained moisture) and the tendon components (inspected for corrosion) were found to be acceptable. As a corrective action, Operations added the Auxiliary

Building tendon sump area to their trend logs and will request facilities to drain the area if the water level in the area approaches the level of the tendon end cover.

The surveillance reports for the past three surveillance periods [Fourth (1990), Fifth (1996) and Sixth (2000)] have each concluded that no abnormal degradation of the post tensioning system has occurred at VCSNS.

B.3.3.2 Conclusion

The Tendon Surveillance Program has been demonstrated to be capable of maintaining the Reactor Building dome, vertical, and hoop tendons above the minimum required prestressing forces. The Tendon Surveillance Program provides reasonable assurance that the aging effects will be managed such that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.4.0 REFERENCES

B-1	NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, April, 2001.
B-2	NUREG-1801, Generic Aging Lessons Learned (GALL) Report, April 2001.
B-3	SCE&G Letter RC-01-0155, S.A. Byrne to USNRC Document Control Desk, Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, August 31, 2001.
B-4	SCE&G Letter RC-02-0055, S.A. Byrne to USNRC Document Control Desk, Response to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, April 3, 2002.
B-5	NRC Inspection Report (IR) 50-395 / 98-01, March 20, 1998 for VCSNS.
B-6	NRC Inspection Report (IR) 50-395 / 93-09, April 21, 1993 for VCSNS.
B-7	Westinghouse Topical Report WCAP-14422, Licensing Renewal Evaluation: Aging Management for Reactor Coolant System Supports, Revision 2-A, December 2000.
B-8	SCE&G Letter RC-98-0207, G.J. Taylor to USNRC Document Control Desk, Response to Generic Letter 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment, November 11, 1998.
B-9	NRC Inspection Report (IR) 50-395 / 82-28, April 19, 1982.
B-10	SCE&G Letter dated January 31, 1990, Response to Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment.
B-11	Letter, Christopher Grimes (NRC) to Douglas Walters(NEI), License Renewal Issue No. 98-0085, “Reactor Vessel Surveillance Program”, December 3, 1999.
B-12	EPRI Final Report MRP-47, Materials Reliability Program Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47), Revision 0, October 2001.
B-13	NRC letter to G. J. Taylor (dated February 27, 1998), Service Water Pond Dam Safety Inspection Results for Virgil C. Summer Nuclear Station.

B-14	USNRC Letter NL-356-99, K.R. Cotton to G.J. Taylor, Generic Letter 97-01, “Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations”: Review of the Responses for the Virgil C. Summer Nuclear Station, December 17, 1999.

APPENDIX C - COMMODITY GROUPS (OPTIONAL)

Appendix C is not being used in the Application to Renew the Operating License of the Virgil C. Summer Nuclear Station.

APPENDIX D - TECHNICAL SPECIFICATION CHANGES

10 CFR 54.22 requires that an application for license renewal include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation. No changes to the Virgil C. Summer Nuclear Station Technical Specifications are necessary in that regard.

APPENDIX E - ENVIRONMENTAL REPORT

The Environmental Report for Virgil C. Summer Nuclear Station is contained in a separate document entitled "**Environmental Report for License Renewal, Virgil C. Summer Nuclear Station**."