# Seabrook Station
# License Renewal Application

NEXTera ENERGY SEABROOK

VOLUME III

## 3.5 AGING MANAGEMENT OF SYSTEMS, STRUCTURES, AND COMPONENT SUPPORTS

### 3.5.1 INTRODUCTION

This section provides the results of the aging management review for those components identified in Section 2.4, Structures and Component Supports, as being subject to aging management review. The Structures and Component Supports or portions of Structures, Component Supports and commodities, which are addressed in this section, are described in the indicated sections.

- Buildings, Structures Within License Renewal (2.4.1)
- Containment Structures (2.4.2)
- Fuel Handling and Overhead Cranes (2.4.3)
- Miscellaneous Yard Structures (2.4.4)
- Primary Structures (2.4.5)
- Supports (2.4.6)
- Turbine Building (2.4.7)
- Water Control Structures (2.4.8)

Table 3.5.1, Summary of Aging Management Evaluations for Structures and Component Supports, provides a summary comparison of the Seabrook Station aging management activities with the aging management activities evaluated in NUREG-1801 for Structures and Component Supports. Text addressing summary items requiring further evaluation is provided in Section 3.5.2.2.

### 3.5.2 RESULTS

The following tables summarize the results of the aging management review for Structures and Component Supports:

| | |
|---|---|
| Table 3.5.2-1 | Summary of Aging Management Evaluation - Buildings, Structures Within License Renewal |
| Table 3.5.2-2 | Summary of Aging Management Evaluation - Containment Structures |
| Table 3.5.2-3 | Summary of Aging Management Evaluation - Fuel Handling and Overhead Cranes |
| Table 3.5.2-4 | Summary of Aging Management Evaluation - Miscellaneous Yard Structures |
| Table 3.5.2-5 | Summary of Aging Management Evaluation - Primary Structures |
| Table 3.5.2-6 | Summary of Aging Management Evaluation - Supports |

Table 3.5.2-7 Summary of Aging Management Evaluation - Turbine Building

Table 3.5.2-8 Summary of Aging Management Evaluation - Water Control Structures

The materials of construction, service environments, aging effects requiring management, and credited aging management programs are provided for each of the system, structures and component supports system in the following Sections:

- Buildings, Structures Within License Renewal (3.5.2.1.1)
- Containment Structures (3.5.2.1.2)
- Fuel Handling and Overhead Cranes (3.5.2.1.3)
- Miscellaneous Yard Structures (3.5.2.1.4)
- Primary Structures (3.5.2.1.5)
- Supports (3.5.2.1.6)
- Turbine Building (3.5.2.1.7)
- Water Control Structures (3.5.2.1.8)

## 3.5.2.1 Materials, Environments, Aging Effects Requiring Management and Aging Management Programs

### 3.5.2.1.1 Buildings, Structures Within License Renewal

**Materials**

The materials of construction for the Buildings, Structures Within License Renewal components are:

- Concrete
- Concrete Block
- Fluorogold
- Rock
- Roofing
- Stainless Steel
- Steel

**Environments**

The Buildings, Structures Within License Renewal components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)

- Air - Outdoor (External)
- Ground Water / Soil (External)
- Raw Water (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Buildings, Structures Within License Renewal components require management:

- Cracking
- Cracking, Loss of Bond, Loss of Material (spalling, scaling)
- Expansion and Cracking
- Fretting or Lockup
- Increase in Porosity and Permeability, Cracking, Loss of Material (spalling, scaling)
- Increase in Porosity and Permeability, Loss of Strength
- Loss of Material
- Loss of Material, Loss of Form
- Separation, Environmental Degradation, Water in Leakage

The following aging management programs manage the aging effects for the Buildings, Structures Within License Renewal components:

- Fire Protection Program (B.2.1.15)
- Structures Monitoring Program (B.2.1.31)

Table 3.5.2-1, Summary of Aging Management Evaluation - Buildings, Structures Within License Renewal, summarizes the results of the aging management review for the Buildings, Structures within License Renewal.

#### 3.5.2.1.2 Containment Structures

**Materials**

The materials of construction for the Containment Structures components are:

- Aluminum
- Concrete
- Elastomer
- Glass
- Roofing
- Stainless Steel

- Steel

**Environments**

The Containment Structures components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)
- Air - Outdoor (External)
- Air - With Borated Water Leakage (External)
- Groundwater/Soil (External)
- Raw Water (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Containment Structures components require management:

- Cracking
- Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)
- Expansion and Cracking
- Increase in Porosity and Permeability, Cracking, Loss of Material (spalling, scaling)
- Increase in Porosity and Permeability, Loss of Strength
- Increased Hardness, Shrinkage and Loss of Strength
- Loss of Material and Cracking
- Loss of material
- Loss of Sealing, Leakage Through Containment
- Separation, Environmental Degradation, Water in Leakage

**Aging Management Programs**

The following aging management programs manage the aging effects for the Containment Structures components:

- ASME Section XI, Subsection IWE Program (B.2.1.27)
- ASME Section XI, Subsection IWL Program (B.2.1.28)
- Boric Acid Corrosion Program (B.2.1.4)
- Fire Protection Program (B.2.1.15)
- Structures Monitoring Program (B.2.1.31)

Table 3.5.2-2, Summary of Aging Management Evaluation - Containment Structures, summarizes the results of the aging management review for the Containment Structures

#### 3.5.2.1.3 Fuel Handling and Overhead Cranes

**Materials**

The materials of construction for the Fuel Handling and Overhead Cranes components are:

- Steel

**Environments**

The Fuel Handling and Overhead Cranes components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)
- Air - With Borated Water Leakage (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Fuel Handling and Overhead Cranes components require management:

- Loss of material

**Aging Management Programs**

The following aging management programs manage the aging effects for the Fuel Handling and Overhead Cranes components:

- Boric Acid Corrosion Program (B.2.1.4)
- Inspection of Heavy Load and Light Load (Related to Refueling) Handling Systems Program (B.2.1.13)
- Structures Monitoring Program (B.2.1.31)

Table 3.5.2-3, Summary of Aging Management Evaluation - Fuel Handling and Overhead Cranes, summarizes the results of the aging management review for the Fuel Handling and Overhead Cranes.

#### 3.5.2.1.4 Miscellaneous Yard Structures

**Materials**

The materials of construction for the Miscellaneous Yard Structures components are:

- Aluminum
- Concrete
- Concrete Block

- Elastomer
- Roofing
- Stainless Steel
- Steel

**Environments**

The Miscellaneous Yard Structures components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)
- Air - Outdoor (External)
- Ground Water / Soil (External)
- Raw Water (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Miscellaneous Yard Structures components require management:

- Crack Initiation and Growth
- Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)
- Cracking
- Expansion and Cracking
- Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling)
- Increased hardness, shrinkage and loss of strength
- Loss of Material Cracking
- Loss of Material
- Separation, Environmental Degradation, Water In Leakage

**Aging Management Programs**

The following aging management programs manage the aging effects for the Miscellaneous Yard Structures components:

- Fire Protection Program (B.2.1.15)
- Structures Monitoring Program (B.2.1.31)

Table 3.5.2-4, Summary of Aging Management Evaluation - Miscellaneous Yard Structures, summarizes the results of the aging management review for the Miscellaneous Yard Structures.

### 3.5.2.1.5 Primary Structures

**Materials**

The materials of construction for the Primary Structures components are:

- Aluminum
- Concrete
- Elastomer
- Lubrite
- Non-Metallic Fire Proofing
- Roofing
- Stainless Steel
- Steel

**Environments**

The Primary Structures components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)
- Air - Outdoor (External)
- Air - With Borated Water Leakage (External)
- Ground Water / Soil (External)
- Raw Water (External)
- Treated Borated Water (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Primary Structures components require management:

- Cracking
- Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)
- Expansion and Cracking
- Increase in Porosity and Permeability, Cracking, Loss of Material (spalling, scaling)
- Increase in Porosity and Permeability, Loss of Strength
- Increased Hardness, Shrinkage and Loss of Strength
- Loss of Material, Cracking
- Loss of material

- Loss of Mechanical Function
- Separation, Environmental Degradation, Water in Leakage

**Aging Management Programs**

The following aging management programs manage the aging effects for the Primary Structures components:

- Boric Acid Corrosion Program (B.2.1.4)
- Fire Protection Program (B.2.1.15)
- Structures Monitoring Program (B.2.1.31)
- Water Chemistry Program (B.2.1.2)

Table 3.5.2-5, Summary of Aging Management Evaluation - Primary Structures, summarizes the results of the aging management review for the Primary Structures.

#### 3.5.2.1.6 Supports

**Materials**

The materials of construction for the Support components are:

- Aluminum
- Boral
- Concrete
- Elastomer
- Lubrite
- Stainless Steel
- Steel

**Environments**

The Support components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)
- Air - Outdoor (External)
- Air - With Borated Water Leakage (External)
- Raw Water (External)
- Treated Borated Water (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Support components require management:

- Cracking
- Loss of material
- Loss of Mechanical Function/ Corrosion, Distortion, Dirt, etc.
- Reduction in Concrete Anchor Capacity
- Reduction of Neutron Absorbing Capability and Loss of Material
- Reduction or Loss of Isolation Function

**Aging Management Programs**

The following aging management programs manage the aging effects for the Support components:

- ASME Section XI, Subsection IWF Program (B.2.1.29)
- Boral Monitoring Program (B.2.2.2)
- Boric Acid Corrosion Program (B.2.1.4)
- Structures Monitoring Program (B.2.1.31)
- Water Chemistry Program (B.2.1.2)

Table 3.5.2-6, Summary of Aging Management Evaluation - Supports, summarizes the results of the aging management review for the Supports.

#### 3.5.2.1.7 Turbine Building

**Materials**

The materials of construction for the Turbine Building components are:

- Aluminum
- Concrete
- Concrete Block
- Elastomer
- Roofing
- Steel

**Environments**

The Turbine Building components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)
- Air - Outdoor (External)
- Raw Water (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Turbine Building components require management:

- Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)/Corrosion of Embedded Steel
- Cracking
- Expansion and Cracking
- Increased hardness, shrinkage and loss of strength
- Increase in Porosity and Permeability, Cracking, Loss of Material (spalling, scaling)
- Loss of Material
- Loss of Material, Cracking
- Separation, Environmental Degradation, Water In leakage

**Aging Management Programs**

The following aging management programs manage the aging effects for the Turbine Building components:

- Fire Protection Program (B.2.1.15)
- Structures Monitoring Program (B.2.1.31)

Table 3.5.2-7, Summary of Aging Management Evaluation - Turbine Building, summarizes the results of the aging management review for the Turbine Building.

### 3.5.2.1.8 Water Control Structures

**Materials**

The materials of construction for the Water Control Structures components are:

- Concrete
- Elastomer
- Roofing
- Steel

**Environments**

The Water Control Structures components are exposed to the following environments:

- Air - Indoor Uncontrolled (External)

- Air - Outdoor (External)
- Ground Water / Soil (External)
- Raw Water (External)

**Aging Effects Requiring Management**

The following aging effects associated with the Water Control Structures components require management:

- Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)/Corrosion of Embedded Steel
- Expansion and Cracking
- Increased hardness, shrinkage and loss of strength
- Increase in Porosity and Permeability, Cracking, Loss of Material (spalling, scaling)
- Concrete Cracking and Spalling
- Increase in Porosity and Permeability, Loss of Strength
- Loss of Material
- Loss of Material, Cracking
- Separation, Environmental Degradation, Water In leakage

**Aging Management Programs**

The following aging management programs manage the aging effects for the Water Control Structures components:

- Fire Protection Program (B.2.1.15)
- Structures Monitoring Program (B.2.1.31)

Table 3.5.2-8, Summary of Aging Management Evaluation - Water Control Structures, summarizes the results of the aging management review for the Water Control Structures.

### 3.5.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801 for Structures and Component Supports

NUREG-1801 indicates that further evaluation is necessary for certain aging effects and other issues. Section 3.5.2.2 of NUREG-1800 discusses these aging effects and other issues that require further evaluation. The following sections, numbered corresponding to the discussions in NUREG-1800, explain the Seabrook Station approach to these areas requiring further evaluation.

#### 3.5.2.2.1 PWR and BWR Containments

##### 3.5.2.2.1.1 Aging of Inaccessible Concrete Areas

NUREG-1800 item 3.5.2.2.1.1 relates to potential aging of inaccessible concrete areas in concrete and steel containments due to aggressive chemical attack and corrosion of embedded steel. NUREG-1801 indicates that further evaluation is necessary if the environment is aggressive.

The Seabrook Station containment inaccessible and accessible concrete areas are designed in accordance with American Concrete Institute (ACI) Specification 318-71. The resulting reinforced concrete is dense, with low permeability.

Degradation due to aggressive chemical attack is applicable to Seabrook Station. Aggressive chemical attack only becomes significant when environmental conditions exceed threshold values (Chlorides > 500 ppm, Sulfates >1500 ppm, or pH < 5.5). Seabrook Station is not located in areas exposed to sulfate attack, nor is it located near industrial plants whose emissions could alter environmental parameters, but is exposed to chloride attack. Groundwater analyses confirm that the Seabrook Station site groundwater is aggressive. Testing performed from November 2008 to September 2009 found pH values between 5.8 and 7.5, chloride values between 19 ppm and 3900 ppm, and sulfate values between 10 ppm and 100 ppm. Resistance to mild acid attack is enhanced through the use of dense concrete that has low permeability and a low water to cement ratio. Seabrook Station Structures Monitoring Program, B.2.1.31 will perform concrete testing and rebar inspection to determine the effects f the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

Corrosion of embedded steel becomes significant if environmental conditions are found to be aggressive. As noted above, Seabrook Station groundwater analyses confirm that the Seabrook Station site groundwater is aggressive. Additionally, 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 the Seabrook Station concrete structures generally prevents corrosion of embedded steel from occurring. As a result, corrosion of embedded steel in the Seabrook Station containment building is managed by the Seabrook Station ASME Section XI, Subsection IWL Program, B.2.1.28.

Seabrook Station ASME Section XI, Subsection IWL Program, B.2.1.28 inspections are conducted per Seabrook Station procedures which provides instructions to perform visual examination of the concrete surfaces of the primary containment in accordance with requirements of IWL-2500. Examinations performed under visual examination (VT-3C) examine concrete surfaces for evidence of damage or degradation shown below:

- Chemical attack, abrasion or erosion sufficient to expose coarse aggregate.
- Water flowing from, or on, the surface of the concrete (except basement Annulus).
- Scaling and/or disintegration sufficient to expose coarse aggregate.
- Cracks, spalls, voids or popouts.
- Efflorescence, exudation and/or encrustation.
- Discoloration indicative of corrosion of embedded steel.
- Exposure of reinforcing steel.
- Cracking, blistering and/or peeling of coatings.

**3.5.2.2.1.2 Cracks and Distortion due to Increased Stress Levels from Settlement; Reduction of Foundation Strength, Cracking and Differential Settlement due to Erosion of Porous Concrete Subfoundations, if Not Covered by Structural Monitoring Program**

NUREG-1800 item 3.5.2.2.1.2 indicates that cracks due to increased stress levels from settlement could occur in Pressurized Water Reactor (PWR) containments. Additionally, reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations could occur in PWR containments. For plants that rely on a dewatering system, NUREG-1801 recommends verification of the continued functionality of the dewatering system during the period of extended operation. For all plants, NUREG-1801 recommends no further evaluation if these issues are managed by the applicant's Structural Monitoring Program.

Seabrook Station does not rely on a dewatering system for control of settlement.

Seabrook Station structures are founded on sound bedrock, fill concrete, or consolidated backfill and do not have any potential areas of settlement or displacement. Similarly, gradation requirements, compaction criteria and compaction test for engineered fill ensure a foundation material that will support the design loads with negligible settlement. The concrete foundations at Seabrook Station are not constructed of porous concrete and are not subject to flowing water.

Therefore, both cracks and distortion due to increased stress levels from settlement, and reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations for the containment, are not aging effects requiring management for the period of extended operation.

#### 3.5.2.2.1.3 Reduction of Strength and Modulus of Concrete Structures due to Elevated Temperature

NUREG-1800 item 3.5.2.2.1.3 relates to reduction of strength and modulus of concrete due to elevated temperatures. NUREG-1801 recommends further evaluation of a plant-specific aging management program for any portion of the concrete containment components that exceed specified temperature limits, i.e., general area temperature greater than 150 °F and local area temperature greater than 200 °F.

Reduction of strength and modulus of concrete due to elevated temperatures is not applicable to Seabrook Station. Containment concrete degradation due to elevated temperatures is not applicable because no containment concrete structural components exceed the specified temperature limits. The containment structure cooling subsystem is designed to maintain the normal ambient air temperature in the containment structure at or below 120°F. The containment structure cooling subsystem also functions to prevent the concrete temperature in the area of the reactor supports from exceeding 150°F, and the neutron detector cavity from exceeding 135°F, during normal operation. If the pipe carries hot fluid, the space between the pipe and the sleeve is insulated to maintain the concrete temperature adjoining the embedded sleeve at or below 200° F during normal plant operation.

#### 3.5.2.2.1.4 Loss of Material due to General, Pitting and Crevice Corrosion

NUREG-1800 item 3.5.2.2.1.4 relates to loss of material due to general, pitting and crevice corrosion for steel elements of accessible and inaccessible areas of containments. The American Society of Mechanical Engineers (ASME) Section XI, Subsection IWE and Title 10 Code of Federal Regulations (CFR) Part 50 Appendix J Programs are recommended to manage these aging effects. NUREG-1801 recommends further evaluation of plant-specific programs to manage these aging effects for inaccessible areas if corrosion is significant.

Corrosion for inaccessible areas (e.g., embedded containment liner) is not expected for Seabrook Station because containment concrete in contact with the embedded containment liner at Seabrook Station was designed, constructed, and inspected in accordance with applicable ACI and American Society for Testing and Materials (ASTM) standards, which provide for a good quality, dense, well cured, and low permeability concrete. Design practices and procedural controls ensured that the concrete was consistent with the recommendations and guidance provided by ACI 201.2R.

The seismic isolation material between the fill mat and the containment liner is sealed at the mat surface level with caulk. This caulked joint is examined for signs of degradation during Seabrook Station ASME Section XI, Subsection IWL Program, B.2.1.28 inspections.

Nonetheless, the absence of concrete aging effects is confirmed by inspections performed per the Seabrook Station ASME Section XI, Subsection IWL Program, B.2.1.28.

#### 3.5.2.2.1.5 Loss of Pre-stress due to Relaxation, Shrinkage, Creep, and Elevated Temperature

NUREG-1800 item 3.5.2.2.1.5 relates to loss of pre-stress forces due to relaxation, shrinkage, creep, and elevated temperature for pre-stressed concrete containments. If loss of pre-stress is identified to be a Time-Limited Aging Analysis (TLAA), then it is required to be evaluated consistent with the 10 CFR 54.21(c).

The Seabrook Station Containment Building is not a pre-stressed concrete containment. Loss of pre-stress forces due to relaxation, shrinkage, creep, and elevated temperature for the containment is not applicable at Seabrook Station.

#### 3.5.2.2.1.6 Cumulative Fatigue Damage

NUREG-1800 item 3.5.2.2.1.6 relates to fatigue analyses of containment components including suppression pool steel shells (including welded joints) and penetrations (including penetration sleeves, dissimilar metal welds, and penetration bellows. If such fatigue analyses are determined to be Time-Limited Aging Analyses (TLAAs), then they are required to be evaluated consistent with the 10 CFR 54.21(c).

The evaluation of Seabrook Station containment penetrations that experience significant cyclic loading is addressed separately in Section 4.7, "*Penetration Load Cycles*".

Fatigue analyses for the Seabrook Station containment liner plates are not part of the current licensing basis and therefore do not meet the definition of a TLAA as based on 10 CFR 54.3.

#### 3.5.2.2.1.7 Cracking due to Stress Corrosion Cracking (SCC)

NUREG-1800 item 3.5.2.2.1.7 relates to cracking due to stress corrosion cracking of stainless steel penetration sleeves, penetration bellows, and dissimilar metal welds. Further evaluation is recommended to ensure that this aging effect is adequately managed.

The Seabrook Station Aging Management Review (AMR) results conclude that cracking due to Stress Corrosion Cracking (SCC) is not an aging effect requiring management for Seabrook Station stainless steel containment penetration sleeves, bellows, and dissimilar metal welds. Both high temperature (> 140 °F) and exposure to an aggressive environment are required for SCC to be applicable. At Seabrook Station, these two conditions are not simultaneously present for any stainless steel penetration sleeves, bellows, or dissimilar metal welds. Further, reviews of Seabrook Station plant-specific operating experience did not identify any Stress Corrosion Cracking (SCC) of these components.

#### 3.5.2.2.1.8 Cracking due to Cyclic Loading

NUREG-1800 item 3.5.2.2.1.8 relates to cracking due to cyclic loading in shells and penetrations. Existing programs include the ASME Section XI, Subsection IWE and 10 CFR 50 Appendix J. However, NUREG-1801 recommends further evaluation, noting that visual examinations implemented by these programs may not have the ability to detect fine cracks that may result from cracking due to cyclic loading.

The Seabrook Station Aging Management Review (AMR) results conclude that cracking due to cyclic loading for containment components without Current Licensing Basis (CLB) fatigue analyses is not an aging effect requiring management. These components are designed to withstand operating stress levels and as such, cracking due to cyclic loading is unlikely to occur. Further, reviews of Seabrook Station operating experience did not identify any events related to cyclic loading induced cracking of containment components.

This subsection also lists components associated with Boiling Water Reactor (BWR) primary containment that require aging management for crack initiation and growth due to Stress Corrosion Cracking (SCC). These components are not applicable to Seabrook Station, a Pressurized Water Reactor (PWR).

#### 3.5.2.2.1.9 Loss of Material (Scaling, Cracking and Spalling) due to Freeze-Thaw

NUREG-1800 item 3.5.2.2.1.9 relates to loss of material (scaling, cracking, and spalling) due to freeze-thaw in concrete containments. ASME Section XI, Subsection IWL program is recommended to manage this aging effect. However, NUREG-1801 recommends further evaluation of this aging effect for plants located in moderate to severe weathering conditions.

Loss of material due to freeze-thaw effects is not an aging effect requiring management for the Seabrook Station concrete containment. The Seabrook Station concrete containment is enclosed by a containment enclosure building and therefore is not exposed to severe weathering conditions. Loss of material (scaling, cracking, and spalling) due to freeze-thaw is only applicable to concrete containments exposed to severe weathering conditions.

### 3.5.2.2.1.10 Cracking due to Expansion and Reaction with Aggregate, and Increase in Porosity and Permeability due to Leaching of Calcium Hydroxide

NUREG-1800 item 3.5.2.2.1.10 relates to cracking due to expansion and reaction with aggregate, and to increase in porosity and permeability due to leaching of calcium hydroxide in concrete elements of containments. ASME Section XI, Subsection IWL is recommended to manage this aging effect. NUREG-1801 recommends further evaluation if the concrete was not constructed in accordance with the recommendations in ACI 201.2R.

At Seabrook Station, concrete was constructed equivalent to recommendations in ACI 201.2R.

Concrete aggregates used in Seabrook Station concrete structures were selected per ASTM C33, which uses ASTM C295 "*Petrographic Examination of Aggregates for Concrete*". Aggregates identified as potentially reactive were not used at Seabrook Station.

However, Seabrook Station conservatively manages cracking due to expansion and reaction with aggregates through the Seabrook Station ASME Section XI, Subsection IWL Program, B.2.1.28 and the Seabrook Station Structures Monitoring Program, B.2.1.31.

Loss of material due to leaching of calcium hydroxide is conservatively considered to be an aging effect requiring management for Seabrook Station. There have been indications of leaching in below grade concrete in Seabrook Station structures other than the Containment Building. 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, well-cured concrete with low permeability. These structures are designed in accordance with ACI 318 and constructed in accordance with ACI 301 and ASTM standards. However, due to the observed indications of leaching, Seabrook Station manages loss of material due to leaching of calcium hydroxide with the Seabrook Station ASME Section XI, Subsection IWL Program, B.2.1.28.

### 3.5.2.2.2 Safety-Related and Other Structures and Component Supports

### 3.5.2.2.2.1 Aging of Structures Not Covered by Structures Monitoring Program

1. Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling) Due to Corrosion of Embedded Steel for Groups 1-5, 7, 9 Structures.

   Concrete in inaccessible areas is evaluated for cracking, loss of bond, and loss of material due to corrosion of embedded steel. The NUREG-1801 description of an aggressive environment is pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm. Recent analysis of groundwater samples has shown an increase in chloride levels to above the threshold. Seabrook Station groundwater is currently classified as aggressive.

Therefore, cracking, loss of bond, and loss of material due to corrosion of embedded steel are aging effects requiring aging management for the period of extended operation.

When applicable, the condition of accessible areas may be used to evaluate the condition of inaccessible areas. Additionally, the Seabrook Station Structures Monitoring Program, B.2.1.31 will include examinations of concrete in soil below grade every 5 years or when excavated for any reason. To monitor the below grade environment, ground water chemistry will be sampled every 5 years for the above parameters as part of the Seabrook Station Structures Monitoring Program, B.2.1.31.

2. Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) Due to Aggressive Chemical Attack for Groups 1-5, 7, 9 Structures.

   Concrete in inaccessible areas is evaluated for increase in porosity and permeability, cracking, and loss of material due to aggressive chemical attack. The NUREG-1801 description of an aggressive environment is pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm. Recent analysis of groundwater samples has shown an increase in chloride levels to above the threshold. Seabrook Station groundwater is currently classified as aggressive.

   Therefore, increase in porosity and permeability, cracking, and loss of material due to aggressive chemical attack are aging effects requiring aging management for the period of extended operation.

   When applicable, the condition of accessible areas may be used to evaluate the condition of inaccessible areas. Additionally, the Seabrook Station Structures Monitoring Program, B.2.1.31 will include examinations of concrete in below grade soil every 5 years or when excavated for any reason. To monitor the below grade environment, ground water chemistry will be sampled every 5 years for the above parameters as part of the Seabrook Station Structures Monitoring Program, B.2.1.31.

3. Loss of Material Due to Corrosion for Groups 1-5, 7, 8 Structures

   Loss of material due to corrosion is an aging effect requiring management for the period of extended operation. The Seabrook Station Structures Monitoring Program, B.2.1.31 will be used to manage this aging effect for Groups 1-5, 7, 8 Structures.

4. Loss of Material (Spalling, Scaling) and Cracking Due to Freeze-Thaw for Groups1-3, 5, 7-9 Structures

Concrete in inaccessible areas is evaluated for loss of material and cracking due to freeze-thaw. Seabrook Station is located in a severe weathering region according to Figure 1 of ASTM C33-07.

Due to the aggregate size used in Seabrook Station, the air content of the concrete is higher then 6% recommended by NUREG-1801 for freeze thaw resistance, but within the acceptable guidelines of ACI 201 and 318. The concrete is a dense, durable mixture of sound, coarse aggregate, cement and water. Because of the slight variation in the concrete, Seabrook Station will manage the aging effect of loss of material and cracking of concrete due to freeze-thaw for the period of extended operation.

5. Cracking Due to Expansion and Reaction With Aggregates for Group 1-5, 7-9 Structures.

   Concrete in inaccessible areas is evaluated for expansion and cracking due to reaction with aggregate. Tests and petrographic examinations performed according to ASTM C227-50 or ASTM C295-54 verified that aggregates used are not reactive. However, Seabrook Station conservatively manages cracking due to expansion and reaction with aggregates through the Seabrook Station Structures Monitoring Program, B.2.1.31.

6. Cracks and Distortion Due to Increased Stress Levels from Settlement for Groups 1-3, 5-9 Structures.

   Seabrook Station structures are founded on sound bedrock, fill concrete, or engineered backfill and do not have any potential areas of settlement or displacement which need be monitored. Similarly, gradation requirements, compaction criteria and compaction test for engineered fill ensure a foundation material which will support the design loads with negligible settlement. A dewatering system is not used at Seabrook Station. Therefore, cracks and distortion of concrete due to increased stress levels from settlement, are not aging effects requiring management for the period of extended operation.

7. Reduction in Foundation Strength, Cracking, Differential Settlement Due to Erosion of Porous Concrete Subfoundation for Groups 1-3, 5-9 Structures.

   Differential settlement and erosion of porous concrete sub-foundations is not applicable to Seabrook Station. The concrete foundations at Seabrook Station are not constructed of porous concrete.

   Therefore, reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations, are not aging effects requiring management for the period of extended operation.

8. Lock Up Due to Wear for Lubrite Radial Beam Seats in Drywell and Other Sliding Support Bearings and Sliding Support Surfaces.

   NUREG-1801 requires aging management for fretting or lockup due to mechanical wear of Lubrite® or similar material. However, Electric Power Research Institute (EPRI) Aging Effects for Structures and Structural Components (Structural Tools), evaluates the aging effect (loss of material) and says that wear is not significant since there is insufficient relative motion and frequency due to thermal cycling during plant heat-up, cool-down, and normal operation.

   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, lock-up due to wear for Lubrite® plates is not an aging effect requiring management at Seabrook Station. Nonetheless, Lubrite® plate inspections are performed in accordance with the Seabrook Station Structures Monitoring Program, B.2.1.31 and Seabrook Station ASME Section XI, Subsection IWF Program, B.2.1.29 to confirm the absence of wear.

### 3.5.2.2.2.2 Aging Management of Inaccessible Areas

1. Loss of material due to freeze-thaw

   NUREG-1800 item 3.5.2.2.2.2 (1) relates to loss of material and cracking due to freeze-thaw in below-grade inaccessible concrete areas of Groups 1-3, 5, and 7-9 structures. Further evaluation of this aging effect is recommended for inaccessible areas of these Groups of structures for plants located in moderate to severe weathering conditions.

   Concrete in inaccessible areas is evaluated for loss of material and cracking due to freeze-thaw. Seabrook Station is located in a severe weathering region according to Figure 1 of ASTM C33-07.

   Due to the aggregate size used in Seabrook Station concrete, the air content of the concrete is higher then 6% as recommended by NUREG-1801 for freeze thaw resistance, but within the acceptable guidelines of ACI 201 and 318. The concrete is a dense, durable mixture of sound, coarse aggregate, cement and water. Because of the slight variation in the concrete, Seabrook Station will manage the aging effect of loss of material and cracking of concrete due to freeze-thaw for the period of extended operation.

   When applicable, the condition of accessible areas may be used to evaluate the condition of inaccessible areas. Additionally, the Seabrook Station Structures Monitoring Program, B.2.1.31 will include examinations of concrete below grade in soil every 5 years or when excavated for any reason. To monitor the below grade environment, ground water chemistry will be sampled every 5 years for the above parameters as part of the Seabrook Station Structures Monitoring Program, B.2.1.31.

2. Cracking due to expansion and reaction with aggregates

   NUREG-1800 item 3.5.2.2.2.2 (2) relates to cracking due to expansion and reaction with aggregates in below-grade inaccessible concrete areas of Groups 1-5, and 7-9 structures. Further evaluation is recommended if the concrete was not constructed in accordance with the recommendations in ACI 201.2R.

   Concrete was constructed equivalent to recommendations in ACI 201.2R.

   Concrete aggregates used in Seabrook Station concrete structures were selected per ASTM C33, which uses ASTM C295 "Petrographic Examination of Aggregates for Concrete". Aggregates identified as potentially reactive were not used at Seabrook Station. Nevertheless, Seabrook Station uses the Seabrook Station Structures Monitoring Program, B.2.1.31 to conservatively manage the aging effect of cracking of concrete due to expansion and reaction with aggregate.

3. Cracks and distortion due to increased stress levels from settlement and reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations.

   NUREG-1800 item 3.5.2.2.2.2 (3) relates to cracks and distortion due to increased stress levels from settlement and reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations in below-grade inaccessible concrete areas of Groups 1-3, 5 and 7-9 structures. If the plant's CLB credits a de-watering system, NUREG-1801 recommends verification of the continued functionality of the de-watering system during the period of extended operation. Otherwise, no further evaluation is required if this activity is included in the scope of the Structures Monitoring Program.

   Seabrook Station does not rely on a dewatering system for control of settlement.

   Differential settlement and erosion of porous concrete sub-foundations is not applicable to Seabrook Station. Seabrook Station structures are founded on sound bedrock, fill concrete, or engineered backfill that is not subject to significant settlement. The concrete foundations at Seabrook Station are not constructed of porous concrete and are not subject to flowing water.

   Therefore, cracks and distortion due to increased stress levels from settlement, and reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations, are not aging effects requiring management for the period of extended operation.

4. Aggressive chemical attack and corrosion of embedded steel

Further evaluation is recommended by NUREG-1801 for aging management of inaccessible concrete areas exposed to an aggressive environment. Possible aging effects are increases in porosity and permeability, cracking, and loss of material (spalling, scaling) due to aggressive chemical attack and cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel. Periodic monitoring of below-grade water chemistry is recommended as an acceptable approach to demonstrate that the below-grade environment is not aggressive.

Degradation due to aggressive chemical attack is applicable to Seabrook Station. Aggressive chemical attack only becomes significant when environmental conditions exceed threshold values (Chlorides > 500 ppm, Sulfates >1500 ppm, or pH < 5.5). Seabrook Station is not located in areas exposed to sulfate attack, nor is it located near industrial plants whose emissions could alter environmental parameters, but is exposed to chloride attack. Groundwater analyses confirm that the Seabrook Station site groundwater is aggressive. Testing performed from November 2008 to September 2009 found pH values between 5.8 and 7.5, chloride values between 19 ppm and 3900 ppm, and sulfate values between 10 ppm and 100 ppm. Resistance to mild acid attack is enhanced through the use of dense concrete that has low permeability and a low water to cement ratio.

Corrosion of embedded steel becomes significant if environmental conditions are found to be aggressive. As noted above, Seabrook Station groundwater analyses confirm that the Seabrook Station site groundwater is aggressive. Seabrook Station concrete is designed with low water to cement ratio, low permeability, and designed in accordance with ACI Standards (ACI 318 or ACI 349). Seabrook Station Structures Monitoring Program, B.2.1.31 will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

Conservatively, cracking, loss of bond, and loss of material due to corrosion of embedded steel are aging effects requiring aging management for the period of extended operation. Seabrook Station will use inspections conducted in accordance with the Seabrook Station Structures Monitoring Program, B.2.1.31 to meet this requirement.

5. Leaching of Calcium Hydroxide

NUREG-1800 indicates that further evaluation is recommended to address increases in porosity and permeability due to leaching of calcium hydroxide in below-grade inaccessible concrete areas in Groups 1-3, 5, and 7-9 structures. An aging management program is recommended only if the concrete was not constructed in accordance with the recommendations in ACI 201.2R. Otherwise, an aging management program is recommended.

Although concrete was constructed equivalent to recommendations in ACI 201.2R, loss of material due to leaching of calcium hydroxide is considered to be an aging effect requiring management for Seabrook Station. There have been indications of leaching in below grade concrete in Seabrook Station structures. 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, well-cured concrete with low permeability. These structures are designed in accordance with ACI 318 and constructed in accordance with ACI 301 and ASTM standards. Nevertheless, Seabrook Station manages loss of material due to leaching of calcium hydroxide with the Seabrook Station Structures Monitoring Program, B.2.1.31.

#### 3.5.2.2.2.3 Reduction of Strength and Modulus of Concrete Structures due to Elevated Temperature

NUREG-1800 item 3.5.2.2.2.3 relates to reduction of strength and modulus of concrete due to elevated temperatures in Group 1-5 concrete structures. For any concrete elements that exceed 150 °F for general areas and 200°F for local areas, further evaluation and implementation of a plant-specific program is recommended.

No in-scope Group 1-5 concrete structures at Seabrook Station exceed, or have areas that exceed, these thresholds.

#### 3.5.2.2.2.4 Aging Management of Inaccessible Areas for Group 6 Structures

1. Increase in Porosity and Permeability, and Loss of Material (Spalling, Scaling), Chemical Attack; Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling), Corrosion of Embedded Steel.

   Evaluation of concrete in inaccessible areas for Increase in Porosity and Permeability, and Loss of Material (Spalling, Scaling), Chemical Attack; Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling), Corrosion of Embedded Steel is applicable to Seabrook Station. Aggressive chemical attack only becomes significant when environmental conditions exceed threshold values (Chlorides > 500 ppm, Sulfates >1500 ppm, or pH < 5.5). Seabrook Station is not located in areas exposed to sulfate attack, nor is it located near industrial plants whose emissions could alter environmental parameters, but is exposed to chloride attack. Groundwater analyses confirm that the Seabrook Station site groundwater is aggressive. Testing performed from November 2008 to September 2009 found pH values between 5.8 and 7.5, chloride values between 19 ppm and 3900 ppm, and sulfate values between 10 ppm and 100 ppm. Seabrook Station Structures Monitoring Program, B.2.1.31 will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

Therefore Increase in Porosity and Permeability, and Loss of Material (Spalling, Scaling), Chemical Attack; Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling), Corrosion of Embedded Steel are aging effects requiring aging management for the period of extended operation.

When applicable, the condition of accessible areas may be used to evaluate the condition of inaccessible areas. Additionally, the Seabrook Station Structures Monitoring Program, B.2.1.31 will include examinations of below grade in soil concrete every 5 years or when excavated for any reason. To monitor the below grade environment, ground water chemistry will be sampled every 5 years for the above parameters as part of the Seabrook Station Structures Monitoring Program, B.2.1.31.

2. Loss of Material (Spalling, Scaling) and Cracking Due to Freeze-Thaw.

   Concrete in inaccessible areas is evaluated for loss of material and cracking due to freeze-thaw. Seabrook Station is located in a severe weathering region according to Figure 1 of ASTM C33-07 and due to the aggregate size the air content is higher then 6%, as recommended by the Generic Aging Lessons Learned (GALL) but, within the acceptable guidelines of ACI 201 and 318. The concrete is a dense, durable mixture of sound, coarse aggregate, cement and water. Therefore loss of material and cracking of concrete due to freeze-thaw is an aging effect requiring aging management for the period of extended operation.

3. Cracking Due to Expansion and Reaction With Aggregates and Increase in

   Porosity and Permeability, and Loss of Strength Due to Leaching of Calcium Hydroxide

   Concrete in inaccessible areas is evaluated for expansion and cracking due to reaction with aggregate. Tests and petrographic examinations performed according to ASTM C227-50 or ASTM C295-54 verified that aggregates used are not reactive. Nevertheless, Seabrook Station manages both cracking due to expansion and reaction with aggregates and Increase in porosity and permeability, and loss of material due to leaching of calcium hydroxide with the Seabrook Station Structures Monitoring Program, B.2.1.31.

#### 3.5.2.2.2.5 Cracking due to Stress Corrosion Cracking and Loss of Material due to Pitting and Crevice Corrosion

Based on the EPRI Aging Effects for Structures and Structural Components (Structural Tools), aging management is not required for crack initiation and growth (cracking) due to stress corrosion cracking of stainless steel in the air/gas environment. The Seabrook Station environment does not contain aggressive contaminates, and the material temperature is less than 140°F. Both temperature and aggressive contaminate levels must breach industry limits for stress corrosion cracking to occur. Therefore, cracking of stainless steel due to stress corrosion cracking is not an aging effect requiring management at Seabrook Station. Loss of material due to pitting and crevice corrosion, however, is an aging effect that is managed at Seabrook Station.

**3.5.2.2.2.6 Aging of Supports Not Covered by Structures Monitoring Program**

NUREG-1800 item 3.5.2.2.2.6 relates to further evaluation of certain component support/aging effect combinations if they are not covered by the Structures Monitoring Program. This includes (1) loss of material due to general and pitting corrosion associated with Groups B2-B5 supports; (2) reduction in concrete anchor capacity due to degradation of the surrounding concrete associated with Groups B1-B5 supports; and (3) reduction/loss of isolation function due to degradation of vibration isolation elements associated with Group B4 supports.

For items (1) through (3), the Seabrook Station responses are shown below:

(1) Loss of material due to general and pitting corrosion associated with Groups B2-B5 supports.

Consistent with NUREG-1800, Seabrook Station manages loss of material due to corrosion in Groups B2-B5 supports with the Seabrook Station Structures Monitoring Program, B.2.1.31.

(2) Reduction in concrete anchor capacity due to degradation of the surrounding concrete associated with Groups B1-B5 supports.

Consistent with NUREG-1800, Seabrook Station manages reduction in concrete anchor capacity due to degradation of the surrounding concrete with the Seabrook Station Structures Monitoring Program, B.2.1.31.

(3) Reduction/loss of isolation function due to degradation of vibration isolation elements associated with Group B4 supports

This item is not applicable to Seabrook Station. Seabrook Station does not have any supports with vibration isolation elements which require AMR. The vibration isolation elements identified by the Seabrook Station integrated plant assessment were determined to be integral parts of active equipment.

**3.5.2.2.2.7 Cumulative Fatigue Damage due to Cyclic Loading**

Due to cyclic loading, cumulative fatigue damage is possible for Groups B1.1, B1.2, and B1.3 component supports. If a TLAA, as defined in 10 CFR 54.3, exists, then the TLAA must be evaluated in accordance with 10 CFR 54.21(c).

The results of Seabrook Station reviews conducted to identify TLAAs in the current licensing basis did not identify any fatigue analyses for component support members, including anchor bolts or welds. Therefore, no evaluation in accordance with 10 CFR 54.21(c) is required.

#### 3.5.2.2.3 Quality Assurance for Aging Management of Nonsafety-Related Components

Quality Assurance Program and Administrative Controls are discussed in Section B.1.3.

### 3.5.2.3 Time-Limited Aging Analyses (TLAAs)

The TLAAs identified below are associated with the Containment systems components and referenced in LRA Section 4.7.

- Crane

## 3.5.3 CONCLUSION

The Structures and Component Supports subject to aging management review have been identified in accordance with the scoping criteria of 10 CFR 54.4. Aging effects have been identified based on plant and industry operating experience as well as industry literature. Programs to manage these aging effects have been identified in this section, and detailed program descriptions are provided in Appendix B. These activities demonstrate that the aging effects associated with the Structures and Component Supports will be adequately managed such that there is reasonable assurance that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| **PWR Concrete (Reinforced and Prestressed) and Steel Containments** | | | | | |
| 3.5.1-1 | Concrete elements: walls, dome, basemat, ring girder, buttresses, containment (as applicable | Aging of accessible and inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel | ISI (IWL) and for inaccessible concrete, an examination of representative samples of below-grade concrete and periodic monitoring of groundwater if environment is non-aggressive. A plant-specific program is to be evaluated if environment is aggressive. | Yes, plant-specific, if environment is aggressive | Seabrook manages accessible and inaccessible concrete components due to corrosion of embedded steel with the ASME Section XI, Subsection IWL Program, B.2.1.28.<br>Aggressive chemical attack is an applicable aging effect requiring management for Seabrook.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.1.<br>The Structures Monitoring Program, B.2.1.31, will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-2 | Concrete elements: all | Cracks and distortion due to increased stress levels from settlement | Structures Monitoring Program. If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation. | Yes, if not within the scope of the applicant's structures monitoring program or a de-watering system is relied upon | Seabrook does not rely on a dewatering system for control of settlement.<br>Cracking and distortion due to increased stress levels from settlement is not an aging effect requiring management.<br>However, Seabrook structures are monitored for settlement as a part of the Structures Monitoring Program, B.2.1.31.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.2. |
| 3.5.1-3 | Concrete elements: foundation, subfoundation | Reduction in foundation strength, cracking, differential settlement due to erosion of porous concrete subfoundation | Structures Monitoring Program. If a de-watering system is relied upon to control erosion of cement from porous concrete subfoundations, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation | Yes, if not within the scope of the applicant's structures monitoring program or a de-watering system is relied upon | Reduction in foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations is not an aging effect requiring management for the Seabrook Containment Structure.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.2. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-4 | Concrete elements: dome, wall, basemat, ring girder, buttresses, containment, concrete fill-in annulus (as applicable) | Reduction of strength and modulus due to elevated temperature | Plant-specific | Yes, plant-specific if temperature limits are exceeded | This item is not applicable to Seabrook.<br>No containment components exceed the specified temperature thresholds.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.3. |
| 3.5.1-5 | BWR only. | | | | |
| 3.5.1-6 | Steel elements: steel liner, liner anchors, integral attachments | Loss of material due to general, pitting, and crevice corrosion | ISI (IWE) and 10 CFR Part 50, Appendix J | Yes, if corrosion is significant for inaccessible areas | Consistent with NUREG-1801.<br>Seabrook manages loss of material with the ASME Section XI, Subsection IWE Program, B.2.1.27.<br>Loss of material due to corrosion is not expected to be significant for inaccessible areas.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.4. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-7 | Prestressed containment tendons | Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature | TLAA evaluated in accordance with 10 CFR 54.21(c) | Yes, TLAA | This item is not applicable to Seabrook.<br>The Seabrook Containment Building is not a pre-stressed concrete containment.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.5. |
| 3.5.1-8 | BWR only. | | | | |
| 3.5.1-9 | Steel, stainless steel elements, dissimilar metal welds: penetration sleeves, penetration bellows; suppression pool shell, unbraced downcomers | Cumulative fatigue damage (CLB fatigue analysis exists) | TLAA evaluated in accordance with 10 CFR 54.21(c) | Yes, TLAA | Consistent with NUREG-1801 for Seabrook containment penetrations that experience significant cyclic loading.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.6. |
| 3.5.1-10 | Stainless steel penetration sleeves, penetration bellows, dissimilar metal welds | Cracking due to stress corrosion cracking | ISI (IWE) and 10 CFR Part 50, Appendix J and additional appropriate examinations / evaluations for bellows assemblies and dissimilar metal welds | Yes, detection of aging is to be evaluated | Cracking due to stress corrosion cracking is not an aging effect requiring management for these stainless steel components.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.7. |
| 3.5.1-11 | BWR only. | | | | |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-12 | Steel, stainless steel elements, dissimilar metal welds: penetration sleeves, penetration bellows; suppression pool shell, unbraced downcomers | Cracking due to cyclic loading | ISI (IWE) and 10 CFR Part 50, Appendix J supplemented to detect fine cracks | Yes, detection of aging is to be evaluated | This item is not applicable to Seabrook.<br>Cracking due to cyclic loading is not an aging effect requiring management for the Seabrook penetration elements.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.8. |
| 3.5.1-13 | BWR only. | | | | |
| 3.5.1-14 | Concrete elements: dome, wall, basemat, ring girder, buttresses, containment (as applicable) | Loss of material (scaling, cracking, and spalling) due to freeze –thaw | ISI (IWL) Evaluation is needed for plants that are located in moderate to severe weathering conditions (weathering index >100 day – inch/yr) (NUREG-1557) | Yes, for inaccessible areas of plants located in moderate to severe weathering conditions | Loss of material due to freeze-thaw effects is not an aging effect requiring management for Seabrook. Seabrook concrete containment is enclosed by a containment enclosure building and therefore not exposed to severe weathering conditions.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.9. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-15 | Concrete elements: walls, dome, basemat, ring girder, buttresses, containment, concrete fill-in annulus (as applicable) | Increase in porosity permeability due to leaching of calcium hydroxide; cracking due to expansion and reaction with aggregate | ISI (IWL) for accessible areas. None for inaccessible areas if concrete was constructed in accordance with the recommendations of ACI 201.2R | Yes, if concrete was not constructed as stated for inaccessible areas | The Seabrook AMR results conclude that cracking due to expansion and reaction with aggregate is not an aging mechanism requiring management for the containment structure at Seabrook.<br>Concrete was constructed equivalent to recommendations in ACI 201.2R.<br>Seabrook manages loss of material due to leaching of calcium hydroxide with the ASME Section XI, Subsection IWL Program, B.2.1.28.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.1.10. |
| 3.5.1-16 | Seals, gaskets, and moisture barriers | Loss of sealing and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers (caulking, flashing, and other sealants) | ISI (IWE) and 10 CFR Part 50, Appendix J | No | Consistent with NUREG-1801.<br>Seabrook manages loss of sealing with the ASME Section XI, Subsection IWE Program, B.2.1.27, and the 10 CFR 50 Appendix J Program, B.2.1.30. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-17 | Personnel airlock, equipment hatch and CRD hatch locks, hinges, and closure mechanisms | Loss of leak tightness in closed position due to mechanical wear of locks, hinges, and closure mechanisms | 10 CFR Part 50, Appendix J and plant Technical Specifications | No | Consistent with NUREG-1801.<br>The 10 CFR 50 Appendix J Program, B.2.1.30, is used to manage loss of leak tightness. |
| 3.5.1-18 | Steel penetration sleeves and dissimilar metal welds; personnel airlock, equipment hatch, and CRD hatch | Loss of material due to general, pitting, and crevice corrosion | ISI (IWE) and 10 CFR Part 50 Appendix J | No | Consistent with NUREG-1801.<br>The ASME Section XI, Subsection IWE, B.2.1.27 manages loss of material due to corrosion, 10 CFR 50 Appendix J Program, B.2.1.30, manages loss of leak tightness. |
| 3.5.1-19 | BWR only. | | | | |
| 3.5.1-20 | BWR only. | | | | |
| 3.5.1-21 | BWR only. | | | | |
| 3.5.1-22 | Prestressed containment: tendons and anchorage components | Loss of material due to corrosion | ISI (IWL) | No | Not Applicable. Seabrook Station does not have prestressed tendons. |
| **Safety Related and Other Structures; and Component Supports** | | | | | |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-23 | All Groups except Group 6: interior and above grade exterior concrete | Cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel | Structures Monitoring Program | Yes, if not within the scope of the applicant's Structures Monitoring Program | Consistent with NUREG-1801.<br>Seabrook manages the aging effects with the Structures Monitoring Program, B.2.1.31.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 1. |
| 3.5.1-24 | All Groups except Group 6: interior and above grade exterior concrete | Increase in porosity and permeability, cracking, loss of material (spalling, scaling) due to aggressive chemical attack | Structures Monitoring Program | Yes, if not within the scope of the applicant's Structures Monitoring Program | The Seabrook AMR results conclude that the groundwater is aggressive and chemical attack is applicable to Seabrook.<br>Therefore, all Seabrook structural components will be monitored by the Structures Monitoring Program, B.2.1.31.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 2. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-25 | All Groups except Group 6: steel components: all structural steel | Loss of material due to corrosion | Structures Monitoring Program If protective coatings are relied upon to manage the effects of aging, the structures monitoring program is to include provisions to address protective coating monitoring and maintenance | Yes, if not within the scope of the applicant's Structures Monitoring Program | Consistent with NUREG-1801.<br>Seabrook manages corrosion of steel components with the Structures Monitoring Program, B.2.1.31.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 3. |
| 3.5.1-26 | All Groups except Group 6: accessible and inaccessible concrete: foundation | Loss of material (spalling, scaling) and cracking due to freeze-thaw | Structures Monitoring Program Evaluation is needed for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) | Yes, if not within the scope of the applicant's Structures Monitoring Program or for inaccessible areas of plants located in moderate to severe weathering conditions | Consistent with NUREG-1801.<br>Seabrook manages Loss of material (spalling, scaling) and cracking due to freeze-thaw with the Structures Monitoring Program, B.2.1.31.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 4. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-27 | All Groups except Group 6: accessible and inaccessible interior / exterior concrete | Cracking due to expansion due to reaction with aggregates | Structures Monitoring Program None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R-77 | Yes, if not within the scope of the applicant's Structures Monitoring Program or concrete was not constructed as stated for inaccessible areas | This item is not applicable to Seabrook.<br>The Seabrook AMR results conclude that reaction with aggregates is not significant and the concrete was constructed consistent with the recommendations of ACI 201.2R.<br>Nonetheless, all Seabrook structural components applicable to this item will be monitored by the Structures Monitoring Program, B.2.1.31.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 5. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-28 | Groups 1-3, 5-9: All | Cracks and distortion due to increased stress levels from settlement | Structures Monitoring Program If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation | Yes, if not within the scope of the applicant's Structures Monitoring Program or a de-watering system is relied upon | This item is not applicable to Seabrook.<br>The Seabrook AMR results conclude that settlement is not significant.<br>Further, a dewatering system is not relied upon for control of settlement at Seabrook.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 6. |
| 3.5.1-29 | Groups 1-3, 5-9: foundation | Reduction in foundation strength, cracking, differential settlement due to erosion of porous concrete subfoundation | Structures Monitoring Program If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation | Yes, if not within the scope of the applicant's Structures Monitoring Program or a de-watering system is relied upon | This item is not applicable to Seabrook.<br>The Seabrook AMR results conclude that settlement is not significant.<br>Further, a dewatering system is not relied upon for control of settlement at Seabrook.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 7. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-30 | Group 4: Radial beam seats in BWR drywell; RPV support shoes for PWR with nozzle support; Steam generator supports | Lock-up due to wear | ISI (IWF) or Structures Monitoring Program | Yes, if not within the scope of ISI of Structures Monitoring Program | 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.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.1, Item 8. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-31 | Groups 1-3, 5, 7-9: below grade concrete components, such as exterior walls below grade and foundation | Increase in porosity and permeability, cracking, loss of material (spalling, scaling) / aggressive chemical attack; cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel | Structures Monitoring Program Examination of representative samples of below grade concrete, and periodic monitoring of groundwater, if the environment is non-aggressive. A plant-specific program is to be evaluated if the environment is aggressive | Yes, plant-specific if environment is aggressive | The Seabrook AMR results conclude that the groundwater is aggressive.<br>The Structures Monitoring Program, B.2.1.31, will manage degradation of accessible and inaccessible concrete components due to corrosion of embedded steel.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.2.<br>The Structures Monitoring Program, B.2.1.31, will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-32 | Groups 1-3, 5, 7-9: exterior above and below grade reinforced concrete foundations | Increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide | Structures Monitoring Program for accessible areas. None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R-77 | Yes, if concrete was not constructed as stated for inaccessible areas | Loss of material due to leaching of calcium hydroxide is considered to be an aging effect requiring management for Seabrook. There have been indications of leaching in below grade concrete in Seabrook structures.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.2, Item 5. |
| 3.5.1-33 | Group 1-5: concrete | Reduction in strength and modulus due to elevated temperature | Plant-specific | Yes, plant specific if temperature limits are exceeded | This item is not applicable to Seabrook concrete components do not exceed the temperature limits specified in NUREG-1800.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.3. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-34 | Group 6: concrete; all | Cracking, loss of bond, loss of material due to corrosion of embedded steel, increase in porosity and permeability, cracking, loss of material due to aggressive chemical attack | Inspection of Water Control Structures Associated with Nuclear Power Plants and for inaccessible concrete, examination of representative samples of below grade concrete, and periodic monitoring of groundwater, if environment is non-aggressive. Plant-specific if environment is aggressive | Yes, plant-specific if environment is aggressive | Concrete in inaccessible areas is evaluated for increase in Porosity and Permeability, and Loss of Material (Spalling, Scaling), Chemical Attack; Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling), Corrosion of Embedded Steel is applicable to Seabrook.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.4, Item 1.<br>The Structures Monitoring Program, B.2.1.31, will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-35 | Group 6: exterior above and below grade concrete foundation | Loss of material (spalling, scaling) and cracking due to freeze-thaw | Inspection of Water Control Structures Associated with Nuclear Power Plants is needed for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUEG 1557) | Yes, for inaccessible areas of plants located in moderate to severe weathering conditions | Consistent with NUREG-1801.<br>Seabrook manages Loss of material (spalling, scaling) and cracking due to freeze-thaw with the Structures Monitoring Program, B.2.1.31.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.4, Item 2. |
| 3.5.1-36 | Group 6: all accessible / inaccessible reinforced concrete | Cracking due to expansion / reaction with aggregates | Accessible areas: Inspection of Water Control Structures Associated with Nuclear Power Plants. None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R77 | Yes, if concrete was not constructed as stated for inaccessible areas | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31, will manage degradation of accessible and inaccessible concrete components for cracking due to expansion / reaction with aggregates.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.4, Item 3. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-37 | Group 6: exterior above and below grade reinforced concrete foundation interior slab | Increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide | For accessible areas, Inspection of Water Control Structures Associated with Nuclear Power Plants. None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R77 | Yes, if concrete was not constructed as stated for inaccessible areas | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31, will manage degradation of accessible and inaccessible concrete components due to Increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.4, Item 3. |
| 3.5.1-38 | Group 7, 8: Tank liners | Cracking due to stress corrosion cracking; loss of material due to pitting and crevice corrosion | Plant-specific | Yes, plant-specific | There are no components at Seabrook that are subject to this aging effect.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.5. |

***Table 3.5.1***

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-39 | Support members; welds; bolted connections; support anchorage to building structure | Loss of material due to general and pitting corrosion | Structures Monitoring Program | Yes, if not within the scope of the applicant's *Structures Monitoring* Program | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31, will manage degradation for components for loss of material due to general and pitting corrosion<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.6, Item1. |
| 3.5.1-40 | Building concrete at locations of expansion and grouted anchors; grout pads for support base plates | Reduction in concrete anchor capacity due to local concrete degradation / service-induced cracking or other concrete aging mechanisms | Structures Monitoring Program | Yes, if not within the scope of the applicant's Structures Monitoring Program | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31, will manage degradation for the aging effects.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.6, Item 2. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-41 | Vibration isolation elements | Reduction or loss of isolation function / radiation hardening, temperature, humidity, sustained vibratory loading | Structures Monitoring Program | Yes, if not within the scope of the applicant's Structures Monitoring Program | This item is not applicable to Seabrook.<br>Seabrook does not have any supports with vibration isolation elements which require AMR.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.6, Item 3. |
| 3.5.1-42 | Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds | Cumulative fatigue damage (CLB fatigue analysis exists) | TLAA evaluated in accordance with 10 CFR 54.21(c) | Yes, TLAA | This item is not applicable to Seabrook.<br>Seabrook does not have any CLB fatigue analyses for support members, anchor bolts, or welds.<br>Further evaluation is provided in LRA Subsection 3.5.2.2.2.7. |
| 3.5.1-43 | Group 1-3, 5, 6: all masonry block walls | Cracking due to restraint shrinkage, creep, and aggressive environment | Masonry Wall Program | No | Seabrook manages cracking of masonry block walls and masonry units with the Structures Monitoring Program, B.2.1.31. In addition, masonry wall Fire Barriers, are managed with the Fire Protection Program, B.2.1.15. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-44 | Group 6 elastomer seals, gaskets, and moisture barriers | Loss of sealing due to deterioration of seals, gaskets, and moisture barriers (caulking, flashing, and other sealants) | Structures Monitoring Program | No | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31. |
| 3.5.1-45 | Group 6: exterior above and below ground concrete foundation; interior slab | Loss of material due to abrasion, cavitation | Inspection of Water Control Structures Associated with Nuclear Power Plants | No | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31, will confirm the absence of aging effects requiring management. |
| 3.5.1-46 | Group 5: fuel pool liners | Cracking due to stress corrosion cracking; loss of material due to pitting and crevice corrosion | Water Chemistry and Monitoring of spent fuel pool water level and level of fluid in the leak chase channel | No | The spent fuel pool is normally maintained less than 140°F, therefore Stress Corrosion Cracking is not an aging effect that requires management. Crevice and pitting corrosion are managed by the Water Chemistry Program, B.2.1.2. |
| 3.5.1-47 | Group 6: all metal structural members | Loss of material due to general (steel only), pitting, and crevice corrosion | Inspection of Water Control Structures Associated with Nuclear Power Plants. If protective coatings are relied upon to manage aging, protective coating monitoring and maintenance provisions should be included | No | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31, will confirm the absence of aging effects requiring management. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-48 | Group 6: earthen water control structures – dams, embankments, reservoirs, channels, canals, and ponds | Loss of material, loss of form due to erosion, settlement, sedimentation, frost action, waves, currents, surface runoff, seepage | Inspection of Water Control Structures Associated with Nuclear Power Plants | No | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31 will confirm the absence of aging effects requiring management. |
| 3.5.1-49 | BWR only. | | | | |
| 3.5.1-50 | Groups B2 and B4: galvanized steel, aluminum, stainless steel support members; welds; bolted connections; support anchorage to building structure | Loss of material due to pitting and crevice corrosion | Structures Monitoring Program | No | Consistent with NUREG-1801.<br>The Structures Monitoring Program, B.2.1.31, will confirm the absence of aging effects requiring management. |
| 3.5.1-51 | Group B1.1: high strength low-alloy bolts | Cracking due to stress corrosion cracking; loss of material due to general corrosion | Bolting Integrity | No | There are no high strength bolts at Seabrook that are subject to this aging effect. |
| 3.5.1-52 | Groups B2 and B4: sliding support bearings and sliding support surfaces | Loss of mechanical function due to corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads | Structures Monitoring Program | No | There are no sliding support bearings of surfaces at Seabrook that are subject to this aging effect. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-53 | Groups B1.1, B1.2, and B1.3: support members: welds; bolted connections; support anchorage to building structure. | Loss of material due to general and pitting corrosion | ISI (IWF) | No | Consistent with NUREG-1801.<br>Seabrook manages the aging effect with the ASME Section XI, Subsection IWF Program, B.2.1.29. |
| 3.5.1-54 | Group B1.1, B1.2, and B1.3: Constant and variable load spring hangers; guides; stops | Loss of mechanical function due to corrosion, distortion, dirt, overload fatigue due to vibratory and cyclic thermal loads | ISI (IWF) | No | Consistent with NUREG-1801.<br>Seabrook manages the aging effect with the ASME Section XI, Subsection IWF Program, B.2.1.29. |
| 3.5.1-55 | Steel, galvanized steel, and aluminum support members; welds; bolted connections; support anchorage to building structure | Loss of material due to boric acid corrosion | Boric Acid Corrosion | No | Consistent with NUREG-1801.<br>Seabrook manages the aging effect of loss of material due to boric acid corrosion in steel, galvanized steel, and aluminum for all types of support members (including safety and non-safety), welds, bolted connections and support anchorage to building structure with the Boric Acid Corrosion Program, B.2.1.4. |

**Table 3.5.1**

**Summary Of Aging Management Evaluations for Structures and Structural Components**

| Item Number | Component | Aging Effect / Mechanism | Aging Management Program | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.5.1-56 | Groups B1.1, B1.2, and B1.3: Sliding surfaces | Loss of material function due to corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads | ISI (IWF) | No | There are no sliding support bearings of surfaces at Seabrook that are subject to this aging effect. |
| 3.5.1-57 | Groups B1.1, B1.2, and B1.3: Vibration isolation elements | Reduction or loss of isolation function/radiation hardening, temperature, humidity, sustained vibratory loading | ISI (IWF) | No | This item is not applicable to Seabrook.<br>The Seabrook AMR results do not include any supports with vibration isolation elements. |
| 3.5.1-58 | Galvanized steel and aluminum support members; welds; bolted connections; support anchorage to building structure exposed to air – indoor uncontrolled | None | None | NA – no aging effect management or aging management program | Consistent with NUREG-1801. |
| 3.5.1-59 | Stainless steel support members; welds; bolted connections; support anchorage to building structure | None | None | NA – no aging effect management or aging management program | Consistent with NUREG-1801. |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Carbon Steel FIRE PUMPHOUSE Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| BSAS Carbon Steel FIRE PUMPHOUSE Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| BSAS Carbon Steel NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| BSAS Carbon Steel NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| BSAS Carbon Steel REVETMENT Exposed to Air Outdoor | Flood Barrier | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 503, 511 |
| BSAS Carbon Steel REVETMENT Exposed to Air Outdoor | Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 503, 511 |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Flood Barrier | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 509, 511 |
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 509, 511 |
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Flood Barrier | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 509, 511 |
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 509, 511 |
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Flood Barrier | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | H, 514 |
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | H, 514 |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Flood Barrier | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 509, 511 |
| BSAS Carbon Steel REVETMENT Exposed to Raw Water | Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 509, 511 |
| BSAS Carbon Steel STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| BSAS Carbon Steel STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| BSAS Concrete NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| BSAS Concrete NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| BSAS Concrete (Sump) FIRE PUMPHOUSE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| BSAS Concrete (Sump) FIRE PUMPHOUSE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| BSAS Concrete (Sump) STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| BSAS Concrete (Sump) STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A6-5 (T-15) | 3.5.1-35 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A6-5 (T-15) | 3.5.1-35 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete DISCHARGE TRANSITION STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Increase In Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A6-6 (T-16) | 3.5.1-37 | E, 511 |
| BSAS Concrete FIRE PUMPHOUSE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| BSAS Concrete FIRE PUMPHOUSE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| BSAS Concrete FIRE PUMPHOUSE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete FIRE PUMPHOUSE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete FIRE PUMPHOUSE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| BSAS Concrete FIRE PUMPHOUSE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete FIRE PUMPHOUSE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A6-5 (T-15) | 3.5.1-35 | A |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A6-5 (T-15) | 3.5.1-35 | A |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete INTAKE TRANSITION STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Increase In Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A6-6 (T-16) | 3.5.1-37 | E, 511 |
| BSAS Concrete Masonry Units FIRE PUMPHOUSE Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete Block | Air Indoor Uncontrolled (External) | Cracking | Fire Protection Program | III.A3-11 (T-12) | 3.5.1-43 | E, 513 |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete Masonry Units FIRE PUMPHOUSE Exposed to Air Indoor Uncontrolled | Structural Support | Concrete Block | Air Indoor Uncontrolled (External) | Cracking | Structures Monitoring Program | III.A3-11 (T-12) | 3.5.1-43 | E, 511 |
| BSAS Concrete Masonry Units NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Concrete Block | Air Indoor Uncontrolled (External) | Cracking | Structures Monitoring Program | III.A3-11 (T-12) | 3.5.1-43 | E, 511 |
| BSAS Concrete Masonry Units NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Outdoor | Structural Support | Concrete Block | Air Outdoor (External) | Cracking | Structures Monitoring Program | III.A3-11 (T-12) | 3.5.1-43 | E, 511 |
| BSAS Concrete NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| BSAS Concrete NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| BSAS Concrete NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete REVETMENT Below Grade | Flood Barrier | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| BSAS Concrete REVETMENT Below Grade | Support | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| BSAS Concrete REVETMENT Below Grade | Flood Barrier | Concrete | Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete REVETMENT Below Grade | Support | Concrete | Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete REVETMENT Below Grade | Flood Barrier | Concrete | Soil (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-3 (T-19) | 3.5.1-34 | E, 511 |
| BSAS Concrete REVETMENT Below Grade | Support | Concrete | Soil (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-3 (T-19) | 3.5.1-34 | E, 511 |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete REVETMENT Below Grade | Flood Barrier | Concrete | Soil (External) | Increase In Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A6-6 (T-16) | 3.5.1-37 | E, 509, 511 |
| BSAS Concrete REVETMENT Below Grade | Support | Concrete | Soil (External) | Increase In Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A6-6 (T-16) | 3.5.1-37 | E, 509, 511 |
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A6-1 (T-18) | 3.5.1-34 | E, 511 |
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A6-2 (T-17) | 3.5.1-36 | E, 511 |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Support | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A6-5 (T-15) | 3.5.1-35 | E, 511 |
| BSAS Concrete REVETMENT Exposed to Air Outdoor | Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A6-5 (T-15) | 3.5.1-35 | E, 511 |
| BSAS Concrete STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| BSAS Concrete STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Concrete STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| BSAS Concrete STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| BSAS Concrete STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Increase In Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| BSAS Concrete STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| BSAS Rock (Riprap) REVETMENT Exposed to Air Outdoor | Flood Barrier | Rock | Air Outdoor (External) | Loss of Material, Loss of Form | Structures Monitoring Program | III.A6-9 (T-22) | 3.5.1-48 | E, 511 |

**Table 3.5.2-1**

**Buildings, Structures Within License Renewal**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| BSAS Rock (Riprap) REVETMENT Exposed to Air Outdoor | Support | Rock | Air Outdoor (External) | Loss of Material, Loss of Form | Structures Monitoring Program | III.A6-9 (T-22) | 3.5.1-48 | E, 511 |
| BSAS Roofing For FIRE PUMPHOUSE Exposed to Air Outdoor | Structural Support | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| BSAS Roofing For NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Outdoor | Structural Support | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| BSAS Roofing For STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Outdoor | Structural Support | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| BSAS Slide Bearing (Fluorogold®) NONESSENTIAL SWITCHGEAR BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Fluorogold® | Air Indoor Uncontrolled (External) | Fretting Or Lockup | Structures Monitoring Program | III.B4-2 (TP-1) | 3.5.1-52 | C |
| BSAS Stainless Steel STEAM GENERATOR BLOWDOWN RECOVERY BUILDING Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | No Aging Management Program Required | III.B5-5 (TP-5) | 3.5.1-59 | A |

**Table 3.5.2-1**
**Buildings, Structures Within License Renewal**
**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GALL; III.A6-12 for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F); water chemistry and temperature will be maintained by the Water Chemistry Program. |
| 508 | Cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program. |
| 509 | For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the same. |
| 510 | Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT. |
| 511 | At Seabrook Station, XI.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI.S5 "Masonry Wall Program" are combined under XI.S6 "Structures Monitoring Program". |

512 Raw water in lined & unlined concrete sumps.

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Carbon Steel Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CE-Carbon Steel Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| CNT-CE-Carbon Steel Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A, 503 |
| CNT-CE-Fire Penetration Seal Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness and Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| CNT-CE-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-4 (T-05) | 3.5.1-31 | A |
| CNT-CE-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-4 (T-05) | 3.5.1-31 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-5 (T-07) | 3.5.1-31 | A |
| CNT-CE-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-5 (T-07) | 3.5.1-31 | A |
| CNT-CE-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A1-7 (T-02) | 3.5.1-32 | A, 509 |
| CNT-CE-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A1-7 (T-02) | 3.5.1-32 | A, 509 |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CE-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CE-Reinforced Concrete Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CE-Stainless Steel Exposed to Air Outdoor | Structural Support | Stainless Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.B2-7 (TP-6) | 3.5.1-50 | A |
| CNT-CEVA-Built-Up Roofing Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | 505, H |
| CNT-CEVA-Carbon Steel Door Exposed to Air with Borated Water Leakage | HELB Shielding | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Door Exposed to Air with Borated Water Leakage | Shelter, Protection | Steel | Air w/Borated Water Leakage (External | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Door Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Carbon Steel Door Exposed to Air Indoor Uncontrolled | HELB Shielding | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| CNT-CEVA-Carbon Steel Door Exposed to Air Indoor Uncontrolled | Shelter, Protection | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| CNT-CEVA-Carbon Steel Door Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| CNT-CEVA-Carbon Steel Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Carbon Steel Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| CNT-CEVA-Carbon Steel Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A, 503 |
| CNT-CEVA-Carbon Steel Fire Door Exposed to Air with Borated Water Leakage | Fire Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Fire Door Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Fire Door Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| CNT-CEVA-Carbon Steel Fire Door Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Carbon Steel Fire Door Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| CNT-CEVA-Carbon Steel Fire Door Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| CNT-CEVA-Carbon Steel Tech Spec Door Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Tech Spec Door Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CEVA-Carbon Steel Tech Spec Door Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| CNT-CEVA-Carbon Steel Tech Spec Door Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Elastomeric Pressure Seal and Caulk Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| CNT-CEVA-Elastomers Exposed to Air Outdoor | Expansion / Separation | Elastomer | Air Outdoor (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| CNT-CEVA-Fire Penetration Seal Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor, Uncontrolled (External) | Increased Hardness and Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| CNT-CEVA-Fire Penetration Seal Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| CNT-CEVA-Penetration Seal Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| CNT-CEVA-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-4 (T-05) | 3.5.1-31 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-4 (T-05) | 3.5.1-31 | A |
| CNT-CEVA-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-5 (T-07) | 3.5.1-31 | A |
| CNT-CEVA-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-5 (T-07) | 3.5.1-31 | A |
| CNT-CEVA-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A1-7 (T-02) | 3.5.1-32 | A, 509 |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A1-7 (T-02) | 3.5.1-32 | A, 509 |
| CNT-CEVA-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CEVA-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Flood Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CEVA-Reinforced Concrete Exposed to Air Outdoor | Structural Pressure Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| CNT-CEVA-Tech Spec Seal Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| CNT-CEVA-Tech Spec Seal Exposed to Air Outdoor | Structural Pressure Barrier | Elastomer | Air Outdoor (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| CNT-CEVA-Thermal Insulation Aluminum Jacketing in Air with Borated Water Leakage | Structural Support | Aluminum | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-6 (TP-3) | 3.5.1-55 | A |
| CNT-CI-Carbon Steel Door Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (TP-25) | 3.5.1-55 | A |
| CNT-CI-Carbon Steel Door Exposed to Air Indoor Uncontrolled | Shelter, Protection | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A4-5 (T-11) | 3.5.1-25 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CI-Carbon Steel Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (TP-25) | 3.5.1-55 | A |
| CNT-CI-Carbon Steel Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A4-5 (T-11) | 3.5.1-25 | A |
| CNT-CI-Conduit Fire Wrap Exposed to Air Indoor Uncontrolled | Fire Barrier | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B1.2-7 (TP-5) | 3.5.1-59 | A |
| CNT-CI-Conduit Fire Wrap Exposed to Air with Borated Water Leakage | Fire Barrier | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CI-Heat Shield Exposed to Air with Borated Water Leakage | Fire Barrier | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CI-Radiant Heat Shield Exposed to Air Indoor Uncontrolled | Fire Barrier | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B1.2-7 (TP-5) | 3.5.1-59 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-3 (T-04) | 3.5.1-23 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-3 (T-04) | 3.5.1-23 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-3 (T-04) | 3.5.1-23 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-3 (T-04) | 3.5.1-23 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A4-2 (T-03) | 3.5.1-27 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A4-2 (T-03) | 3.5.1-27 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A4-2 (T-03) | 3.5.1-27 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A4-2 (T-03) | 3.5.1-27 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-4 (T-06) | 3.5.1-24 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-4 (T-06) | 3.5.1-24 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-4 (T-06) | 3.5.1-24 | A |
| CNT-CI-Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A4-4 (T-06) | 3.5.1-24 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CI-Stainless Steel Exposed to Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B1.29 (TP-4) | 3.5.1-59 | A |
| CNT-CI-Stainless Steel Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B1.2-7 (TP-5) | 3.5.1-59 | A |
| CNT-CI-Stainless Steel Raw Water | Structural Support | Stainless Steel | Raw Water (External) | Loss of Materia | Structures Monitoring Program | V.D1-15 (E-01) | 3.2.1-7 | E, 514 |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Flood Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Missile Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shelter, Protection | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS- Reinforced Concrete Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |
| CNT-CS-Airlock Hatch Sight Glass Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Glass | Air Indoor Uncontrolled (External) | None | None | V.F-6 (EP-15) | 3.2.1-52 | A |
| CNT-CS-Carbon Steel Electrical Penetration Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Electrical Penetration Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Electrical Penetration Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-1 (C-12) | 3.5.1-18 | A |
| CNT-CS-Carbon Steel Electrical Penetration Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-1 (C-12) | 3.5.1-18 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Carbon Steel Equipment Hatch Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Equipment Hatch Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Equipment Hatch Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-6 (C-16) | 3.5.1-18 | A |
| CNT-CS-Carbon Steel Equipment Hatch Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-6 (C-16) | 3.5.1-18 | A |
| CNT-CS-Carbon Steel Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Carbon Steel Exposed to Air with Borated Water Leakage for HVAC Penetrations | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Exposed to Air with Borated Water Leakage for HVAC Penetrations | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A1-11 (C-09) | 3.5.1-6 | A |
| CNT-CS-Carbon Steel Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A1-11 (C-09) | 3.5.1-6 | A |
| CNT-CS-Carbon Steel Exposed to Air Indoor Uncontrolled For HVAC Penetrations | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-1 (C-12) | 3.5.1-18 | A |
| CNT-CS-Carbon Steel Exposed to Air Indoor Uncontrolled For HVAC Penetrations | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-1 (C-12) | 3.5.1-18 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Carbon Steel Mechanical (Piping) Penetration Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Mechanical (Piping) Penetration Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Mechanical (Piping) Penetration Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-1 (C-12) | 3.5.1-18 | A |
| CNT-CS-Carbon Steel Mechanical (Piping) Penetration Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-1 (C-12) | 3.5.1-18 | A |
| CNT-CS-Carbon Steel Personnel Hatch Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (CT-25) | 3.5.1-55 | A |
| CNT-CS-Carbon Steel Personnel Hatch Exposed to Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (CT-25) | 3.5.1-55 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Carbon Steel Personnel Hatch Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-6 (C-16) | 3.5.1-18 | A |
| CNT-CS-Carbon Steel Personnel Hatch Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWE Program | II.A3-6 (C-16) | 3.5.1-18 | A |
| CNT-CS-Elastomers Electrical Penetration Assembly Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Loss of Sealing, Leakage Through Containment | ASME Section XI, Subsection IWE Program | II.A3-7 (C-18) | 3.5.1-16 | A |
| CNT-CS-Mechanical (Piping) Penetration Stainless Steel Flued Head Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Mechanical (Piping) Penetration Stainless Steel Flued Head Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Mechanical (Piping) Penetration Stainless Steel Flued Head Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Mechanical (Piping) Penetration Stainless Steel Flued Head Exposed to Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-7 (C-05) | 3.5.1-1 | A |
| CNT-CS-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Expansion and Cracking | ASME Section XI, Subsection IWL Program | II.A1-3 (C-04) | 3.5.1-15 | A |
| CNT-CS-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | ASME Section XI, Subsection IWL Program | II.A1-4 (C-03) | 3.5.1-1 | A, 516 |
| CNT-CS-Reinforced Concrete Below Grade | Shelter, Protection | Concrete | Soil (External) | Increase in Porosity and Permeability, Loss of Strength | ASME Section XI, Subsection IWL Program | II.A1-6 (C-02) | 3.5.1-15 | A, 509, 516 |
| CNT-CS-Reinforced Concrete Below Grade | Structural Support | Concrete | Soil (External) | Increase in Porosity and Permeability, Loss of Strength | ASME Section XI, Subsection IWL Program | II.A1-6 (C-02) | 3.5.1-15 | A, 509, 516 |
| CNT-CS-Stainless Steel Electrical Penetration Assembly Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Stainless Steel Electrical Penetration Assembly Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Stainless Steel Electrical Penetration Assembly Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Stainless Steel Electrical Penetration Assembly Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Stainless Steel Exposed to Air with Borated Water Leakage | Shielding | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Stainless Steel Exposed to Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Stainless Steel Exposed to Air Indoor Uncontrolled | Shielding | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Stainless Steel Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Stainless Steel Fuel Transfer Tube Bellows Exposed to Air with Borated Water Leakage | Expansion / Separation | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Stainless Steel Fuel Transfer Tube Bellows Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Stainless Steel Fuel Transfer Tube Bellows Exposed to Air Indoor Uncontrolled | Expansion / Separation | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Stainless Steel Fuel Transfer Tube Bellows Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Stainless Steel Fuel Transfer Tube Exposed to Air with Borated Water Leakage | Structural Pressure Barrier | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Stainless Steel Fuel Transfer Tube Exposed to Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |
| CNT-CS-Stainless Steel Fuel Transfer Tube Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**

**Containment Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| CNT-CS-Stainless Steel Fuel Transfer Tube Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | Cracking | ASME Section XI, Subsection IWE Program | II.A3-2 (C-15) | 3.5.1-10 | A |
| CNT-CS-Thermal Insulation Stainless Steel Jacketing in Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage | None | None | III.B2-9 (TP-4) | 3.5.1-59 | A |

Containment Enclosure (CE), Containment Enclosure Ventilation Area (CEVA), Containment Internals (CI), Containment Structure (CS)

**Table 3.5.2-2**
**Containment Structures**
**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GAL; III.A6-12 is for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F), water chemistry and temperature will be maintained by the Water Chemistry Program. |
| 508 | Cracking, loss of bond, and loss of material (Spalling, Scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program. |
| 509 | For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the same. |
| 510 | Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT. |

511 At Seabrook Station, XI,.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI,.S5 "Masonry Wall Program" are combined under XI,.S6 "Structures Monitoring Program".

512 Raw water in lined & unlined concrete sumps.

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CBS-CR-18-A Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CBS-CR-18-A Radioactive Pipe Tunnel Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load "Related to Refueling" Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CBS-CR-18-A Radioactive Pipe Tunnel Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CBS-CR-18-A Radioactive Pipe Tunnel Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CBS-CR-18-B Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CBS-CR-18-B Radioactive Pipe Tunnel Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CBS-CR-18-B Radioactive Pipe Tunnel Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CBS-CR-18-B Radioactive Pipe Tunnel Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CC-CR-15-A Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CC-CR-15-A CC Water Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CC-CR-15-A CC Water Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CC-CR-15-A CC Water Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-07) | 3.3.1-74 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CC-CR-15-B CC Water Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CC-CR-15-B CC Water Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CC-CR-15-B CC Water Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-07) | 3.3.1-74 | A |
| 1-CC-CR-15-B Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CC-CR-41 CC Heat Exchanger Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CC-CR-41 CC Heat Exchanger Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CC-CR-41 CC Heat Exchanger Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CC-CR-41 Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CS-CR-13 CVCS Heat Exchanger Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CS-CR-13 CVCS Heat Exchanger Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CS-CR-13 CVCS Heat Exchanger Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CS-CR-13 Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CS-CR-14-A Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CS-CR-14-A Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CS-CR-14-A Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CS-CR-14-A Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CS-CR-14-B Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CS-CR-14-B Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CS-CR-14-B Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CS-CR-14-B Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CS-CR-14-C Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CS-CR-14-C Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CS-CR-14-C Charging Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CS-CR-14-C Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CS-CR-5 Filter Cask Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CS-CR-5 Filter Cask Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-CS-CR-5 Filter Cask Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CS-CR-5 Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-CS-CR-6 Boric Acid Batching Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-CS-CR-6 Boric Acid Batching Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-CS-CR-6 Boric Acid Batching Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-CS-CR-6 Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-DG-CR-28-A Diesel Generator Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-DG-CR-28-A Diesel Generator Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-DG-CR-28-A Diesel Generator Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-DG-CR-28-B Diesel Generator Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-DG-CR-28-B Diesel Generator Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-DG-CR-28-B Diesel Generator Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-FH-RE-1 Crane Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.3.1-55 | A |
| 1-FH-RE-1 Spent Fuel Cask Handling Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-FH-RE-1 Spent Fuel Cask Handling Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-FH-RE-1 Spent Fuel Cask Handling Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-FH-RE-2 Bridge & Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-FH-RE-2 Spent Fuel Bridge & Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-FH-RE-2 Spent Fuel Bridge & Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-FH-RE-2 Spent Fuel Bridge & Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-FH-RE-24-E Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-FH-RE-24-E Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-FH-RE-24-E Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-FH-RE-24-E Tensioner Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-FH-RE-24-F Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-FH-RE-24-F Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-FH-RE-24-F Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-FH-RE-24-F Tensioner Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-FH-RE-24-G Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-FH-RE-24-G Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-FH-RE-24-G Radial Arm Stud Tensioner Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-FH-RE-24-G Tensioner Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-FH-RE-5 Manipulator Crane Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-FH-RE-5 Refueling Machine or Manipulator Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-FH-RE-5 Refueling Machine or Manipulator Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-FH-RE-5 Refueling Machine or Manipulator Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-FW-CR-27 Emergency Feed Pump Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-FW-CR-27 Emergency Feed Pump Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-FW-CR-27 Emergency Feed Pump Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-MM-CR-3 Polar Gantry Crane Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-MM-CR-3 Polar Gantry Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-MM-CR-3 Polar Gantry Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-MM-CR-3 Polar Gantry Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-MM-CR-49 Jib Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-MM-CR-49 Personnel Hatch Jib Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-MM-CR-49 Personnel Hatch Jib Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-MM-CR-49 Personnel Hatch Jib Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-MS-CR-25-A MS/FW West Pipechase Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-MS-CR-25-A MS/FW West Pipechase Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-MS-CR-25-A MS/FW West Pipechase Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-MS-CR-25-B MS/FW East Pipechase Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-MS-CR-25-B MS/FW East Pipechase Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-MS-CR-25-B MS/FW East Pipechase Service Crane Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-SI-CR-40-A Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| 1-SI-CR-40-A Safety Injection Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-SI-CR-40-A Safety Injection Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-SI-CR-40-A Safety Injection Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |
| 1-SI-CR-40-B Monorail Hoist Carbon Steel In Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.3.1-55 | A |

**Table 3.5.2-3**

**Fuel Handling and Overhead Cranes**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| 1-SI-CR-40-B Safety Injection Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-3 (A-07) | 3.3.1-73 | A |
| 1-SI-CR-40-B Safety Injection Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.B-3 (A-07) | 3.3.1-73 | E, 514 |
| 1-SI-CR-40-B Safety Injection Pump Service Monorail Hoist Carbon Steel In Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Inspection of Overhead Heavy Load and Light Load Handling Systems | VII.B-1 (A-05) | 3.3.1-74 | A |

**Table 3.5.2-3**
**Fuel Handling and Overhead Cranes**
**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GALL; III.A6-12 is for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F), water chemistry and temperature will be maintained by the Water Chemistry Program. |
| 508 | Cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program. |
| 509 | For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the same. |
| 510 | Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT. |
| 511 | At Seabrook Station, XI.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI.S5 "Masonry Wall Program" are combined under XI.S6 "Structures Monitoring Program". |
| 512 | Raw water in lined & unlined concrete sumps. |

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Aluminum STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Aluminum | Air Outdoor (External) | Crack Initiation and Growth | Structures Monitoring Program | III.B2-7 (Tp-6) | 3.5.1-50 | H, 514 |
| MYS - Aluminum STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Aluminum | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.B2-7 (TP-6) | 3.5.1-50 | A |
| MYS - Carbon Steel CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| MYS - Carbon Steel Door ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel Door ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Carbon Steel Door STATION BLACKOUT STRUCTURES Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel Door STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| MYS - Carbon Steel ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| MYS - Carbon Steel NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel NON SAFETY RELATED ELECTRICAL MANHOLES Exposed to Weather | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Carbon Steel SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Exposed to Weather | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| MYS - Carbon Steel SERVICE WATER ACCESS VAULT Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel STATION BLACKOUT STRUCTURES Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| MYS - Carbon Steel STATION BLACKOUT STRUCTURES Exposed to Weather | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| MYS - Carbon Steel STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Fire Protection Program | III.A3-10 (T-06) | 3.5.1-24 | E, 513 |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Fire Protection Program | III.A3-10 (T-06) | 3.5.1-24 | E, 513 |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete Masonry Unit (CMU) STATION BLACKOUT STRUCTURES Exposed to Weather | Fire Barrier | Concrete Block | Air Outdoor (External) | Cracking | Fire Protection Program | III.A3-11 (T-12) | 3.5.1-43 | E, 513 |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete Masonry Unit (CMU) STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Concrete BLOCK | Air Outdoor (External) | Cracking | Fire Protection Program | III.A3-11 (T-12) | 3.5.1-43 | E, 513 |
| MYS - Concrete Masonry Unit (CMU) STATION BLACKOUT STRUCTURES Exposed to Weather | Fire Barrier | Concrete Block | Air Outdoor (External) | Cracking | Structures Monitoring Program | III.A3-11 (T-12) | 3.5.1-43 | A, 511 |
| MYS - Concrete Masonry Unit (CMU) STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Concrete BLOCK | Air Outdoor (External) | Cracking | Structures Monitoring Program | III.A3-11 (T-12) | 3.5.1-43 | A, 511 |
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| MYS - Concrete NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete NON SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete NON SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete NON SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete NON SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| MYS - Concrete SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete SAFETY RELATED MANHOLES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete SERVICE WATER ACCESS VAULT Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete SERVICE WATER ACCESS VAULT Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete SERVICE WATER ACCESS VAULT Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete SERVICE WATER ACCESS VAULT Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| MYS - Concrete SERVICE WATER ACCESS VAULT Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete SERVICE WATER ACCESS VAULT Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| MYS - Concrete SERVICE WATER ACCESS VAULT Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| MYS - Concrete SERVICE WATER ACCESS VAULT Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete SERVICE WATER ACCESS VAULT Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| MYS - Concrete SERVICE WATER ACCESS VAULT Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete STATION BLACKOUT STRUCTURES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete STATION BLACKOUT STRUCTURES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete STATION BLACKOUT STRUCTURES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| MYS - Concrete STATION BLACKOUT STRUCTURES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| MYS - Concrete STATION BLACKOUT STRUCTURES Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| MYS - Concrete STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| MYS - Concrete STATION BLACKOUT STRUCTURES Exposed to Weather | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| MYS - Concrete Sump CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| MYS - Concrete Sump CONTROL ROOM MAKEUP AIR INTAKE STRUCTURE Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - Concrete Sump NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| MYS - Concrete Sump NON SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-4**
**Miscellaneous Yard Structures**
**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Concrete Sump SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| MYS - Concrete Sump SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| MYS - EPDM Roof ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | A, 505 |
| MYS - EPDM Roof STATION BLACKOUT STRUCTURES Exposed to Weather | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | A, 505 |
| MYS - Penetration Seal ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| MYS - Seismic Isolation Joint SAFETY RELATED ELECTRICAL DUCT BANKS/MANHOLES Air Indoor Uncontrolled | Expansion/ Separation | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness and Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| MYS - Stainless Steel ENCLOSURE FOR CONDENSATE STORAGE TANK Exposed to Weather | Structural Support | Stainless Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.B2-7 (TP-6) | 3.5.1-50 | A |

**Table 3.5.2-4**

**Miscellaneous Yard Structures**

**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GALL; III.A6-12 is for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F); water chemistry and temperature will be maintained by the Water Chemistry Program. |
| 508 | Cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program. |
| 509 | For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the same. |
| 510 | Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT. |

511 At Seabrook Station, XI.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI.S5 "Masonry Wall Program" are combined under XI.S6 "Structures Monitoring Program".

512 Raw water in lined & unlined concrete sumps.

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - (Structural) Fire Proofing - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Non-Metallic Fire-Proofing | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | C, 506 |
| PST - (Structural) Fire Proofing - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Non-Metallic Fire-Proofing | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.G-29 (A-91) | 3.3.1-67 | C, 506 |
| PST - (Structural) Fire Proofing - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Non-Metallic Fire-Proofing | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | C, 506 |
| PST - (Structural) Fire Proofing - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Non-Metallic Fire-Proofing | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.G-29 (A-91) | 3.3.1-67 | C, 506 |
| PST - Built-Up Roofing - CDG- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Built-Up Roofing - EFP- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| PST - Built-Up Roofing - FSB- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| PST - Built-Up Roofing - PAB- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| PST - Built-Up Roofing - PCEW- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| PST - Built-Up Roofing - PHA- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Built-Up Roofing - TFA- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| PST - Built-Up Roofing - WPB- Exposed to Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| PST - Carbon Steel -CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -CDG- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel - CEHMS- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel Door - CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel Door - CDG- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A1-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel Door - EFP- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel Door - EFP- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel Door - FSB- Exposed to Air Indoor Uncontrolled | HELB Shielding | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A5-12 (T-11) | 3.5.1-25 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel Door - FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A5-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel Door - FSB- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A5-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel Door - FSB- in Air with Borated Water Leakage | HELB Shielding | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel Door - FSB- in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel Door - PAB- Exposed to Air Indoor Uncontrolled | HELB Shielding | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel Door - PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel Door - PAB- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel Door - PAB- in Air with Borated Water Leakage | HELB Shielding | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel Door - PAB- in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel Door - PCEW- Exposed to Air Indoor Uncontrolled | HELB Shielding | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel Door - PCEW- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel Door - PCEW- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel Door - WPB- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel Door - WPB- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel Door - WPB- in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel -EFP- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -EFP- Exposed to Air Outdoor | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel -EFP- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel Fire Door -CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| PST - Carbon Steel Fire Door -EFP- Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel Fire Door -FSB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| PST - Carbon Steel Fire Door -FSB- in Air with Borated Water Leakage | Fire Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel Fire Door -PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| PST - Carbon Steel Fire Door -PAB- in Air with Borated Water Leakage | Fire Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel Fire Door -PCEW- Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel Fire Door -WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| PST - Carbon Steel Fire Door -WPB- in Air with Borated Water Leakage | Fire Barrier | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel -FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A5-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -FSB- in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel -PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel -PAB- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel -PAB- in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel -PCEW- Exposed to Air Indoor Uncontrolled | Flood Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -PCEW- Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -PCEW- Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel -PHA-Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -PHA-Exposed to Air Outdoor | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel -PHA-Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel -TFA-Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -TFA-Exposed to Air Outdoor | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel -TFA-Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel -TFA- in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Carbon Steel -WPB-Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| PST - Carbon Steel -WPB-Exposed to Air Outdoor | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| PST - Carbon Steel -WPB-Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Carbon Steel -WPB- in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| PST - Conduit Fire Wrap – CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Aluminum | Air Indoor Uncontrolled (External) | None | None | III.B2-4 (TP-8) | 3.5.1-58 | A |
| PST - Conduit Fire Wrap – PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Aluminum | Air Indoor Uncontrolled (External) | None | None | III.B2-4 (TP-8) | 3.5.1-58 | A |
| PST - Conduit Fire Wrap – PAB- in Air with Borated Water Leakage | Fire Barrier | Aluminum | Air w/Borated Water Leakage (External)) | Loss of Material | Boric Acid Program | III.B2-6 (TP-3) | 3.5.1-55 | A |
| PST - Conduit Fire Wrap – WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Aluminum | Air Indoor Uncontrolled (External) | None | None | III.B2-4 (TP-8) | 3.5.1-58 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Conduit Fire Wrap – WPB- in Air with Borated Water Leakage | Fire Barrier | Aluminum | Air w/Borated Water Leakage (External)) | Loss of Material | Boric Acid Program | III.B2-6 (TP-3) | 3.5.1-55 | A |
| PST - Elastomers -CDG- Exposed to Air Indoor Uncontrolled | Control Bldg Habitability | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| PST - Elastomers -CDG- Exposed to Air Indoor Uncontrolled | Flood Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| PST - Elastomers -CDG- Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -CDG- Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Elastomers -EFP-Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -EFP-Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -FSB-Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -FSB-Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -PAB-Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Elastomers -PAB-Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -PCEW-Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -PCEW-Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -PHA-Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -PHA-Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X-1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Elastomers -TFA- Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -TFA- Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -WPB- Exposed to Air Outdoor | Expansion/ Separation | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Elastomers -WPB- Exposed to Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Fire Penetration Seal -CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Fire Penetration Seal -EFP- Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| PST - Fire Penetration Seal -FSB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| PST - Fire Penetration Seal -PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| PST - Fire Penetration Seal -PCEW- Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| PST - Fire Penetration Seal -WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Lubrite® Plate -PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Lubrite® | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | Structures Monitoring Program | III.B2-2 (TP-1) | 3.5.1-52 | A |
| PST - Reinforced Concrete - CDG- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - CDG- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - CDG- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - CDG- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - CDG- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - CDG- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - CDG- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A1-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - CDG- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A1-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - CDG- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - CDG- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - CDG- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A1-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - CDG- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A1-4 (T-05) | 3.5.1-31 | A, 512 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - CDG- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A1-2 (T-03) | 3.5.1-27 | A, 512 |
| PST - Reinforced Concrete - CEHMS- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - CEHMS- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - CEHMS- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - CEHMS- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - CEHMS- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - CEHMS- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - CEHMS- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - CEHMS- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - CEHMS- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - EFP- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - EFP- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - EFP- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - EFP- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - EFP- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - EFP- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - EFP- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - EFP- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - EFP- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - EFP- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - EFP- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - EFP- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 512 |
| PST - Reinforced Concrete - EFP- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A, 512 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - FSB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - FSB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - FSB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - FSB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - FSB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-5 (T-07) | 3.5.1-31 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - FSB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - FSB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A5-7 (T-02) | 3.5.1-31 | A, 509 |
| PST - Reinforced Concrete - FSB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A5-7 (T-02) | 3.5.1-31 | A, 509 |
| PST - Reinforced Concrete - FSB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A5-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - FSB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A5-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-9 (T-04) | 3.5.1-23 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A5-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - FSB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A5-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - FSB- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A5-4 (T-05) | 3.5.1-31 | A, 512 |
| PST - Reinforced Concrete - FSB- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A5-2 (T-03) | 3.5.1-27 | A, 512 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PAB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PAB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PAB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PAB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PAB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PAB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PAB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - PAB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - PAB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PAB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PAB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PCEW- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PCEW- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PCEW- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and. Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PCEW- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - PCEW- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PCEW- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**; Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | HELB Shielding | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PCEW- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PCEW- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 512 |
| PST - Reinforced Concrete - PCEW- Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A, 512 |
| PST - Reinforced Concrete - PHA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PHA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PHA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PHA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PHA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PHA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - PHA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PHA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - PHA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PHA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**
**Primary Structures**
**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - PHA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - PHA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - TFA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - TFA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - TFA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - TFA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - TFA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - TFA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - TFA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - TFA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - TFA- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - TFA- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - TFA- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - WPB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - WPB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| PST - Reinforced Concrete - WPB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - WPB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - WPB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - WPB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| PST - Reinforced Concrete - WPB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - WPB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| PST - Reinforced Concrete - WPB- Below Grade | Flood Barrier | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - WPB- Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Shielding | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Reinforced Concrete - WPB- Exposed to Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| PST - Stainless Steel -CDG- Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B2-8 (TP-5) | 3.5.1-59 | C |
| PST - Stainless Steel -CDG- Exposed to Raw Water | Shelter, Protection | Stainless Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | VII.H2-18 (Ap-55) | 3.3.1-80 | E, 512 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Stainless Steel - CEHMS- Exposed to Air Outdoor | Structural Support | Stainless Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.B2-7 (TP-6) | 3.5.1-50 | C |
| PST - Stainless Steel -FSB- Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B2-8 (TP-5) | 3.5.1-59 | C |
| PST - Stainless Steel -FSB- Exposed to Raw Water | Shelter, Protection | Stainless Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | VII.H2-18 (Ap-55) | 3.3.1-80 | E, 512 |
| PST - Stainless Steel -FSB- Exposed to Treated Borated Water | Shelter, Protection | Stainless Steel | Treated Borated Water (External) | Cracking | Water Chemistry Program | III.A5-13 (T-14) | 3.5.1-46 | A, 507 |
| PST - Stainless Steel -FSB- Exposed to Treated Borated Water | Shelter, Protection | Stainless Steel | Treated Borated Water (External) | Loss of Material | Water Chemistry Program | III.A5-13 (T-14) | 3.5.1-46 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Stainless Steel -FSB-Fuel Transfer Tube in Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-5) | 3.5.1-59 | A |
| PST - Stainless Steel -FSB-Fuel Transfer Tube in Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B2-8 (TP-5) | 3.5.1-59 | C |
| PST - Stainless Steel -FSB-in Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-5) | 3.5.1-59 | A |
| PST - Stainless Steel -PAB-Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B2-8 (TP-5) | 3.5.1-59 | C |
| PST - Stainless Steel -PAB-Exposed to Raw Water | Shelter, Protection | Stainless Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | VII.H2-18 (Ap-55) | 3.3.1-80 | E, 512 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Stainless Steel -PAB- in Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-5) | 3.5.1-59 | A |
| PST - Stainless Steel -WPB- Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B2-8 (TP-5) | 3.5.1-59 | C |
| PST - Stainless Steel -WPB- Exposed to Raw Water | Shelter, Protection | Stainless Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | VII.H2-18 (Ap-55) | 3.3.1-80 | E, 512 |
| PST - Stainless Steel -WPB- in Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B2-9 (TP-5) | 3.5.1-59 | A |
| PST - Tech Spec CEVA Seal -EFP- Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Tech Spec CEVA Seal -EFP- Exposed to Air Outdoor | Structural Pressure Barrier | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Tech Spec CEVA Seal -FSB- Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| PST - Tech Spec CEVA Seal -PAB- Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| PST - Tech Spec CEVA Seal -PAB- Exposed to Air Outdoor | Structural Pressure Barrier | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Tech Spec CEVA Seal -PCEW- Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST - Tech Spec CEVA Seal -PCEW- Exposed to Air Outdoor | Structural Pressure Barrier | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |
| PST - Tech Spec Control Room Seal -CDG- Exposed to Air Indoor Uncontrolled | Control Bldg Habitability | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| PST - Tech Spec Fuel Storage Building Seal -FSB- Exposed to Air Indoor Uncontrolled | Structural Pressure Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 515 |
| PST – Thermal Insulation Aluminum Jacketing -FSB- in Air with Borated Water Leakage | Structural Support | Aluminum | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-6 (TP-3) | 3.5.1-55 | A |
| PST – Thermal Insulation Aluminum Jacketing -PAB- in Air with Borated Water Leakage | Structural Support | Aluminum | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-6 (TP-3) | 3.5.1-55 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| PST – Thermal Insulation Aluminum Jacketing -TFA- in Air with Borated Water Leakage | Structural Support | Aluminum | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-6 (TP-3) | 3.5.1-55 | A |
| PST – Thermal Insulation Aluminum Jacketing -WPB- in Air with Borated Water Leakage | Structural Support | Aluminum | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Program | III.B2-6 (TP-3) | 3.5.1-55 | A |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-5**

**Primary Structures**

**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GALL; III.A6-12 is for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F); water chemistry and temperature will be maintained by the Water Chemistry Program. |

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

508 Cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program.

509 For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the same.

510 Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT.

511 At Seabrook Station, XI.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI.S5 "Masonry Wall Program" are combined under XI.S6 "Structures Monitoring Program".

512 Raw water in lined & unlined concrete sumps.

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

Control Building and Diesel Generator Building **(CDG)**, Containment Equipment Hatch Missile Shield **(CEHMS)**, Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Area (Control Building to Containment) and Pre-Action Valve Building **(EFP)**, Fuel Storage Building **(FSB)**, Primary Auxiliary Building and the Residual Heat Removal Equipment Vault **(PAB)**, Main Steam and Feedwater Pipe Chases – East & West **(PCEW)**, Personnel Hatch Area **(PHA)**, Tank Farm (Tunnels) - Including Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank **(TFA)**, Waste Process Building **(WPB)**

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| ASME Class 1 - Constant & Variable Load Spring Hangers - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | ASME Section XI, Subsection IWF Program | III.B1.1-2 (T-28) | 3.5.1-54 | A |
| ASME Class 1 - Constant & Variable Load Spring Hangers – in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B1.1-14 (T-25) | 3.5.1-55 | A |
| ASME Class 1 - Lubrite® - Exposed to Air Indoor Uncontrolled | Structural Support | Lubrite® | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | ASME Section XI, Subsection IWF Program | III.B1.1-5 (T-32) | 3.5.1-56 | A |
| ASME Class 1 - Pipe Supports - Concrete - Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B1.1-1 (T-29) | 3.5.1-40 | A |
| ASME Class 1 - Stainless Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B1.1-9 (TP-5) | 3.5.1-59 | A |
| ASME Class 1 - Stainless Steel - in Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | None | III.B1.1-10 (TP-4) | 3.5.1-59 | A |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| ASME Class 1 Support - Carbon Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWF Program | III.B1.1-13 (T-24) | 3.5.1-53 | A |
| ASME Class 1 Support - Carbon Steel – in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B1.1-14 (T-25) | 3.5.1-55 | A |
| ASME Class 2/3 - Constant and Variable Load Spring Hangers - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | ASME Section XI, Subsection IWF Program | III.B1.2-2 (T-28) | 3.5.1-54 | A |
| ASME Class 2/3 - Constant & Variable Load Spring Hangers – in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B1.2-11 (T-25) | 3.5.1-55 | A |
| ASME Class 2/3 - Lubrite® - Exposed to Air Indoor Uncontrolled | Structural Support | Lubrite® | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | ASME Section XI, Subsection IWF Program | III.B1.2-3 (T-32) | 3.5.1-56 | A |
| ASME Class 2/3 Pipe Supports - Concrete - Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B1.2-1 (T-29) | 3.5.1-40 | A |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| ASME Class 2/3 - Stainless Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B1.2-7 (TP-5) | 3.5.1-59 | A |
| ASME Class 2/3 - Stainless Steel - in Air with Borated Water Leakage | Structural Support | Stainless Steel | Air w/Borated Water Leakage (External) | None | Boric Acid Corrosion Program | III.B1.2-8 (TP-4) | 3.5.1-55 | A |
| ASME Class 2/3 - Stainless Steel - in Raw Water | Structural Support | Stainless Steel | Raw Water (External) | Loss of Material | ASME Section XI, Subsection IWF Program | III.B1.1-11 (TP-10) | 3.5.1-49 | H, 509, 514 |
| ASME Class 2/3 - Stainless Steel - in Raw Water | Structural Support | Stainless Steel | Raw Water (External) | Loss of Material | ASME Section XI, Subsection IWF Program | III.B1.1-11 (TP-10) | 3.5.1-49 | A, 509 |
| ASME Class 2/3 - Stainless Steel - in Treated Water | Structural Support | Stainless Steel | Treated Borated Water (External) | Loss of Material | Water Chemistry Program | III.A5-13 (T-14) | 3.5.1-46 | A |
| ASME Class 2/3 Support - Carbon Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | ASME Section XI, Subsection IWF Program | III.B1.2-10 (T-24) | 3.5.1-53 | A |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| ASME Class 2/3 Support Carbon Steel - in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B1.2-11 (TP-25) | 3.5.1-55 | A |
| Boral Poison Sheet in Spent Fuel Racks - in Treated Water | Absorb Neutrons | Boral | Treated Borated Water (External) | Reduction of Neutron Absorbing Capacity and Loss of Material | Boral Monitoring Program | VII.A2-5 (A-88) | 3.3.1-13 | A |
| Emergency Diesel Generator (EDG) - Concrete - Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B4-1 (T-29) | 3.5.1-40 | A |
| Emergency Diesel Generator (EDG) - Lubrite® - Exposed to Air Indoor Uncontrolled | Structural Support | Lubrite® | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | Structures Monitoring Program | III.B4-2 (TP-1) | 3.5.1-52 | A |
| Emergency Diesel Generator (EDG) Support - Carbon Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.B4-10 (T-30) | 3.5.1-39 | A |
| HVAC System Components - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B4-1 (T-29) | 3.5.1-40 | A |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| New Fuel Storage Racks Support - Carbon Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | VII.A1-1 (A-94) | 3.3.1-86 | A |
| New Fuel Storage Racks Support - Carbon Steel – in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B5-8 (T-25) | 3.5.1-55 | A |
| Non-ASME - Constant and Variable Load Spring Hangers - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | Structures Monitoring Program | III.B1.2-2 (T-28) | 3.5.1-54 | H, 514 |
| Non-ASME - Constant and Variable Load Spring Hangers – in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B1.2-11 (T-25) | 3.5.1-55 | A |
| Non-ASME Piping & Components - Concrete - Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B2-1 (T-29) | 3.5.1-40 | A |
| Non-ASME Piping & Components - Concrete - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B2-1 (T-29) | 3.5.1-40 | A |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Non-ASME Piping & Components - Lubrite® - Exposed to Air Indoor Uncontrolled | Structural Support | Lubrite® | Air Indoor Uncontrolled (External) | Loss of Mechanical Function | Structures Monitoring Program | III.B2-2 (TP-1) | 3.5.1-52 | A |
| Non-ASME - Stainless Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B2-8 (TP-5) | 3.5.1-59 | A |
| Non-ASME Support - Carbon Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.B2-10 (T-30) | 3.5.1-39 | A |
| Non-ASME Support - Carbon Steel - Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.B2-10 (T-30) | 3.5.1-39 | A |
| Non-ASME Support - Carbon Steel – in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B2-11 (T-25) | 3.5.1-55 | A |
| Panels - Aluminum – Exposed to Air Indoor Uncontrolled | Structural Support | Aluminum | Air Indoor Uncontrolled (External) | None | None | III.B3-2 (TP-8) | 3.5.1-58 | A |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Panels - Aluminum – in Air with Borated Water Leakage | Structural Support | Aluminum | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B3-4 (T-3) | 3.5.1-55 | A |
| Panels - Carbon Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.B3-7 (T-30) | 3.5.1-39 | A |
| Panels - Carbon Steel - Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.B3-7 (T-30) | 3.5.1-39 | A |
| Panels - Carbon Steel - in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B3-8 (T-25) | 3.5.1-55 | A |
| Panels - Concrete - Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B3-1 (T-29) | 3.5.1-40 | A |
| Panels - Concrete - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B3-1 (T-29) | 3.5.1-40 | A |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Platform - Concrete - Exposed to Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B5-1 (T-29) | 3.5.1-40 | A |
| Platform - Concrete - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Reduction in Concrete Anchor Capacity | Structures Monitoring Program | III.B5-1 (T-29) | 3.5.1-40 | A |
| Platform Supports - Carbon Steel - Exposed to Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.B5-7 (T-30) | 3.5.1-39 | A |
| Platform Supports - Carbon Steel - Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.B5-7 (T-30) | 3.5.1-39 | A, 503 |
| Platform Supports - Carbon Steel – in Air with Borated Water Leakage | Structural Support | Steel | Air w/Borated Water Leakage (External) | Loss of Material | Boric Acid Corrosion Program | III.B5-8 (T-25) | 3.5.1-55 | A |
| Spent Fuel Rack Support - Stainless Steel - in Treated Water | Structural Support | Stainless Steel | Treated Borated Water (External) | Cracking | Water Chemistry Program | VII.A2-7 (A-97) | 3.3.1-90 | A, 507 |

**Table 3.5.2-6**

**Supports**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Thermal Insulation Aluminum Jacketing - Exposed to Air Indoor Uncontrolled | Structural Support | Aluminum | Air Indoor Uncontrolled (External) | None | None | III.B2-4 (TP-8) | 3.5.1-58 | A |
| Thermal Insulation Stainless Steel Jacketing - Exposed to Air Indoor Uncontrolled | Structural Support | Stainless Steel | Air Indoor Uncontrolled (External) | None | None | III.B2-8 (TP-4) | 3.5.1-59 | A |

**Table 3.5.2-6**
**Supports**
**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GALL; III.A6-12 is for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F); water chemistry and temperature will be maintained by the Water Chemistry Program. |
| 508 | Cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program. |
| 509 | For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the |

same.

510 Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT.

511 At Seabrook Station, XI,.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI,.S5 "Masonry Wall Program" are combined under XI,.S6 "Structures Monitoring Program".

512 Raw water in lined & unlined concrete sumps.

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

**Table 3.5.2-7**

**Turbine Building**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| TUR - Aluminum - in Air Indoor Uncontrolled | Structural Support | Aluminum | Air Indoor Uncontrolled (External) | None | None | III.B3-2 | 3.5.1-58 | A |
| TUR - Built-Up Roofing - Exposed to Air Outdoor | Structural Support | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III .A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| TUR - Carbon Steel - Exposed to Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III .A3-12 (T-11) | 3.5.1-25 | A, 503 |
| TUR - Carbon Steel - in Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III .A3-12 (T-11) | 3.5.1-25 | A |
| TUR - Concrete (Sump) - Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III .A3-4 (T-05) | 3.5.1-31 | A, 509 |
| TUR - Concrete (Sump) - Exposed to Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III .A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-7**

**Turbine Building**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| TUR - Concrete - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III .A3-9 (T-04) | 3.5.1-23 | A |
| TUR - Concrete - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III .A3-2 (T-03) | 3.5.1-27 | A |
| TUR - Concrete - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III .A3-10 (T-06) | 3.5.1-24 | A |
| TUR - Concrete - Exposed to Air Outdoor | Structural Support | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III .A3-6 (T-01) | 3.5.1-26 | A |
| TUR - Concrete - in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III .A3-9 (T-04) | 3.5.1-23 | A |
| TUR - Concrete - in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III .A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-7**

**Turbine Building**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| TUR - Concrete - in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III .A3-10 (T-06) | 3.5.1-24 | A |
| TUR - Concrete Masonry Units - in Air Indoor Uncontrolled | Structural Support | Concrete Block | Air Indoor Uncontrolled (External) | Cracking | Structures Monitoring Program | III .A3-11 (T-12) | 3.5.1-43 | A |
| TUR - Fire Penetration Seal - in Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| TUR - Fire Penetration Seal - in Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-1 (A-19) | 3.3.1-61 | E, 514 |
| TUR - Penetration Seal - Exposed to Air Outdoor | Structural Support | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |

**Table 3.5.2-7**
**Turbine Building**
**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GALL; III.A6-12 is for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F); water chemistry and temperature will be maintained by the Water Chemistry Program. |
| 508 | Cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program. |
| 509 | For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the same. |
| 510 | Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT. |

511 At Seabrook Station, XI.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI.S5 "Masonry Wall Program" are combined under XI.S6 "Structures Monitoring Program".

512 Raw water in lined & unlined concrete sumps.

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Built-Up Roofing - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| WCS - Built-Up Roofing - SERVICE WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Roofing | Air Outdoor (External) | Separation, Environmental Degradation, Water In-Leakage | Structures Monitoring Program | III.A6-12 (TP-7) | 3.5.1-44 | H, 505 |
| WCS - Carbon Steel Door - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| WCS - Carbon Steel Door - SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| WCS - Carbon Steel Door - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| WCS - Carbon Steel Door - SERVICE WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Carbon Steel Fire Door - SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| WCS - Carbon Steel Fire Door - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-3 (A-21) | 3.3.1-63 | A |
| WCS - Carbon Steel - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Structural Support | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| WCS - Carbon Steel - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 511 |
| WCS - Carbon Steel - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 511 |
| WCS - Carbon Steel - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | H, 511 |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Carbon Steel - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 511 |
| WCS - Carbon Steel - SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| WCS - Carbon Steel - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| WCS - Carbon Steel - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 511 |
| WCS - Carbon Steel - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 511 |
| WCS - Carbon Steel - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | H, 511 |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Carbon Steel - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Steel | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-11 (T-21) | 3.5.1-47 | E, 511 |
| WCS - Carbon Steel - SERVICE WATER PUMPHOUSE Siding in Air Indoor Uncontrolled | Shelter, Protection | Steel | Air Indoor Uncontrolled (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A |
| WCS - Carbon Steel - SERVICE WATER PUMPHOUSE Siding in Air Outdoor | Shelter, Protection | Steel | Air Outdoor (External) | Loss of Material | Structures Monitoring Program | III.A3-12 (T-11) | 3.5.1-25 | A, 503 |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - CIRCULATING WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-7 (T-20) | 3.5.1-45 | E, 511 |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Ultimate Heat Sink | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Ultimate Heat Sink | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Concrete | Raw Water (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Ultimate Heat Sink | Concrete | Raw Water (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A |
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Structural Support | Concrete | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-7 (T-20) | 3.5.1-45 | E, 511 |
| WCS – Concrete - SERVICE WATER COOLING TOWER in Raw Water | Ultimate Heat Sink | Concrete | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-7 (T-20) | 3.5.1-45 | E, 511 |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete -SERVICE WATER COOLING TWR Including Swgr Rms in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - SERVICE WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-5 (T-07) | 3.5.1-31 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A, 509 |
| WCS - Concrete - SERVICE WATER PUMPHOUSE Below Grade | Structural Support | Concrete | Ground Water/Soil (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Loss of Material | Fire Protection Program | VII.G-29 (A-91) | 3.3.1-67 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Concrete Cracking and Spalling | Fire Protection Program | VII.G-28 (A-90) | 3.3.1-65 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Structural Support | Concrete | Air Indoor Uncontrolled (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-9 (T-04) | 3.5.1-23 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-10 (T-06) | 3.5.1-24 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Missile Barrier | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Concrete | Air Outdoor (External) | Loss of Material, Cracking | Structures Monitoring Program | III.A3-6 (T-01) | 3.5.1-26 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Cracking, Loss of Bond, Loss of Material (Spalling, Scaling) | Structures Monitoring Program | III.A3-4 (T-05) | 3.5.1-31 | A, 509 |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Expansion and Cracking | Structures Monitoring Program | III.A3-2 (T-03) | 3.5.1-27 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Increase in Porosity and Permeability, Loss of Strength | Structures Monitoring Program | III.A3-7 (T-02) | 3.5.1-32 | A |
| WCS - Concrete - SERVICE WATER PUMPHOUSE in Raw Water | Structural Support | Concrete | Raw Water (External) | Loss of Material | Structures Monitoring Program | III.A6-7 (T-20) | 3.5.1-45 | E, 511 |
| WCS - Fire Penetration Seal - SERVICE WATER COOLING TWR including Swgr Rms in Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| WCS - Fire Seal - SERVICE WATER PUMPHOUSE in Air Indoor Uncontrolled | Fire Barrier | Elastomer | Air Indoor Uncontrolled (External) | Increased Hardness, Shrinkage and Loss of Strength | Fire Protection Program | VII.G-1 (A-19) | 3.3.1-61 | A |
| WCS - Penetration Seal - SERVICE WATER COOLING TOWER Including Switchgear Rooms in Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |

**Table 3.5.2-8**

**Water Control Structures**

**Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG 1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| WCS - Penetration Seal - SERVICE WATER PUMPHOUSE in Air Outdoor | Shelter, Protection | Elastomer | Air Outdoor (External) | Increased Hardness, Shrinkage and Loss of Strength | Structures Monitoring Program | VII.G-2 (A-20) | 3.3.1-61 | E, 515 |

**Table 3.5.2-8**
**Water Control Structures**
**Summary of Aging Management Evaluation**

**Standard Notes**

| Note | Description |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program. |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |
| 501 | Not used. |
| 502 | Aging effect includes "Fretting or Lockup" due to wear. |
| 503 | Crevice and pitting will be included along with loss of material-corrosion due to a saltwater atmosphere environment. |
| 504 | Fatigue analysis exists and TLAA applies. |
| 505 | Built-up roofing is not in GALL; III.A6-12 is for elastomer-material is similar, aging effect is similar, environment is same, and AMP is Structures Monitoring. |
| 506 | Component is cementitious fire proofing/insulating material and will exhibit similar aging effects as concrete. |
| 507 | Spent Fuel Pool temperature < 60°C (<140° F); water chemistry and temperature will be maintained by the Water Chemistry Program. |
| 508 | Cracking, loss of bond, and loss of material (spalling, scaling)/corrosion of embedded steel-is not listed in GALL III.A.6 as an aging effect for concrete in raw water. Seabrook manages this effect with Structures Monitoring Program. |
| 509 | For aging management purposes, buried, below grade, soil, and ground water/ raw & treated water environments are treated the same. |
| 510 | Reduction in concrete anchor capacity is an aging effect that is addressed in LRAM-SUPT. |
| 511 | At Seabrook Station, XI.S7 "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" and XI.S5 "Masonry Wall Program" are combined under XI.S6 "Structures Monitoring Program". |

512 Raw water in lined & unlined concrete sumps.

513 Seabrook Station will age manage this condition through the Fire Protection Program.

514 Seabrook Station will age manage this condition through the Structures Monitoring Program.

515 Increased hardness, shrinkage, or loss of strength of elastomer seals due to weathering is addressed by GALL only for Fire Barrier seals. Seabrook Station will manage such aging effects for non-Fire Barrier elastomer seals with the Structures Monitoring Program.

516 Seabrook Station Structures Monitoring Program will perform concrete testing and rebar inspection to determine the effects of the aggressive groundwater on the concrete. The concrete testing and the rebar inspection will represent all concrete below grade.

## 3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

### 3.6.1 INTRODUCTION

This section provides the results of the aging management review for the electrical commodity groups identified in Section 2.5, Scoping and Screening Results: Electrical and Instrumentation and Control (I&C) Systems/Commodity Groups.

The electrical and instrumentation and control commodity groups requiring aging management review are listed below. The following sections identify materials, environments, aging effects requiring management and associated aging management programs (AMPs) for each electrical commodity group identified in Section 2.5.4.

- Non-EQ Electrical Cables and Connections
- Metal Enclosed Bus
- Fuse Holders (Not Part of a Larger Assembly) Metallic Clamps
- Cable Connections (Metallic Parts)
- $SF_6$ Insulated Bus, Connections and Insulators

### 3.6.2 RESULTS

The following tables summarize the results of the aging management review.

- Table 3.6-1 Summary Of Aging Management Evaluations for the Electrical / I&C Components / Commodities
- Table 3.6-2 Summary Of Aging Management Evaluations - Electrical / I&C Components / Commodities

### 3.6.2.1 Materials, Environments, Aging Effects Requiring Management and Aging Management Programs

#### 3.6.2.1.1 Non-EQ Electrical Cables and Connections

**Materials**

The materials of construction for the Non-Environmentally Qualified (Non-EQ) electrical cables and connections are:

- Various Organic Polymers

**Environments**

The Non-EQ electrical cables and connections are exposed to the following environments:

- Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen
- Adverse localized environment caused by exposure to moisture and voltage

**Aging Effects Requiring Management**

The following aging effects associated with Non-EQ electrical cable and connections require management:

- Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure
- Localized damage and breakdown of insulation leading to electrical failure

**Aging Management Programs**

The following aging management programs manage the aging effects for the Non-EQ electrical cables and connections:

- Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (B.2.1.32)

- Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (B.2.1.33)
- Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (B.2.1.34)

#### 3.6.2.1.2 Metal Enclosed Bus

**Materials**

The materials of construction for the Metal Enclosed Bus are:

- Aluminum
- Copper
- Elastomer
- Epoxy
- Polyester glass
- Porcelain
- Silver plated aluminum
- Silver plated copper
- Stainless Steel
- Steel

**Environments**

The Metal Enclosed Bus are exposed to the following environments:

- Air - Indoor Uncontrolled
- Air-Outdoor

**Aging Effects Requiring Management**

The following aging effects associated with Metal Enclosed Bus require management:

- Loosening of bolted connections
- Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure.
- Moisture and debris intrusion

**Aging Management Programs**

The following aging management programs manage the aging effects for the Metal Enclosed Bus

- Metal Enclosed Bus (B.2.1.35)
- Structures Monitoring Program (B.2.1.31)

### 3.6.2.1.3 Fuse Holders (Not Part of a Larger Assembly) Metallic Clamps

**Materials**

The materials of construction of Fuse Holders (Not Part of a Larger Assembly) Metallic Clamps are:

- Various conductive metals including copper alloy.

**Environments**

The metallic clamps of Fuse Holders (Not Part of a Larger Assembly) Metallic Clamps are exposed to the following environments:

- Air Indoor Uncontrolled

**Aging Effects Requiring Management**

The following aging effect associated with Fuse Holders (Not Part of a Larger Assembly) Metallic Clamps requires management:

- Thermal fatigue in the form of high resistance

**Aging Management Programs**

The following aging management programs manage the aging effects for the Fuse Holders (Not Part of a Larger Assembly) Metallic Clamps:

- Fuse Holders (B.2.1.36)

#### 3.6.2.1.4 Cable Connections (Metallic Parts)

**Materials**

The materials of construction for the Cable Connections (Metallic Parts) are:

- Various conductive metals used for electrical contacts

**Environments**

The Cable Connections (Metallic Parts) are exposed to the following environments:

- Air Indoor Controlled
- Air Indoor Uncontrolled
- Air Outdoor
- Air with Borated Water Leakage

**Aging Effects Requiring Management**

The following aging effects associated with Cable Connections (Metallic Parts) require management:

- Loosening of bolted connections
- Corrosion of connector contact surfaces

**Aging Management Programs**

The following aging management programs manage the aging effects for Cable Connections (Metallic Parts):

- Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements (B.2.1.37)
- Boric Acid Aging Management Program (B.2.1.4)

### 3.6.2.1.5 $SF_6$ Insulated Bus, Connections and Insulators

**Materials**

The materials of construction for the Sulfur Hexafluoride ($SF_6$) Insulated Bus, Connections and Insulators are:

- Aluminum
- Elastomer
- Epoxy
- Gas ($SF_6$)
- Stainless Steel
- Silver Plated Aluminum

**Environments**

The $SF_6$ Insulated Bus, Connections and Insulators are exposed to the following environments:

- Gas ($SF_6$)
- Air-Outdoor

**Aging Effects Requiring Management**

The following aging effects associated with $SF_6$ Insulated Bus, Connections and Insulators require management:

- Loss of pressure boundary
- Loss of insulating properties due to changes in $SF_6$ properties.

**Aging Management Programs**

The following aging management programs manage the aging effects for the $SF_6$ Bus:

- 345 kV $SF_6$ Bus (B.2.2.1)

### 3.6.2.2 AMR Results for Which Further Evaluation is Recommended by the GALL Report

NUREG-1801 provides the basis for identifying those programs that warrant further evaluation by the reviewer in the License Renewal Application (LRA). For the Electrical / I&C commodities, those programs are addressed in the following subsections.

#### 3.6.2.2.1 Electrical Equipment Subject to Environmental Qualification

Environmental qualification is a Time Limited Aging Analysis (TLAA) as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed in Section 4.4, "Environmental Qualification (EQ) of Electric Equipment," of this application.

#### 3.6.2.2.2 Degradation of Insulator Quality due to Presence of Any Salt Deposits and Surface Contamination, and Loss of Material due to Mechanical Wear

The $SF_6$ switchyard connects Seabrook Station to the off site transmission grid. The $SF_6$ bus is included as part of the recovery path in the event of a Station Blackout event.

The design of the $SF_6$ switchyard does not include high voltage insulators that are commonly associated with an open air switchyard design.

**Conclusion**

The Seabrook Station switchyard design does not include high voltage insulators, therefore the aging mechanisms and effects are not applicable.

#### 3.6.2.2.3 Loss of Material due to Wind Induced Abrasion and Fatigue, Loss of Conductor Strength due to Corrosion, and Increased Resistance of Connection due to Oxidation or Loss of Pre-load

The $SF_6$ switchyard connects Seabrook Station to the off site transmission grid.

The in-scope portion of the $SF_6$ switchyard does not include transmission conductors and connections or switchyard bus and connections that are commonly associated with an open air switchyard design.

**Conclusion**

The in-scope portion of the Seabrook Station switchyard design does not include transmission conductors and connections or switchyard bus and connections, therefore the aging mechanisms and effects are not applicable.

#### 3.6.2.2.4 Quality Assurance for Aging Management of Nonsafety-Related Components

Quality Assurance (QA) provisions applicable to License Renewal are discussed in Section B.1.3.

### 3.6.2.3 AMR Results Not Consistent with or Not Addressed in the GALL Report

The 345kV $SF_6$ switchyard connects Seabrook Station to the off site transmission grid. The $SF_6$ bus is included as part of the recovery path in the event of a Station Blackout event. As discussed in Section 3.6.2.2.2 and 3.6.2.2.3, the design of the $SF_6$ switchyard does not include high voltage insulators and transmission lines that are normally associated with an open air switchyard. The $SF_6$ bus is a phase isolated and independent bus in which each phase conductor is enclosed by an individual metal housing separated from adjacent conductor housings by an air space. The conductor is centered in the housing by insulators. The insulating parameters are accomplished by maintaining the space achieved by the insulators and the insulating properties of the $SF_6$ gas.

The critical conditions which are essential to the bus operation are maintaining the pressure boundary and the air, moisture and sulfur dioxide ($SO_2$) content.

The presence of moisture could lead to electrical failure and the presence of $SO_2$ is an indication of partial discharge in the system.

The external surface of the $SF_6$ bus is managed for loss of material. The pressure boundary and the quality of the $SF_6$ gas is managed by the 345kV $SF_6$ Bus Aging Management Program (B.2.2.1).

### 3.6.3 CONCLUSION

The electrical commodity groups that are subject to aging management review have been identified in accordance with the scoping criteria of 10 CFR 54.4. Aging effects have been identified based on plant and industry operating experience as well as industry literature. Programs to manage these aging effects have been identified in this section, and detailed program descriptions are provided in Appendix B. These activities demonstrate that the aging effects associated with the electrical commodity groups will be adequately managed such that there is reasonable assurance that the intended functions will be maintained consistent with the current licensing basis during the period of extended operation.

**Table 3.6.1**

**Summary of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Item Number | Component | Aging Effect/Mechanism | Aging Management Programs | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.6.1-1 | Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements | Degradation due to various aging mechanisms | Environmental Qualification Of Electric Components | Yes, TLAA (See Subsection 3.6.2.2.1) | Environmental Qualification (EQ) of Electric Components is a TLAA. Further evaluation is documented in Section 4.4 and Subsection 3.6.2.2.1 |
| 3.6.1-2 | Electrical cables, connections and fuse holders (insulation) not subject to 10 CFR 50.49 EQ requirements | Reduced insulation resistance and electrical failure due to various physical, thermal, radiolytic, photolytic, and chemical mechanisms | Electrical Cables and Connections Not Subject To 10 CFR 50.49 EQ Requirements | No | Consistent with NUREG-1801. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, B.2.1.32, will be used to manage reduced insulation resistance and electrical failure due to various mechanisms, in adverse localized environments, for insulated cables and connections, including connection insulation for splices, terminal blocks and fuse holders. |
| 3.6.1-3 | Conductor insulation for electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance (IR) | Reduced insulation resistance and electrical failure due to various physical, thermal, radiolytic, photolytic, and chemical mechanisms | Electrical Cables And Connections Used In Instrumentation Circuits Not Subject To 10 CFR 50.49 EQ Requirements | No | Consistent with NUREG-1801. This AMP manages the aging of the Nuclear Instrumentation cables. Radiation Monitoring cables are included in the EQ Program.The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits Program, B.2.1.33, will be used to manage reduced insulation resistance and electrical failure, due to various mechanisms, in adverse localized environments, for insulated cables and connections used in nuclear instrumentation circuits. |

**Table 3.6.1**

**Summary of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Item Number | Component | Aging Effect/Mechanism | Aging Management Programs | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.6.1-4 | Conductor insulation for inaccessible medium voltage (2 kV to 35 kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements | Localized damage and breakdown of insulation leading to electrical failure due to moisture intrusion, water trees | Inaccessible Medium Voltage Cables Not Subject To 10 CFR 50.49 EQ Requirements | No | Consistent with NUREG-1801. The Inaccessible Medium Voltage Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program, B.2.1.34, will be used to manage localized damage and breakdown of insulation leading to electrical failure, due to moisture intrusion and water trees, in adverse localized environments, for medium voltage cables. |
| 3.6.1-5 | Connector contacts for electrical connectors exposed to borated water leakage | Corrosion of connector contact surfaces due to intrusion of borated water | Boric Acid Corrosion | No | Consistent with NUREG-1801. The Boric Acid Corrosion Program, B.2.1.4, will be used to manage corrosion of connector contact surfaces for electrical connectors exposed to borated water leakage. |
| 3.6.1-6 | Fuse Holders (Not Part of a Larger Assembly): Fuse holders – metallic clamp | Fatigue due to ohmic heating, thermal cycling, electrical transients, frequent manipulation, vibration, chemical contamination, corrosion, and oxidation | Fuse Holders | No | Consistent with NUREG-1801. The Fuse Holders Program, B.2.1.36, will manage, increase of resistance due to corrosion, and oxidation for fuse holders metallic clamp. Fatigue due to ohmic heating, thermal cycling, electrical transients, frequent manipulation, vibration, chemical contamination are not viable aging effects. See Note 603 of Table 3.6.2 |
| 3.6.1-7 | Metal enclosed bus - Bus/connections | Loosening of bolted connections due to thermal cycling and ohmic heating | Metal Enclosed Bus | No | Consistent with NUREG-1801. The Metal Enclosed Bus Program, B.2.1.35, will be used to manage the aging effect of loosening of bolted connections for the metal enclosed bus. |

**Table 3.6.1**

**Summary of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Item Number | Component | Aging Effect/Mechanism | Aging Management Programs | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.6.1-8 | Metal enclosed bus – Insulation/insulators | Reduced insulation resistance and electrical failure due to various physical, thermal, radiolytic, photolytic, and chemical mechanisms | Metal Enclosed Bus | No | Consistent with NUREG-1801. The Metal Enclosed Bus Program, B.2.1.35, will be used to manage the aging effects of reduced insulation resistance and electrical failure for metal enclosed bus |
| 3.6.1-9 | Metal enclosed bus – Enclosure assemblies | Loss of material due to general corrosion | Structures Monitoring Program | No | Consistent with NUREG-1801. The Structures Monitoring Program, B.2.1.31, will be used to manage the aging effect of loss of material due to general corrosion for metal enclosed bus. |
| 3.6.1-10 | Metal enclosed bus – Enclosure assemblies | Hardening and loss of strength due to elastomer degradation | Structures Monitoring Program | No | Consistent with NUREG-1801. The Structures Monitoring Program, B.2.1.31, will be used to manage the aging effects of hardening and loss of strength due to elastomer degradation for metal enclosed bus. |
| 3.6.1-11 | High voltage insulators | Degradation of insulation quality due to presence of any salt deposits and surface contamination, Loss of material caused by mechanical wear due to wind blowing on transmission conductors | A plant-specific aging management program is to be evaluated. | Yes, plant specific (See subsection 3.6.2.2.2) | The Seabrook Station design does not contain high-voltage insulators that are typically associated with an open air switchyards. (See subsection 3.6.2.2.2) |

**Table 3.6.1**

**Summary of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Item Number | Component | Aging Effect/Mechanism | Aging Management Programs | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.6.1-12 | Transmission conductors and connections, Switchyard bus and connections | Loss of material due to wind induced abrasion and fatigue, Loss of conductor strength due to corrosion, Increased resistance of connection due to oxidation or loss of preload | A plant-specific aging management program is to be evaluated. | Yes, plant specific (See subsection 3.6.2.2.3) | The Seabrook Station design does not contain transmission conductors and connections, or switchyard bus and connections that are typically associated with an open air switchyards (See subsection 3.6.2.2.3) |
| 3.6.1-13 | Cable Connections – Metallic parts | Loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation | Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements | No | Consistent with NUREG-1801. The Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program, B.2.1.37, will be used to manage loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation of the metallic parts of cable connections. The Seabrook Station AMP is consistent with the final issue of LR ISG 2007-02. |
| 3.6.1-14 | Fuse Holders (Not Part of a Larger Assembly) Insulation material | None | None | No Aging Effect Requiring Management or Aging Management Program is required. | Consistent with NUREG-1801. |

**Table 3.6.2-1**

**Summary Of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Cable Connections (Metallic Parts) | Electrical Continuity | Various metals used for electrical contacts | Air – indoor controlled<br>Air – indoor uncontrolled<br>Air-outdoor | Loosening of bolted connections | Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | VI.A-1 (LP-12) | 3.6.1-13 | A<br>601 |
| Non-EQ Electrical Cables and Connections | Electrical Continuity | Various organic polymers (e.g., EPR, SR, EPDM, XLPE) | Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen | Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure | Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | VI.A-2 (L-01) | 3.6.1-2 | A |
| Non-EQ Electrical Cables and Connections used in instrumentation circuits that are sensitive to reduction in conductor insulation resistance (IR) | Electrical Continuity | Various organic polymers (e.g., EPR, SR, EPDM, XLPE) | Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen | Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure | Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits | VI.A-3 (L-02) | 3.6.1-3 | A<br>602 |

**Table 3.6.2-1**

**Summary Of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Non-EQ inaccessible medium-voltage cables | Electrical Continuity | Various organic polymers (e.g., EPR, SR, EPDM, XLPE) | Adverse localized environment caused by exposure to moisture and voltage | Localized damage and breakdown of insulation leading to electrical failure | Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | VI.A-4 (L-03) | 3.6.1-4 | A |
| Cable Connections (Metallic Parts) (Connector contacts for electrical connectors exposed to borated water Leakage) | Electrical Continuity | Various metals used for electrical contacts | Air with borated water leakage | Corrosion of connector contact surfaces | Boric Acid Corrosion | VI.A-5 (L-04) | 3.6.1-5 | A |
| Electrical Equipment Subject to 10 CFR 50.49 EQ Requirements | Electrical Continuity | Various Polymeric and Metallic Materials | Adverse localized environment | Various degradation | Environmental Qualification (EQ) of Electric Components | VI.B-1 | 3.6.1-1 | A |
| Fuse Holders (Not Part of a Larger Assembly); Insulation | Electrical Continuity | Insulation material – bakelite, phenolic melamine or ceramic, molded polycarbonate and other | Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen or > 60-year service limiting temperature | Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure | Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | VI.A-6 (LP-03) | 3.6.1-2 | A |

**Table 3.6.2-1**

**Summary Of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Fuse Holders (Not Part of a Larger Assembly); Insulation | Electrical Continuity | Insulation material – bakelite, phenolic melamine or ceramic, molded polycarbonate and other | Air – indoor uncontrolled (Internal/External) | None | None | VI.A-7 (LP-02) | 3.6.1-14 | A |
| Fuse Holders (Not Part of a Larger Assembly); Metallic Clamp | Electrical Continuity | Various conductive metals including copper alloy | Air – indoor | Thermal fatigue in the form of high resistance | Fuse Holders | VI.A-8 (LP-01) | 3.6.1-6 | A 603 |
| Metal Enclosed Bus Bus/connections | Electrical Continuity | Aluminum Copper Stainless steel, steel | Air – indoor Uncontrolled<br>Air -outdoor | Loosening of bolted connections | Metal Enclosed Bus | VI.A-11 (LP-04) | 3.6.1-7 | A 604 |
| Metal Enclosed Bus Enclosure assemblies | Support | Elastomers | Air – indoor Uncontrolled<br>Air -outdoor | Hardening and loss of strength | Structures Monitoring Program | VI.A-12 (LP-10) | 3.6.1-10 | A |
| Metal Enclosed Bus Enclosure assemblies | Support | Steel / Aluminum | Air – indoor Uncontrolled<br>Air -outdoor | Loss of material | Structures Monitoring Program | VI.A-13 (LP-06) | 3.6.1-9 | A |
| Metal Enclosed Bus Insulation/insulators | Insulation - Electrical | Porcelain | Air – indoor Uncontrolled<br>Air -outdoor | Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure | Metal Enclosed Bus | VI.A-14 (LP-05) | 3.6.1-8 | A |

**Table 3.6.2-1**

**Summary Of Aging Management Evaluations for the Electrical / I&C Components / Commodities**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-1801 Vol. 2 Item | Table 3.X.1 Item | Note |
|---|---|---|---|---|---|---|---|---|
| Metal Enclosed Bus Insulation/insulators | Insulation - Electrical | Polyester glass, Epoxy | Air – indoor Uncontrolled<br><br>Air -outdoor | Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure | Metal Enclosed Bus | VI.A-14 (LP-05) | 3.6.1-8 | F |
| $SF_6$ Insulated Bus, Connections and Insulators (Insulation/insulators) | Insulation - Electrical | $SF_6$ gas Epoxy | Gas ($SF_6$) (internal) | Loss of dielectric strength | 345 KV $SF_6$ Bus | None | None | J |
| $SF_6$ Insulated Bus, Connections and Insulators (Enclosure Assemblies) | Pressure Boundary | Aluminum Stainless Steel | Gas ($SF_6$) (internal)<br><br>Air-Outdoor (external) | Loss of material | 345 KV $SF_6$ Bus | None | None | J |
| $SF_6$ Insulated Bus, Connections and Insulators (Enclosure Assemblies) | Pressure Boundary | Elastomer | Gas ($SF_6$) (internal)<br><br>Air-Outdoor (external) | Hardening and loss of strength | 345 KV $SF_6$ Bus | None | None | J |

**Standard Notes:**

| | |
|---|---|
| A | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| B | Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP. |
| C | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP. |
| D | Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP |
| E | Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program |
| F | Material not in NUREG-1801 for this component. |
| G | Environment not in NUREG-1801 for this component and material. |
| H | Aging effect not in NUREG-1801 for this component, material and environment combination. |
| I | Aging effect in NUREG-1801 for this component, material and environment combination is not applicable. |
| J | Neither the component nor the material and environment combination is evaluated in NUREG-1801. |

**Plant Specific Notes:**

601. This AMP complies with the final issue LR ISG 2007-02.
602. This AMP manages the aging of the Nuclear Instrumentation cable. Radiation Monitoring cables are managed by the EQ Program.
603. Fatigue due to ohmic heating, thermal cycling, electrical transients, frequent manipulation, vibration and chemical contamination does not require an AMP. Increased resistance due to corrosion or oxidation does require an AMP.
604. This portion of the AMP applies to the intermediate bolted connections on the non-segregated bus only. The connections on the isolated phase bus are welded.

# CHAPTER 4

# TIME-LIMITED AGING ANALYSES

# 4.0 TIME-LIMITED AGING ANALYSES

Chapter 4 describes the Time-Limited Aging Analyses (TLAAs) for Seabrook Station Unit 1 in accordance with 10 CFR 54.3(a) and 54.21(c). Subsequent sections describe TLAAs within these common general categories:

- Reactor Vessel Neutron Embrittlement Analysis (4.2)
- Metal Fatigue Analysis of Piping And Components (4.3)
- Environmental Qualification (EQ) of Electric Components (4.4)
- Absence of TLAA for Concrete Containment Tendon Prestress (4.5)
- Containment Liner Plate Fatigue Usage and Containment Penetration Pressurization Cycles (4.6)
- Plant-Specific Time Limited Aging Analyses (4.7)

The information on each specific TLAA within these general categories is organized under three subsections:

**Summary Description**

A brief description of the TLAA topic and affected components will be presented.

**Analysis**

The current licensing basis (CLB) analysis of the TLAA including implications of the period of extended operation (PEO).

**Disposition**

The disposition of the TLAA for the PEO, in accordance with 10 CFR 54.21(c)(1):

- Validation - 10 CFR 54.21(c)(1)(i) - The analysis remains valid for the period of extended operation.
- Revision - 10 CFR 54.21(c)(1)(ii) - The analysis has been projected to the end of the period of extended operation, or
- Aging Management - 10 CFR 54.21(c)(1)(iii) - The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

In some instances, a discussion is provided regarding the absence of a TLAA for components at Seabrook Station. In these cases a conclusion is presented in lieu of a disposition statement.

## 4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

### 4.1.1 BACKGROUND

10 CFR 54.3 and 10 CFR 54.21 address time-limited aging analyses (TLAAs) in license renewal applications. 10 CFR 54.21(c) provides the following content requirements for TLAAs:

*(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.*

10 CFR 54.3 defines a time-limited aging analysis as:

*Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:*

*(1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);*

*(2) Consider the effects of aging;*

*(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;*

*(4) Were determined to be relevant by the licensee in making a safety determination;*

*(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and*

*(6) Are contained or incorporated by reference in the CLB.*

### 4.1.2 METHODOLOGY

The process used to identify time-limited aging analyses for Seabrook Station is consistent with the guidance provided in NEI 95-10. Site-specific analyses and evaluations that could potentially meet the six criteria of 10 CFR 54.3 were identified by performing electronic keyword searches and by manually searching current licensing basis documents, including:

- Updated Final Safety Analysis Report (UFSAR)
- Docketed licensing correspondence
- Design Basis Documents
- Analyses, calculations, evaluation reports, and capsule surveillance reports from Westinghouse
- Applicable Westinghouse Owner's Group (WOG) reports, analyses, and supporting calculations
- Site-specific analyses, calculations, and evaluations
- Technical Specifications and Technical Specification Bases Documents
- Previous Applicant's License Renewal Applications

Industry documents that list generic time-limited aging analyses were also reviewed to provide additional assurance of the completeness of the plant-specific list. These documents included Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Vol. 2, Rev. 1, Standard Review Plan for License Renewal, NUREG-1800, Chapter 4, Rev. 1, NEI 95-10, Industry Guidance for Implementing the Requirements of 10 CFR 54 the License Renewal Rule, and previously submitted License Renewal Applications from other plants.

NUREG-1801 identifies numerous aging effects that require evaluation as possible TLAAs in accordance with 10 CFR 54.21(c). Each of these was reviewed in the appropriate aging management review, or in this chapter, and dispositioned as a TLAA if identified as such under the 10 CFR 54.3(a) criteria. Tables 3.1.1, 3.2.1, 3.3.1, 3.4.1, 3.5.1 and 3.6.1, as discussed in Section 3.0, Aging Management Reviews, list the TLAA line items of the NUREG-1801 Volume 1 Summary Tables, and identify the specific sections relating to the required further evaluations.

### 4.1.3 IDENTIFICATION OF EXEMPTIONS

**Summary Description**

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed operating license should include a list of unit-specific exemptions granted pursuant to 10 CFR 50.12, that are in effect, based on time-limited aging analysis, as defined in 10 CFR 54.3. Each active exemption has been reviewed to determine exemptions that are based on time-limited aging analyses.

The UFSAR, Facility Operating License, Safety Evaluation Report and associated supplements, and docketed correspondence was searched for active exemptions granted under 10 CFR 50.12 to determine exemptions that are based on time-limited aging analyses.

**Analysis**

The NextEra Energy Seabrook Facility Operating License identifies two exemptions granted pursuant to 10 CFR 50.12.

- NextEra Energy Seabrook, LLC, is exempt from the Section III.D.2(b)(ii) containment airlock testing requirements of Appendix J to 10 CFR 50, because of the special circumstances described in Section 6.2.6 of SER Supplement 5 and authorized by 10 CFR 50.12(a)(2)(ii) and (iii) (51 FR 37684 October 23, 1986).

- NRC Materials License No. SNM-1963, issued December 19, 1985, granted an exemption pursuant to 10 CFR 70.24 with respect to requirements for criticality alarms. NextEra Energy Seabrook, LLC, is hereby exempted from provisions of 10 CFR 70.24 insofar as this section applies to the storage and handling of new fuel assemblies in the new fuel storage vault, spent fuel pool (when dry), and shipping containers.

UFSAR Section 3.9(N) identifies an exemption from a portion of 10 CFR Part 50, "*Domestic Licensing of Production and Utilization Facilities*" Appendix A, General Design Criterion 4 "*Environmental and Dynamic Effects Design Bases*". Acceptance of this exemption is documented in NUREG-0896 Supplement 5, Appendix K. The exemption permitted Seabrook Station to eliminate the protective devices and the dynamic loading effects associated with the postulated primary loop pipe breaks for Seabrook Station, Units 1. The exemption was limited until the completion of the second refueling outage pending outcome of commission rulemaking regarding Leak-Before Break analysis.

### Conclusion

No active exemptions granted pursuant to 10 CFR 50.12 and based on a time-limited aging analysis, as defined in 10 CFR 54.3, have been identified for Seabrook Station.

### 4.1.4 SUMMARY OF RESULTS

Table 4.1-1: Time-Limited Aging Analyses Applicable to Seabrook Station, summarizes TLAAs identified within the current licensing basis of Seabrook Station. This table provides a list of Seabrook Station TLAAs with the disposition method used for each and the section of the LRA where each is discussed.

NUREG-1800, Table 4.1-2 and NUREG-1800, Table 4.1-3, list examples of potential TLAAs, depending upon the applicant's current licensing basis (CLB). Table 4.1-2: Review of Analyses Listed in NUREG-1800 Tables 4.1-2 and 4.1-3 summarizes the results of the Seabrook Station applicability review of these potential TLAAs and the section of the LRA where each is discussed.

| Table 4.1-1 Time-Limited Aging Analyses Applicable to Seabrook Station | | | |
|---|---|---|---|
| **TLAA Category** | **Description** | **Disposition Method(s)** | **LRA Section** |
| **1.** | **Reactor Vessel Neutron Embrittlement** | | **4.2** |
| | Neutron Fluence Analyses | §54.21(c)(1)(ii) | 4.2.1 |
| | Upper Shelf Energy Analyses | §54.21(c)(1)(ii) | 4.2.2 |
| | Pressurized Thermal Shock Analyses | §54.21(c)(1)(ii) | 4.2.3 |
| | Reactor Vessel Pressure-Temperature Limits, Including Low Temperature Overpressure Protection Limits | §54.21(c)(1)(iii) | 4.2.4 |
| **2.** | **Metal Fatigue Of Piping And Components** | | **4.3** |
| | Nuclear Steam Supply System (NSSS) Pressure Vessel and Component Fatigue Analyses | §54.21(c)(1)(i) | 4.3.1 |
| | Supplementary ASME Section III, Class 1 Piping and Component Fatigue Analyses | §54.21(c)(1)(i) | 4.3.2 |
| | Absence of a TLAA for Thermal Stresses in Piping Connected to Reactor Coolant Systems: NRC Bulletin 88-08 | N/A | 4.3.2.1 |
| | NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification | §54.21(c)(1)(i) | 4.3.2.2 |
| | Reactor Vessel Internal Aging Management | §54.21(c)(1)(i) | 4.3.3 |
| | Environmentally-Assisted Fatigue Analyses | §54.21(c)(1)(ii)<br>§54.21(c)(1)(iii) | 4.3.4 |
| | Steam Generator Tube, Loss of Material and Fatigue from Flow-Induced Vibration | §54.21(c)(1)(i) | 4.3.5 |
| | Absence of TLAAs for Fatigue Crack Growth, Fracture Mechanics Stability, or Corrosion Analyses Supporting Repair of Alloy 600 Materials | N/A | 4.3.6 |
| | Non-Class 1 Component Fatigue Analyses | §54.21(c)(1)(i) | 4.3.7 |
| **3.** | **Environmental Qualification of Electric Components** | §54.21(c)(1)(iii) | **4.4** |
| **4.** | **Absence of TLAA for Concrete Containment Tendon Prestress** | **N/A** | **4.5** |
| **5.** | **Containment Liner Plate Fatigue Usage and Containment Penetration Pressurization Cycles** | | **4.6** |
| | Containment Liner Plate Fatigue Usage | §54.21(c)(1)(ii) | 4.6.1 |
| | Pressurization Cycles: Personnel Airlock, Equipment Hatch and Fuel Transfer Tube Assembly Absence of TLAA for Containment Penetrations | §54.21(c)(1)(i) | 4.6.2 |

| Table 4.1-1 Time-Limited Aging Analyses Applicable to Seabrook Station | | | |
|---|---|---|---|
| TLAA Category | Description | Disposition Method(s) | LRA Section |
| 6 | **Plant-Specific Time Limited Aging Analyses** | | 4.7 |
| | Absence of a TLAA for Reactor Vessel Underclad Cracking Analyses | N/A | 4.7.1 |
| | Reactor Coolant Pump Flywheel Fatigue Crack Growth Analyses | §54.21(c)(1)(i) | 4.7.2 |
| | Leak-Before Break Analyses | §54.21(c)(1)(i) | 4.7.3 |
| | High Energy Line Break (HELB) Postulation Based on Cumulative Usage Factor | §54.21(c)(1)(i) | 4.7.4 |
| | Fuel Transfer Tube Bellows Design Cycles | §54.21(c)(1)(i) | 4.7.5 |
| | Crane Load Cycle Limits | §54.21(c)(1)(i) | 4.7.6 |
| | Polar Gantry Crane | §54.21(c)(1)(i) | 4.7.6.1 |
| | Cask Handling Crane | §54.21(c)(1)(i) | 4.7.6.2 |
| | Service Level I Coatings Qualification | §54.21(c)(1)(iii) | 4.7.7 |
| | Absence of a TLAA for Reactor Coolant Pump Code: Case N-481 | N/A | 4.7.8 |
| | Canopy Seal Clamp Assemblies | §54.21(c)(1)(i) | 4.7.9 |
| | Hydrogen Analyzer | §54.21(c)(1)(i) | 4.7.10 |
| | Mechanical Equipment Qualification | §54.21(c)(1)(ii) | 4.7.11 |
| | Absence of a TLAA for Metal Corrosion Allowance | N/A | 4.7.12 |
| | Absence of a TLAA for Inservice Flaw Growth Analyses that Demonstrate Structural Stability for 40 years | N/A | 4.7.13 |
| | Diesel Generator Thermal Cycle Evaluation | §54.21(c)(1)(i) | 4.7.14 |

**Table 4.1-2 Review of Analyses Listed in NUREG-1800 Tables 4.1-2 and 4.1-3**

| NUREG-1800 Examples | Applicability to Seabrook | LRA Section |
|---|---|---|
| **NUREG-1800, Table 4.1-2 – Examples of Potential TLAAs** | | |
| Reactor vessel neutron embrittlement | Yes | 4.2 |
| Concrete containment tendon prestress | No | 4.5 |
| Metal Fatigue | Yes | 4.3 |
| Environmental qualification of electrical equipment | Yes | 4.4 |
| Metal corrosion allowance | No | 4.7.12 |
| Inservice flaw growth analyses that demonstrate structure stability for 40 years | No | 4.7.13 |
| Inservice local metal containment corrosion analyses | No | N/A |
| High-energy line break postulation based on fatigue CUF | Yes | 4.7.4 |
| **NUREG-1800, Table 4.1-3 – Additional Examples of Plant-Specific TLAAs** | | |
| Intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic SS cladding | No | 4.7.1 |
| Low-temperature overpressure protection (LTOP) analyses | Yes | 4.2.4 |
| Fatigue analyses for the main steam supply lines to the turbine-driven auxiliary feedwater pumps | Yes | 4.3.7 |
| Fatigue analyses for the reactor coolant pump flywheel | Yes | 4.7.2 |
| Fatigue analysis of polar crane | Yes | 4.7.6.1 |
| Flow-induced vibration endurance limit, for the reactor vessel internals | Yes | 4.3.3 |
| Transient cycle count assumptions for the reactor vessel internals | | |
| Ductility reduction of fracture toughness for the reactor vessel internals | | |
| Leak-before-break | Yes | 4.7.3 |
| Fatigue analysis for the containment liner plate | Yes | 4.6.1 |
| Concrete penetration pressurization cycles | Yes | 4.6.2 |
| Reactor vessel circumferential weld inspection relief (BWR) | N/A | N/A |

## 4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

Carbon and low-alloy steels exposed to high levels of high-energy neutron irradiation exposure (fluence) are susceptible to reduction of fracture toughness, an increase in material strength and decrease in ductility. Fracture toughness is temperature dependent, and is indirectly measured in foot-pounds of absorbed energy in a Charpy impact test. In most materials, toughness increases with temperature up to a maximum value called Upper Shelf Energy (USE). Neutron embrittlement is measured in terms of Charpy transition temperature shift, Charpy upper-shelf energy decrease, and yield and ultimate tensile strength increase. Neutron embrittlement varies with material but is directly dependent upon the integrated total neutron exposure for energy levels above 1 MeV. Based upon the materials and projected fluence levels, the only reactor vessel shell items expected to be susceptible to neutron embrittlement are the reactor vessel shell components in the beltline region immediately surrounding the core.

In order to reduce the potential for brittle fracture during reactor vessel operation, Pressure-Temperature (P-T) limit curves are developed that require the reactor vessel temperature to reach specified minimum limits prior to the application of significant pressure loading to assure the materials have adequate ductility to resist the loads. Since these minimum temperatures are increased as a function of predicted cumulative fluence, the reduced material toughness as a function of fluence is offset. Adequate fracture toughness is assured at or above the minimum temperatures specified by the P-T limit curves.

In order to develop P-T limit curves, a number of tests and calculations must first be performed. The initial nil-ductility reference temperature ($RT_{NDT}$) is the temperature at which a material transitions from brittle-to-ductile behavior, and this temperature is determined for each reactor vessel beltline material prior to neutron exposure. Samples of each material are tested again after various degrees of neutron exposure up to end-of- life (EOL) fluence levels to determine how much this transition temperature will increase during plant operation as a function of neutron irradiation. This is performed as part of the reactor vessel surveillance program, and the acceptable fluence intervals for these tests are specified by ASTM E-185 requirements. This increase or shift in the nil-ductility reference temperature ($\Delta RT_{NDT}$) is the amount of temperature increase required for the material to continue to act in a ductile manner for a given fluence level. The P-T curves are periodically updated for an incremental fluence increase using the initial $RT_{NDT}$ and $\Delta RT_{NDT}$ values associated with the fluence value used, along with appropriate uncertainty margins. As the actual plant exposure approaches the fluence value used in a particular set of P-T limit curves, new curves are prepared for higher fluence values, up to the EOL fluence value.

For Seabrook Station, the reactor vessel material $\Delta RT_{NDT}$ and USE values, calculated on the basis of predicted 40-year End-of-Life (EOL) neutron fluence, are determined as part of the current licensing basis, and support safety determinations. Therefore, these calculations are TLAAs. For license renewal, these must be updated to account for the fluence expected to occur during 60 years of plant operation (55 Effective Full Power Years). The governing requirements for these updated analyses are summarized below.

NRC Regulations 10 CFR 50.60 and 10 CFR 50.61 provides fracture toughness requirements and acceptance criteria applicable to the Seabrook Station reactor vessel. NRC Regulation 10 CFR 50.60, "*Acceptance Criteria For Fracture Prevention Measures For Light Water Nuclear Power Reactors For Normal Operation,*" requires that all light water nuclear power reactors meet the requirements of 10 CFR 50, Appendix G, "*Fracture Toughness Requirements,*" and 10 CFR 50, Appendix H, "*Reactor Vessel Material Surveillance Program Requirements.*" Appendix G specifies fracture toughness requirements for the reactor coolant pressure boundary to provide margins of safety against fracture during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The Seabrook Station Reactor Vessel Integrity Surveillance Program, B.2.1.19 is required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Materials and fluence data obtained from this program are used in these fracture toughness analyses.

NRC Regulation 10 CFR 50.61, "*Fracture toughness requirements for protection against pressurized thermal shock events,*" provides requirements for computing the reference temperature, $RT_{PTS}$, for the (EOL) fluence for each of the reactor vessel beltline materials, which is a measure of the fracture toughness after exposure to EOL fluence. It also provides a Pressurized Thermal Shock (PTS) screening criterion for each type of beltline material, which limits how high the minimum reference temperature can be raised. The $RT_{PTS}$ screening criteria serve as limits on the degree of $\Delta RT_{NDT}$ that can be applied to account for neutron embrittlement. The $RT_{PTS}$ values are a function of material composition and neutron fluence, and they increase as cumulative fluence increases, possibly approaching the screening criterion if the material is highly susceptible to neutron embrittlement. If the $RT_{PTS}$ value is projected to exceed the screening criterion using the EOL fluence, licensees are required to implement flux reduction programs to prevent this from occurring.

## 4.2.1 NEUTRON FLUENCE ANALYSES

### Summary Description

The current license period reactor vessel embrittlement analyses that evaluate reduction of fracture toughness of the Seabrook Station reactor vessel beltline materials are based on predicted 40-year EOL fluence values. The fluence analysis and the neutron embrittlement analyses that are based upon the fluence analysis are TLAAs as defined by 10 CFR 54.21(c) that must be evaluated for the increased neutron fluence associated with 60 years of operation. These TLAAs include the analyses for fracture toughness, or upper shelf energy, Pressurized Thermal Shock limits, $RT_{NDT}$ (nil-ductility transition temperature), Adjusted Reference Temperatures (ART), Low-Temperature Overpressure Protection limits, and Reactor Vessel Pressure-Temperature limit curves. The neutron fluence TLAA is evaluated in this subsection, and the others are evaluated in subsections 4.2.2, 4.2.3, and 4.2.4.

### Analysis

#### Estimation of EFPY for Seabrook Station based on 60-years of Plant Life

End-of-Life fluence is based on a predicted value of EFPY over the life of the plant. Seabrook Station began commercial operation on August 19, 1990. As of October 30, 2009, Seabrook Station has been operated for approximately 17 EFPY. If Seabrook Station is operated at the maximum licensed power level at a 100% capacity factor between outages until the end of period of extended operation on March 15, 2050, Seabrook Station will reach approximately 55 EFPY. This capacity factor is based on assumed outage durations of twenty (20) days during refueling outages and 100 percent power levels at all times other than during these outages.

#### 60-Year Neutron Fluence Projections

For license renewal, Seabrook Station updated fluence projections based upon 55 EFPY as input, to the neutron embrittlement analyses prepared for 60 years of operation.

The reactor vessel beltline neutron fluence values for 60 years of operation were calculated for the Seabrook Station reactor pressure vessel beltline material. The analysis methods used to calculate the predicted 60-year Seabrook Station vessel fluence values satisfy the requirements set forth in Regulatory Guide 1.190, "*Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*". In compliance with these guidelines, comparisons to surveillance capsule flux wire and dosimeter measurements were performed to determine the accuracy of the RPV fluence model. An uncertainty analysis was also performed to determine if a statistical bias exists in the model. It was determined that the Seabrook Station fluence model does not have a statistical bias and that the best-estimate fluence presented is suitable for use in evaluating the effects of embrittlement on RPV material as specified in (CFR) 10 CFR 50, Appendix G, *"Fracture Toughness Requirements"* and NRC Regulatory Guide 1.99, *"Radiation Embrittlement of Reactor Vessel Materials"*.

The fluence values were calculated using the RAMA Fluence Methodology (RAMA). The RAMA Fluence Methodology was developed for the Electric Power Research Institute, Inc. (EPRI) for the purpose of calculating fast neutron fluence in reactor pressure vessels and vessel internal components. As prescribed in NRC Regulatory Guide 1.190, RAMA has been benchmarked against industry standard benchmarks for both pressurized water reactor (PWR) and boiling water reactor designs. In addition, RAMA has been compared with several plant-specific dosimetry measurements and reported fluence from several commercial operating reactors. The results of the benchmarks and comparisons to measurements show that RAMA accurately predicts specimen activities, RPV fluence, and vessel component fluence in all light water reactor types. Under funding from EPRI and the Boiling Water Reactor Vessel and Internals Project, the RAMA methodology has been reviewed by the NRC and subsequently given generic approval for determining fast neutron fluence in boiling water reactor pressure vessels and vessel internal components that include the core shroud and top guide. This prior work has been extended in the Seabrook Station analysis to additional PWR benchmarks and plant-specific dosimetry comparisons, further validating the use of RAMA for all light water reactor designs.

In accordance with 10 CFR 50, Appendix H, *"Reactor Vessel Material Surveillance Program Requirements,"* any materials exceeding neutron fluence exposure greater than $1.0 \times 10^{17}$ n/cm$^2$ (E > 1.0 MeV) must be evaluated for changes in fracture toughness due to neutron embrittlement. Reactor pressure vessel materials that were not originally considered to be part of the vessel beltline, because neutron radiation exposure was less than $1.0 \times 10^{17}$ n/cm$^2$ (E > 1.0 MeV), were evaluated to determine their cumulative neutron radiation exposure at 55 EFPY. Fluence calculations were performed for the Seabrook Station reactor pressure vessel upper shell plates and welds to determine if they would exceed a neutron radiation exposure of $1.0 \times 10^{17}$ n/cm$^2$ (E > 1.0 MeV) at 55 EFPY. The materials that exceed this threshold were included as the extended beltline materials. Table 4.2.1-1 summarizes the results of the fluence projections to 55 EFPY for Seabrook Station. Refer to Table 4.2.3-1 for the Heat Numbers associated with the Seabrook Station vessel materials. It should be noted that the intermediate/upper shell circumferential weld, the upper shell plates, and axial welds are part of the extended beltline. The nozzles and the nozzle welds are projected to be below $1x10^{17}$ $n/cm^2$ (E > 1 MeV) at 55 EFPY and will not become part of the extended beltline.

| Table 4.2.1-1 55 EFPY Surface Fluence Projections for Beltline and Extended Beltline Materials For Seabrook Station | |
|---|---|
| Reactor Vessel Location | Seabrook Station 55 EFPY Fluence ($n/cm^2$, E > 1.0 MeV) |
| Lower Shell #1 (R1808-2) | 3.59E+19 |
| Lower Shell #2 (R1808-1) | 3.59E+19 |
| Lower Shell #3 (R1808-3) | 3.59E+19 |
| Intermediate Shell #1 (R1806-2) | 3.63E+19 |
| Intermediate Shell #2 (R1806-1) | 3.63E+19 |
| Intermediate Shell #3 (R1806-3) | 3.63E+19 |
| Upper Shell #1 (R1807-1)(1) | 8.92E+17 |
| Upper Shell #2 (R1807-2)(1) | 8.92E+17 |
| Upper Shell #3 (R1807-3)(1) | 8.92E+17 |
| Int./Lower Shell Circ. Weld (101-171) | 3.59E+19 |
| Int./Upper Shell Circ. Weld (103-121)(1) | 8.22E+17 |
| Upper Shell Axial Weld #1 (42°)(1) | 8.88E+17 |
| Upper Shell Axial Weld #2 (162°)(1) | 6.09E+17 |
| Upper Shell Axial Weld #3 (282°)(1) | 5.24E+17 |
| Intermediate Shell Axial Weld #1 (0°) | 2.06E+19 |
| Intermediate Shell Axial Weld #2 (120°) | 2.40E+19 |
| Intermediate Shell Axial Weld #3 (240°) | 2.40E+19 |
| Lower Shell Axial Weld #1 (90°) | 2.05E+19 |
| Lower Shell Axial Weld #2 (210°) | 3.46E+19 |
| Lower Shell Axial Weld #3 (330°) | 3.46E+19 |

(1) Extended Beltline Region

## Disposition

**Revision, 10 CFR 54.21(c)(1)(ii)** – The fluence analyses have been projected to the end of the period of extended operation. The materials to be included in the extended beltline requiring additional evaluation have been identified.

## 4.2.2 UPPER SHELF ENERGY ANALYSES

### Summary Description

The current Charpy Upper Shelf Energy (USE) analyses were prepared for each reactor vessel beltline material for Seabrook Station based upon projected neutron fluence values for 40 years of service. These are TLAAs requiring evaluation using the projected 60-year fluence values.

### Analysis

Title 10 CFR Part 50 Appendix G "*Fracture Toughness Requirements*" contains screening criteria that establish limits on how far the USE value for a reactor pressure vessel material may be allowed to decrease due to neutron irradiation exposure. The regulation requires the initial USE value to be greater than 75 ft-lbs in the non-irradiated condition and that the value is greater than 50 ft-lbs in the fully irradiated conditions as determined by Charpy V-notch testing on pulled capsules throughout the licensed life of the plant.

Per Regulatory Guide 1.99, Revision 2, the Charpy USE should be assumed to decrease as a function of fluence, according to Figure 2 of the Regulatory Guide, when surveillance data is not used (Position 1.2 of the Regulatory Guide). If surveillance data is used, the decrease in USE may be obtained by plotting the reduced plant surveillance data on Figure 2 of the Regulatory Guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all of the data (Position 2.2 of the Regulatory Guide). Charpy USE for the beltline forgings and welds and for the extended beltline materials was evaluated without the use of surveillance data which was determined to be conservative.

Predictions of the Charpy USE for EOL (55 EFPY) are summarized in Table 4.2.2-1 for Seabrook Station, using the corresponding 1/4T fluence projection, the copper content of the beltline materials and using Figure 2 in Regulatory Guide 1.99.

The USE values for the beltline and extended beltline materials are projected to remain above the 50 ft-lbs requirement through the period of extended operation for Seabrook Station as indicated in Table 4.2.2-1.

| Table 4.2.2-1 Predicted USE Values at 55 EFPY for Seabrook Station Vessel Beltline Materials | | | | | |
|---|---|---|---|---|---|
| RPV Material | Cu (%) | 1/4T Fluence ($E^{19}$ $n/cm^2$) | Initial USE (ft-lb) | USE Decrease (%) | USE (ft-lb) |
| Lower Shell #1 (R1808-2) | 0.06 | 2.14 | 77 | 22.7 | 59.5 |
| Lower Shell #2 (R1808-1) | 0.06 | 2.14 | 78 | 22.7 | 60.3 |
| Lower Shell #3 (R1808-3) | 0.07 | 2.14 | 78 | 22.7 | 60.3 |
| Intermediate Shell #1 (R1806-2) | 0.06 | 2.16 | 102 | 22.8 | 78.8 |
| Intermediate Shell #2 (R1806-1) | 0.045 | 2.16 | 82 | 22.8 | 63.3 |
| Intermediate Shell #3 (R1806-3) | 0.075 | 2.16 | 115 | 22.8 | 88.8 |
| Upper Shell #1 (R1807-1) | 0.08 | 0.0468 | 66 | 9.2 | 59.9 |
| Upper Shell #2 (R1807-2) | 0.09 | 0.0468 | 66.5 | 9.2 | 60.4 |
| Upper Shell #3 (R1807-3) | 0.06 | 0.0468 | 107 | 9.2 | 97.2 |
| Int./Lower Shell Circumferential Weld (101-171) | 0.047 | 2.14 | 160 | 22.7 | 123.7 |
| Int./Upper Shell Circumferential Weld (103-121) | 0.045 | 0.0489 | 147 | 9.3 | 133.3 |
| Upper Shell Axial Weld #1 (42°) | 0.05 | 0.0529 | >97.7 | 9.5 | 88.4 |
| Upper Shell Axial Weld #2 (162°) | 0.05 | 0.0363 | >97.7 | 8.7 | 89.2 |
| Upper Shell Axial Weld #3 (282°) | 0.05 | 0.0312 | >97.7 | 8.3 | 89.5 |
| Intermediate Shell Axial Weld #1 (0°) | 0.047 | 1.23 | 160 | 19.9 | 128.2 |
| Intermediate Shell Axial Weld #2 (120°) | 0.047 | 1.43 | 160 | 20.6 | 127 |
| Intermediate Shell Axial Weld #3 (240°) | 0.047 | 1.43 | 160 | 20.6 | 127 |
| Lower Shell Axial Weld #1 (90°) | 0.047 | 1.22 | 160 | 19.9 | 128.2 |
| Lower Shell Axial Weld #2 (210°) | 0.047 | 2.06 | 160 | 22.5 | 124 |
| Lower Shell Axial Weld #3 (330°) | 0.047 | 2.06 | 160 | 22.5 | 124 |

### Disposition

**Revision, 10 CFR 54.21(c)(1)(ii)** – The USE analyses have been projected to the end of the period of extended operation and the resulting USE values for all of the vessel beltline materials have each been demonstrated to exceed the minimum acceptance limit of 50 ft-lbs.

### 4.2.3 PRESSURIZED THERMAL SHOCK ANALYSES

#### Summary Description

Title 10 CFR Part 50.61(b)(1) provides rules for the protection of PWRs against pressurized thermal shock (PTS). Licensees are required to assess the projected values of nil-ductility reference temperature whenever a significant change occurs in the projected values of $RT_{PTS}$, or upon request for a change in the expiration date for the facility operating license. The current $RT_{PTS}$ analyses, evaluated for 32 EFPY fluence values predicted for 40 years of operation, are TLAAs requiring evaluation for 60 years.

#### Analysis

Reactor vessel beltline fluence is one of the factors used in determining the margin of acceptability of the reactor vessel to PTS as a result of neutron embrittlement. The margin is the difference between the maximum nil-ductility reference temperature in the limiting beltline material and the screening criteria established in accordance with 10 CFR 50.61(b)(2). The screening criteria for the limiting reactor vessel materials are 270°F for beltline plates, forgings, and axial weld materials, and 300°F for beltline circumferential weld materials.

In a letter submittal to NRC dated January 18, 1999, a revision to the limiting Seabrook Station reactor vessel beltline material was documented. That submittal identified the limiting material to be plate R1808-1 having a calculated $RT_{PTS}$ @ EOL of 120°F. This $RT_{PTS}$ value corresponds to fluence at the reactor vessel inside diameter of $2.37 \times 10^{19}$ n/cm$^2$. The results of the new $RT_{PTS}$ analyses, evaluated for 55 EFPY (or 60 years of operation), are presented in Table 4.2.3-1. The limiting $RT_{PTS}$ value for the Seabrook Station axially-oriented welds and plates is 123.3°F, which corresponds to the Lower Shell #2 plate (R1808-1) with a projected fluence at 55 EFPY of $3.59 \times 10^{19}$ n/cm$^2$. The limiting $RT_{PTS}$ value for the Seabrook Station circumferentially-oriented welds at 55 EFPY is 8°F, which corresponds to the Intermediate-to-Lower Shell Circumferential Weld Seam (101-171). All of the Seabrook Station reactor vessel materials that have a surface fluence value exceeding $1.0 \times 10^{17}$ n/cm$^2$ (E > 1.0 MeV) at 55 EFPY have been demonstrated to have an $RT_{PTS}$ value less than the applicable screening criterion, which is 270°F for plates, forgings, and axially-oriented welds (longitudinal welds), and is 300°F for circumferentially-oriented welds. Therefore, the $RT_{PTS}$ analyses have been satisfactorily projected for 60 years of operation.

| Table 4.2.3-1 Calculation of $RT_{PTS}$ Values for 55 EFPY at the Clad/Base Metal Interface for Seabrook Station | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| Reactor Vessel Beltline Region Location | Heat | Cu (%) | Ni (%) | Chem Factor (°F) | Surface Fluence ($E^{19}$ $n/cm^2$) | $\Delta RT_{PTS}$ (°F) | $RT_{NDT(U)}$ (°F) | Margin (°F) | $RT_{PTS}$ (°F) | $RT_{PTS}$ (°F) Acceptance Criteria |
| Lower Shell #1 (R1808-2) | D1081-2 | 0.06 | 0.58 | 37.0[1] | 3.59 | 49.3 | 10 | 34.0 | 93.3 | 270 |
| Lower Shell #2 (R1808-1) | D1081-3 | 0.06 | 0.58 | 37.0[1] | 3.59 | 49.3 | 40 | 34.0 | 123.3 | 270 |
| Lower Shell #3 (R1808-3) | D1136-2 | 0.07 | 0.59 | 45.0[2] | 3.59 | 60.0 | 40 | 17.0 | 117.0 | 270 |
| Intermediate Shell #1 (R1806-2) | A2749-2 | 0.06 | 0.64 | 37.0[1] | 3.63 | 49.4 | 0 | 34.0 | 83.4 | 270 |
| Intermediate Shell #2 (R1806-1) | C4036-2 | 0.045 | 0.61 | 28.5[1] | 3.63 | 38.0 | 40 | 34.0 | 112.0 | 270 |
| Intermediate Shell #3 (R1806-3) | C4197-1 | 0.075 | 0.63 | 47.5[1] | 3.63 | 63.4 | 10 | 34.0 | 107.4 | 270 |
| Upper Shell #1 (R1807-1) | C4049-1 | 0.08 | 0.60 | 51.0[1] | 0.0892 | 20.1 | 30 | 20.1 | 70.2 | 270 |
| Upper Shell #2 (R1807-2) | C4049-2 | 0.09 | 0.61 | 58.0[1] | 0.0892 | 22.9 | 30 | 22.9 | 75.7 | 270 |
| Upper Shell #3 (R1807-3) | C4235-2 | 0.06 | 0.67 | 37.0[1] | 0.0892 | 14.6 | 10 | 14.6 | 39.2 | 270 |
| Int./Lower Shell Circumferential Weld (101-171) | 4P6052 | 0.047 | 0.049 | 30.0[2] | 3.59 | 40.0 | -60 | 28.0 | 8.0 | 300 |
| Int./Upper Shell Circumferential Weld (103-121) | 90128 | 0.045 | 0.06 | 31.3[1] | 0.0822 | 11.9 | -56 | 36.0 | -8.1 | 300 |
| Upper Shell Axial Weld #1 (42°) | 86998 | 0.05 | 0.11 | 38.7[1] | 0.0888 | 15.2 | -10 | 15.2 | 20.5 | 270 |
| Upper Shell Axial Weld #2 (162°) | 86998 | 0.05 | 0.11 | 38.7[1] | 0.0609 | 12.6 | -10 | 12.6 | 15.2 | 270 |
| Upper Shell Axial Weld #3 (282°) | 86998 | 0.05 | 0.11 | 38.7[1] | 0.0524 | 11.6 | -10 | 11.6 | 13.2 | 270 |
| Intermediate Shell Axial Weld #1 (0°) | 4P6052 | 0.047 | 0.049 | 30.0[2] | 2.06 | 35.9 | -60 | 28.0 | 3.9 | 270 |
| Intermediate Shell Axial Weld #2 (120°) | 4P6052 | 0.047 | 0.049 | 30.0[2] | 2.40 | 37.1 | -60 | 28.0 | 5.1 | 270 |
| Intermediate Shell Axial Weld #3 (240°) | 4P6052 | 0.047 | 0.049 | 30.0[2] | 2.40 | 37.1 | -60 | 28.0 | 5.1 | 270 |
| Lower Shell Axial Weld #1 (90°) | 4P6052 | 0.047 | 0.049 | 30.0[2] | 2.05 | 35.9 | -60 | 28.0 | 3.9 | 270 |
| Lower Shell Axial Weld #2 (210°) | 4P6052 | 0.047 | 0.049 | 30.0[2] | 3.46 | 39.7 | -60 | 28.0 | 7.7 | 270 |
| Lower Shell Axial Weld #3 (330°) | 4P6052 | 0.047 | 0.049 | 30.0[2] | 3.46 | 39.7 | -60 | 28.0 | 7.7 | 270 |

Notes:
1. Table (Position 1.1)
2. Surveillance Non-Ratio (Position 2.1)

### Disposition

**Revision, 10 CFR 54.21(c)(1)(ii)** – The $RT_{PTS}$ analyses have been projected to the end of the period of extended operation and are shown to be within the maximum allowable PTS screening criteria limits.

## 4.2.4 REACTOR VESSEL PRESSURE–TEMPERATURE LIMITS, INCLUDING LOW TEMPERATURE OVERPRESSURE PROTECTION LIMITS

### Summary Description

Title 10 Part 50, Appendix G, "*Fracutre Toughness Requirements*" requires that the reactor pressure vessel be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel is exposed to increased neutron irradiation, its fracture toughness is reduced. The P-T limits must account for the anticipated reactor vessel fluence.

The calculations associated with the operating P-T limit curves involve time-limited assumptions defined by the current operating term, for example, 40 years, and they must satisfy the criteria of 10 CFR 54.3(a) for license renewal. For example, the P-T limit curves are based upon fluence values associated with 40 years of operation. Therefore, P-T limit curves are TLAAs requiring evaluation for 60 years of operation.

### Analysis

The provisions of 10 CFR 50, Appendix G, require Seabrook to operate within the currently licensed P-T limit curves. These curves are required to be maintained and updated as necessary to maintain plant operation consistent with 10 CFR 50. The Reactor Vessel Integrity Surveillance Program maintains the P-T limit curves for the period of extended operation. Prior to the period of extended operation, updated P-T limit calculations will be prepared using fluence values valid for the Seabrook Station reactor vessel beltline region materials, inlet and outlet nozzles, and closure head flange locations for normal heatup, normal cooldown, and in-service leak and hydrostatic test conditions. The current heatup and cooldown limit curves are valid for 20 EFPY. In determining the allowable operating pressure-temperature limits, the minimum bolt-up temperatures, minimum temperature of core criticality, pressure test limits and low-temperature overpressure protection (LTOP) system limits are determined. These P-T limits are expressed in the form of a set of curves of allowable pressure versus temperature (P-T limit curves). These curves are updated on a periodic basis to account for increasing vessel fluence.

Heatup and cooldown P-T limit curves for 55 EFPY will be prepared using the most limiting value of $RT_{NDT}$ (reference nil ductility transition temperature) corresponding to the limiting material in the beltline region of the reactor vessel. This is determined by using the unirradiated reactor vessel material fracture toughness properties adjusted to account for the estimated irradiation-induced shift ($\Delta RT_{NDT}$).

$RT_{NDT}$ increases as the material is exposed to fast-neutron flux. Therefore, to find the most limiting $RT_{NDT}$ at any time period in the reactor's life, $\Delta RT_{NDT}$ due to the radiation exposure associated with that time period must be added to the original unirradiated $RT_{NDT}$. Using the Adjusted Reference Temperature (ART) values, pressure-temperature limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G, as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code.

The 1/4 and 3/4 thickness (1/4T and 3/4T) fluences and material properties were used to determine the limiting material and calculate its pressure-temperature limits at 55 EFPY, which is bounding for the end of the period of extended operation. The limiting materials were determined from the values of ART at the 1/4T and 3/4T locations and are summarized in Table 4.2.4-1 for Seabrook Station.

**Table 4.2.4-1 Summary of the Limiting ART Values to be used in Generation of the Seabrook Station Reactor Vessel Heatup and Cooldown Curves through 55 EFPY**

| EFPY | 1/4T Limiting ART | 3/4T Limiting ART |
|---|---|---|
| | Lower Shell Plate R1808-1 | Lower Shell Plate R1808-1 |
| 55 | 118.6 °F | 108.1 °F |

Seabrook Station P-T limit curves for normal heatup and cooldown of the primary reactor coolant system at 20 EFPY were developed utilizing the 1995 Edition through the 1996 Addenda of the ASME Code Section XI, Appendix G methodology and Code Case N-641. Code Case N-641 provides alternative procedures for calculating the allowable pressure-temperature relationships and LTOP effective temperatures. Code Case N-641 divided Article G-2215 of the 1998 through the 2000 Summer Addenda Edition of Section XI, Appendix G into Articles G-2215.1 and G-2215.2 for allowable pressures and the LTOP System, respectively. Section 2215.1 of Code Case N-641 replaced all $K_{IA}$ designations with $K_{IC}$, thus removing the option to use the more restrictive $K_{IA}$ reference toughness. Article G-2215.2 provided the methodology to determine the LTOP system effective temperature.

The LTOP system provides Reactor Coolant System (RCS) pressure relief capability when system temperature is below 290°F. Two (2) pressurizer power operated relief valves (PORVs) provide the automatic relief capability during the design basis transients and automatically prevent RCS pressure from exceeding the P-T limits of 10 CFR 50, Appendix G. Using the NRC-approved methodology provided in WCAP-14040-A, Rev. 2, "*Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*", the analysis determines the LTOP system setpoints for Seabrook Station. At the lowest RCS temperature of 60°F (bolt-up temperature), the corresponding Appendix G limit is 621 psig. The overpressure protection system operates below a temperature of 290°F and it relies on a combination of residual heat removal (RHR) system relief valves and programmable PORVs to prevent the system from reaching a high pressure at low temperatures under the most severe design transients including mass injection and heat injection. Included in these setpoints are margins to accommodate overshoot and instrument uncertainty. The LTOP system pressure versus temperature settings are shown in Table 4.2.4-2.

**Table 4.2.4-2 LTOP System Setpoints for Seabrook Station**

| Setpoint | | Comments |
|---|---|---|
| **Temperature (°F)** | **Pressure (psig)** | |
| 71 | 561 | Overpressure protection in this range provided by the RHR relief valve |
| 100 | 561 | |
| 122 | 561 | |
| 141 | 561 | |
| 161 | 561 | |
| 187 | 561 | |
| 202 | 950 | |
| 209 | 1035 | Overpressure protection in this range provided by programmable PORVs |
| 215 | 1108 | |
| 228 | 1265 | |
| 237 | 1446 | |
| 249 | 1723 | |
| 256 | 1897 | |
| 258 | 1966 | |
| 269 | 2345 | |
| 283 | 2500 | |
| 290 | 2500 | |

### Disposition

**Aging Management, 10 CFR 54.21(c)(1)(iii)** – The provisions of 10 CFR 50, Appendix G, require Seabrook to operate within the currently licensed P-T limit curves. These curves are required to be maintained and updated as necessary to maintain plant operation consistent with 10 CFR 50. The Reactor Vessel Integrity Surveillance Program maintains the P-T limit curves for the period of extended operation. Therefore, the P-T limit curves TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii). Prior to the period of extended operation, updated P-T limit calculations will be prepared for the Seabrook Station reactor vessel beltline region materials, inlet and outlet nozzles, and closure head flange locations for normal heatup, normal cooldown, and in-service leak and hydrostatic test conditions. The Reactor Vessel Integrity Surveillance Program, B.2.1.19 monitors reactor vessel embrittlement. This program provides data to update the P-T limits and, therefore, permits Seabrook Station to manage the P-T limits going forward in accordance with 10 CFR 54(c)(1)(iii). Seabrook Station will submit updates to the P-T curves and LTOP limits to the NRC at the appropriate time to comply with 10 CFR 50 Appendix G.

## 4.3 METAL FATIGUE ANALYSIS OF PIPING AND COMPONENTS

Metal fatigue was evaluated in the design process for Seabrook Station pressure boundary components, including the reactor vessel, reactor coolant pumps, steam generators, pressurizer, piping, valves, and components of primary, secondary, auxiliary, steam, and other systems. The current design analyses for these components have been determined to be Time-Limited Aging Analyses (TLAAs) requiring evaluation for the period of extended operation. Fatigue TLAAs for Seabrook Station pressure boundary components are characterized by determining the applicable design code and design specifications that specify the fatigue design requirements. These design codes are listed in Table 4.3.1-1. NUREG-1801 provides a listing of components that are likely to have TLAAs in place that require evaluation for License Renewal. Each of these has been reviewed and the applicable TLAAs are evaluated in the following sections, as appropriate.

This section is divided into seven subsections that each addresses a specific grouping of components that were analyzed in accordance with the same design requirements.

These Sections are as follows:

- Nuclear Steam Supply System (NSSS) Pressure Vessel And Component Fatigue Analyses (4.3.1)
- Supplementary ASME Section III, Class 1 Piping And Component Fatigue Analyses (4.3.2)
- Reactor Vessel Internals Aging Management (4.3.3)
- Environmentally-Assisted Fatigue Analyses (4.3.4)
- Steam Generator Tube, Loss Of Material And Fatigue Usage From Flow-Induced Vibration (4.3.5)
- Absence Of TLAAs For Fatigue Crack Growth, Fracture Mechanics Stability, Or Corrosion Analyses Supporting Repair Of Alloy 600 Materials (4.3.6)
- Non-Class 1 Component Fatigue Analyses (4.3.7)

The evaluations referenced in these sections fall into the following categories:

- Explicit fatigue analyses for NSSS pressure vessels and components prepared in accordance with ASME Section III, Class A or Class 1 rules developed as part of the original design.

- Supplemental explicit fatigue analyses for piping and components that were prepared in accordance with ASME Section III rules to evaluate transients that were identified after the original design analyses were completed, such as pressurizer surge line thermal stratification, and reactor vessel internal component fatigue analyses.

- New fatigue analyses were prepared for license renewal to evaluate the effects of the reactor water environment on the sample of high-fatigue locations applicable to newer vintage Westinghouse Plants, as identified in Section 5.5 of NUREG/CR-6260. The environmental fatigue methodology and results of these analyses are presented in Section 4.3.4. New explicit analyses were prepared in accordance with ASME Section III, Class 1 rules for each of these components. For these locations environmental fatigue correction factors were computed and applied to the Cumulative Usage Factor (CUF) values developed in the Class 1 fatigue analyses.

## 4.3.1 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) PRESSURE VESSEL AND COMPONENT FATIGUE ANALYSES

### Summary Description

Nuclear Steam Supply System (NSSS) pressure vessels and primary components for Seabrook Station were designed in accordance with ASME Section III, Class 1 requirements and are required to have explicit analyses of cumulative fatigue usage. Table 4.3.1-1 identifies the applicable design codes for these components.

**Table 4.3.1-1**
**Original Design Codes for NSSS Components at Seabrook Station**

| Component | Codes | Edition/Addendum |
|---|---|---|
| Reactor Vessel | ASME Section III, Class 1 | 1971 with Addenda through Summer 1972 |
| Reactor Vessel Closure Head | ASME Section III, Class 1 | 1971 with Addenda through Summer 1972 |
| Pressurizer | ASME Section III, Class 1 | 1971 with Addenda through Summer 1972 |
| Steam Generators | ASME Section III, Class 1 | 1971 with Addenda through Summer 1972 |
| Reactor Coolant Pump Casings | ASME Section III, Class 1 | 1971 with Addenda through Summer 1972 |

ASME Section III, Class 1 fatigue analyses determine the CUF value that results from the component being exposed to the list of postulated transients during the expected life of the component (Table 4.3.1-2). This analysis is performed using the appropriate S-N (Stress amplitude / Number of cycles) fatigue curve from ASME Section III for the component material type. The curve shows the number of cycles the component can withstand without cracking for given amplitude of applied alternating stress. The resulting CUF of less than 1.0 indicates the cumulative effects from the postulated transients will not result in the initiation of fatigue cracking.

These ASME Section III, Class 1 fatigue analyses are based upon explicit numbers and amplitudes of thermal and pressure transients described in the design specifications. The intent of the design basis transient definitions is to bound not just specific operations but a wide range of possible events with varying ranges of severity in temperature, pressure, and flow. The most limiting numbers of transients used in these NSSS component analyses are shown in Table 4.3.1-2, and are considered to be design limits.

**Table 4.3.1-2**
**Summary of Reactor Coolant System Design Transients**

| Transient Description | Limiting Design Basis Number of Occurrences for 40 Year Operating Period |
|---|---|
| **Normal Condition Transients:** | |
| Plant Heatup @ ≤ 100 °F/hr | 200 |
| Plant Cooldown @ ≤ 100 °F/hr | 200 |
| Unit Loading @ 5% full power/min | 13,200 (1) |
| Unit Unloading @ 5% full power/min | 13,200 (1) |
| Step Load Increase of 10% of full power | 2,000 |
| Step Load Decrease of 10% of full power | 2,000 |
| Large step load decrease with steam dump | 200 |
| Steady state fluctuations | Initial – $1.5 \times 10^5$<br>Random – $3.0 \times 10^5$ |
| Feedwater Cycling at Hot Shutdown | 2,000 |
| Loop out of service<br>Normal loop shutdown<br>Normal loop startup | <br>80<br>70 |
| Feedwater Heaters out of service<br>One heater out of service<br>One bank of heaters out of service | <br>120<br>120 |
| Unit loading between 0% to 15% of full power | 500 (2) |
| Unit unloading between 0% to 15% of full power | 500 (2) |
| Boron concentration equalization | 26,400 |
| Refueling | 80 |
| Reduced temperature return to power | 2,000 |
| Reactor Coolant Pumps startup/shutdown | 3,000 (3) |
| **Upset Transients:** | |
| Loss of load without immediate turbine trip | 80 |
| Loss of all offsite power (blackout with natural circulation in the RCS) | 40 |
| Partial loss of flow (loss of one pump) | 80 |

**Table 4.3.1-2**
**Summary of Reactor Coolant System Design Transients**

| Transient Description | Limiting Design Basis Number of Occurrences for 40 Year Operating Period |
|---|---|
| Reactor trip from full power: | |
| *Without cooldown* | 230 |
| *With cooldown, without safety injection* | 160 |
| *With cooldown and safety injection* | 10 |
| Inadvertent reactor coolant depressurization | 20 |
| Inadvertent startup of inactive loop | 10 |
| Control rod drop | 80 |
| Inadvertent ECCS actuation | 60 |
| Operating Basis Earthquake (5 earthquakes of 10 cycles each) | 50 |
| Excessive feedwater flow | 30 |
| RCS Cold Overpressurization | 10 |
| **Emergency Transients:** | |
| Small LOCA | 5 |
| Small steam break | 5 |
| Complete loss of flow | 5 |
| **Faulted Transients:** | |
| Main reactor coolant pipe break (LOCA) | 1 |
| Large steam line break | 1 |
| Feedwater line break | 1 |
| Reactor Coolant Pump locked rotor | 1 |
| Control rod ejection | 1 |
| Steam Generator tube rupture | Included under Reactor Trip with cooldown and safety injection |
| Safe Shutdown Earthquake | 1 |
| **Test Transients:** | |
| Primary side hydrostatic test | 10 |
| Secondary side hydrostatic test | 10 |
| Turbine roll test | 20 |
| Primary side leak test | 200 |
| Secondary side leak test | 80 |
| Tube leak test | 800 |

1. For the design transient of Unit Loading and Unit Unloading @ 5% full power/min., the Reactor Vessel, Steam Generators and Pressurizers are designed for 13,200 cycles, where the Class 1 piping is

designed for 18,300 cycles. The most limiting value of these major components is used as a monitoring limit in the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1.

2. For the design transients of Unit load and unload between 0% to 15% of full power, the Reactor Vessel, Steam Generators and Class 1 piping are designed for 500 cycles, where the Pressurizer is designed for 1,510 cycles. The most limiting value of these major components is used as a monitoring limit in the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1.

3. For the design transient of Reactor Coolant Pump startup/shutdown, the limit specified in the UFSAR is 3800 cycles. The Pressurizer is designed for 4,000 cycles, where Steam Generators are designed for 3,000 cycles. The Steam Generators has the most limiting value (3,000 cycles) of these components is used as a monitoring limit in the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1.

Each Seabrook component designed in accordance with ASME Section III, Class 1 rules was analyzed and shown to have a CUF less than the design limit of 1.0. Since each Class 1 fatigue analysis is based upon a number of cycles postulated to bound 40 years of service, they have been identified as TLAAs that require evaluation for 60 years.

## Analysis

In order to determine if the ASME Section III, Class 1 fatigue analyses will remain valid for 60 years of service, a review of fatigue monitoring data was performed to determine the number of cumulative cycles for each transient type that have occurred during past plant operations. Then, the average rate of occurrence was determined, and predictions of future transient occurrences were made. For each transient type, the 60-year projected number of occurrences was determined by adding the number of past occurrences to the number of predicted future occurrences. These 60-year projections were then compared to the number of design cycles used in the fatigue analyses to determine if the design cycles remain bounding for 60 years of operations. If the 60-year projected numbers of cycles is less than the number of cycles used in the design fatigue analyses, then the fatigue analyses based upon the design transients will remain valid for 60 years of operation; if the design transient severity is also bounding of the actual transient severity.

Therefore, an evaluation was performed to determine if the severity of the actual plant transients that have occurred during past operations remains bounded by the transient severity provided for each transient definition in the design specification. This evaluation was to assure that the past cycles were appropriately characterized during fatigue monitoring activities in the past. The administrative and operating procedures were also reviewed in order to assess the effectiveness of the design transient cycle counting program and to validate the cyclic assumptions. This evaluation determined that the actual transient severity was bounded by the design transient severity for each transient type. The cycle counting procedure was also determined to have been effective in properly characterizing actual plant transients.

The overall conclusion of these evaluations is that the existing design transients bound transients projected for 60 years of plant operations.

**60-Year Transient Projection Methodology**

***Projection Methodology***

For Seabrook Station, the baseline period started on August 19, 1990 and ended on April 1, 2009, a total of 18.6 calendar years. For each transient type, the average rate of occurrence was determined by dividing the cumulative number of occurrences as of April 1, 2009 by 18.6 years of past operation. For each transient type, future cycles were predicted by multiplying the average rate of past occurrences by the number of calendar years remaining between April 1, 2009 and March 15, 2050. The 60-year projection was determined by adding the cumulative number of occurrences as of April 1, 2009 to the number of cycles predicted to occur in the 41 years of future operation. This methodology is considered to produce a conservative estimate of cycle values due to the declining trend of most of the transients since the beginning of plant operation.

***Testing Events***

One cycle has occurred for the following plant events:

- Turbine Roll Test
- Primary Side Leakage Test
- RCS Hydrostatic Pressure Test
- Secondary Side Leakage Test
- Tube Leakage Test
- Secondary Side Hydrostatic Pressure Test

These cycles occur before initial startup or during component installation and then typically do not occur again in the plant's lifetime. Therefore, the 60-year cycle projection for these events was taken as the current cycle count of one.

***Events with Zero Cycles***

No cycles have occurred for the following plant events:

- Operating Basis Earthquake
- RCS Loop Out of Service

- Inadvertent RCS Depressurization
- Inadvertent startup of inactive loop
- Excessive Feedwater Flow
- Reactor Trip with cooldown and safety injection
- Partial Loss of Flow
- Inadvertent Reactor Coolant Pump Startup
- RCS Cold Overpressurization

These events occur infrequently over the lifetime of a plant. Therefore, one cycle was predicted for the 60-year cycle projection for each of these plant events. The only exception is RCS Loop Out of Service. It is assumed to have zero cycles occur for the 60-year cycle projection, because the plant is not licensed to operate in this state.

***Relatively Frequent Events***

The following plant events occur frequently over the course of the lifetime of a plant:

- Feedwater Cycling
- RCP Startup and RCP Shutdown

The 60-year cycle projection for the RCP Startup and Shutdown cycles was linearly extrapolated based on the rate of accumulation for the data available through April 1, 2009 and additional 10% of cycles were included to accommodate any slight increase in occurrence over the years up to 60 years. The Feedwater Cycling events were not counted in the plant records and were prorated annually on the basis of the design number of cycles as the design number provides a conservative estimate for the non-monitored events considered.

***Relatively Infrequent Events***

The following plant events occur infrequently over the course of the lifetime of a plant:

- Step Load Increase ≤ 10%
- Step Load Decrease ≤ 10%

- Large Step Load Decrease with Steam Dump
- Unit Loading Between 0% and 15% Power
- Unit Unloading Between 0% and 15% Power
- Loss of Load without Immediate Reactor Trip
- Loss of Power
- Reactor Trip with cooldown but not safety injection
- Control Rod Drop
- Inadvertent Safety Injection (SI) Actuation
- Auxiliary Spray Actuation

The 60-year cycle projection for these events was linearly extrapolated based on the rate of accumulation for the data available through 4/1/2009.

***Fatigue-Insignificant Events***

Steady State Fluctuation and Boron Concentration Equalization occur frequently over the course of the lifetime of a plant, but have an insignificant effect on fatigue usage for any Class 1 component. No 60-year cycle projection is made for these events and they are not included Table 4.3.1-3.

| Table 4.3.1-3<br>60-Year Design Transient Projections for<br>NSSS Class 1 Components at Seabrook Station | | | |
|---|---|---|---|
| Transient | Current Cycles (through 4/1/2009 – 18.6 Years of Operation) | 60-Year Projected Cycles | NSSS Design Cycles |
| **Normal Condition Transients:** | | | |
| Plant Heatup ≤ 100°F/hr | 27 | 87 | 200 |
| Plant Cooldown ≤ 100°F/hr | 26 | 84 | 200 |
| Pressurizer Cooldown ≤ 200°F/hr | 36 | 116 | 200 |
| Unit Loading @ 5%/min | 104 | 334 | 13,200 |
| Unit Unloading @ 5%/min | 80 | 257 | 13,200 |
| Step Load Increase of 10% of full Power | 1 | 4 | 2,000 |
| Step Load Decrease of 10% of full Power | 3 | 10 | 2,000 |
| Large Step Load Decrease (50%) with Steam Dump | 4 | 13 | 200 |
| Unit Loading Between 0% and 15% Power | 27 | 13 | 500 |
| Unit Unloading Between 0% and 15% Power | 26 | 10 | 500 |
| RCP Startup | 152 | 536 | 3,000 (4) |
| RCP Shutdown | 152 | 536 | 3,000 (4) |
| Feedwater Cycling at Hot Shutdown | 620 (1) | 2,000 | 2,000 (5) |
| Loop out of service | 0 (2) | 1 (2) | 80 |
| Feedwater Heaters out of service | 12 | 39 | 120 (5) |
| Refueling | 12 | 39 | 80 |
| Reduced Temperature Return to Power | 0 | 1 | 2,000 |
| **Upset Condition Transients:** | | | |
| Loss of Load without Immediate Reactor Trip | 2 | 7 | 80 |
| Loss of Power | 2 | 7 | 40 |
| Partial Loss of Flow | 0 | 1 | 80 |
| **Reactor Trip from Full Power** | | | |
| Reactor Trip – no cooldown | 30 | 97 | 230 |
| Reactor Trip - With cooldown, without safety injection | 2 | 7 | 160 |
| Reactor Trip - With cooldown, | 0 | 1 | 10 |

**Table 4.3.1-3**
**60-Year Design Transient Projections for NSSS Class 1 Components at Seabrook Station**

| Transient | Current Cycles (through 4/1/2009 – 18.6 Years of Operation) | 60-Year Projected Cycles | NSSS Design Cycles |
|---|---|---|---|
| and safety injection | | | |
| Inadvertent Auxiliary Spray to Pressurizer | 2 | 7 | 10 |
| Inadvertent RCS Depressurization | 0 | 1 | 20 |
| Inadvertent startup of inactive loop | 0 | 1 | 10 |
| Control Rod Drop | 1 | 4 | 80 |
| Inadvertent Safety Injection Actuation (SI) | 1 | 3 | 60 |
| Operating Basis Earthquake | 0 | 10 (3) | 50 |
| Excessive Feedwater Flow | 0 | 1 | 30 |
| **Test Condition Transients:** | | | |
| Turbine Roll Test | 1 | 1 | 20 |
| Primary Side Hydrostatic Test | 1 | 1 | 10 |
| Secondary Side Hydrostatic Test | 1 | 1 | 10 |
| Primary Side Leak Test | 1 | 1 | 200 |
| Secondary Side Leak Test | 1 | 1 | 80 |
| Tube Leak Test | 1 | 1 | 800 |

(1) Prorated on the basis of the design number of events occurring in 60 years.

(2) Zero cycles assumed because the event is not allowed by procedure. One cycle is projected to occur during 60 years

(3) One earthquake with 10 cycles

(4) For the design transient of Reactor Coolant Pump startup/shutdown, the limit specified in the UFSAR is 3800 cycles. The Pressurizer is designed for 4,000 cycles, where Steam Generators are designed for 3,000 cycles. The Steam Generators has the most limiting value (3,000 cycles) of these components is used as a monitoring limit in the Metal Fatigue of Reactor Coolant Pressure Boundary Program (B.2.3.1).

(5) The plant does not monitor these events. The original design analysis number is assumed to be the anticipated number of cycles at the end of the period of extended operation. Cycles shown in the current cycles column (column 2) represent a review of plant records to validate assumption.

## Disposition

**Validation, 10 CFR 54.21(c)(1)(i)** – The 40-year design transients bound the numbers of cycles projected to occur during 60 years of plant operations at Seabrook Station. Therefore, the NSSS Class 1 fatigue analyses that are based upon the 40-year design transients remain valid for the period of extended operation.

## 4.3.2 SUPPLEMENTARY ASME SECTION III, CLASS 1 PIPING AND COMPONENT FATIGUE ANALYSES

### Summary Description

In addition to the original design assumptions, the Seabrook Station Pressurizer fatigue evaluations were updated to include the added thermal stratification effects of insurge and outsurge events on the pressurizer lower head and surge nozzle. These new conditions did not change the number of original design cycles, but rather incorporated the effects as new sub-events within the original design transients.

Each of the Seabrook Station piping systems, including the Reactor Coolant System main loop piping, were originally designed in accordance with ASME Section III 1971 Edition with addenda through Winter 1972. Since then, a number of updated fatigue analyses have been prepared for piping systems and components to address transients that have been identified in the industry that were not originally considered. These analyses have been performed in accordance with ASME Section III, Class 1 rules to enable these transients to be thoroughly evaluated. These transients include those associated with potential valve leakage transients identified in NRC Bulletin 88-08 for the auxiliary spray line, charging lines, safety injection lines, and thermal stratification of the pressurizer surge line, as described in NRC Bulletin 88-11.

These analyses are separated from those evaluated in the previous sections because the transient definitions have been modified, or additional transients have been postulated for these components, in addition to those previously described. Therefore, the cycle projections for these components must address these revised transients or additional transient types to determine if they also remain bounded for 60 years of service. Each of these analyses is dispositioned separately within this section for clarity.

### 4.3.2.1 Absence of a TLAA for Thermal Stresses in Piping Connected to Reactor Coolant Systems: NRC Bulletin 88-08

### Summary Description

NRC Bulletin 88-08 *"Thermal Stresses in Piping Connected to Reactor Cooling Systems"* was issued June 22, 1988, because of observed pipe cracking due to valve leakage in unisolable lines. Three supplements were issued on June 24, 1988, August 4, 1988, and April 11, 1989, respectively. The Bulletin and supplements required licensees to identify potential locations that might be subject to high thermal stresses either from thermal stratification or temperature oscillations due to leaking valves and inspect the potential locations to assure that susceptible locations will not fail for the remaining life of the unit.

Seabrook Station evaluated the effects of thermal stresses due to leaking valves in seven piping sections that are unisolable from the RCS that were pressurized by the charging pumps and could potentially experience in-leakage to the RCS from leaking valves. Prior to initial criticality, a one-time non-destructive examination was performed for the four high head safety injection lines with acceptable results. A temperature monitoring program was deployed for potentially susceptible lines. The monitoring is still installed and credited in the Seabrook Station Management of Thermal Fatigue Program for Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146/146S).

The NRC approved Seabrook Station's response to NRC Bulletin 88-08.

**Analysis**

NRC Bulletin 88-08 and Supplements 1 and 2 addressed the evaluation of thermal stresses in unisolable piping from cold fluid in-leakage to the RCS through leaking valves. NRC Bulletin 88-08 Supplement 3 addressed the evaluation of thermal stresses in unisolable piping from hot fluid out-leakage from the RCS through leaking valves.

Seabrook Station evaluated the possibility and effects of fluid in-leakage by identifying seven piping sections that are unisolable from the RCS and pressurized by the charging pumps where the effects of thermal stresses due to leaking valves could potentially experience in-leakage to the RCS from leaking valves. Four of these lines are the high head safety injection lines and three of these lines are charging system lines (the normal and alternate charging loop charging lines and the pressurizer auxiliary spray line).

In 1988, prior to initial criticality, a one-time non-destructive examination was performed for the four high head safety injection lines, which showed acceptable results. Non-destructive examinations were not considered to be necessary for the three charging system lines because they had not yet been subjected to excessive thermal cycling at that time.

Seabrook Station deployed a temperature monitoring program for the high head safety injection lines and three charging system lines in 1989, prior to initial criticality. This program installed temperature detectors on the unisolable piping sections to detect adverse temperature distributions, and established appropriate temperature limits, requirements for periodic review of the temperature instrument values and action limits in the event of exceeding the temperature limit.

Seabrook Station evaluated the possibility and effects of fluid out-leakage and concluded that unisolable piping sections connected to the Reactor Coolant System at Seabrook Station are not subject to stresses from thermal stratification or temperature oscillations resulting from the mechanism described in NRC Bulletin Supplement 3. There is no specific TLAA for these seven locations addressed by NRC Bulletin 88-08 and related supplements.

**Conclusion**

Unisolable piping sections connected to the Reactor Coolant System at Seabrook Station are not subject to stresses from thermal stratification or temperature oscillations resulting from the mechanism described in NRC Bulletin 88-08 and related supplements. There is no specific TLAA for locations addressed by NRC Bulletin 88-08 and related supplements.

#### 4.3.2.2 NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification

**Summary Description**

NRC Bulletin 88-11, issued on December 20, 1988, requested utilities to establish and implement a program to confirm the integrity of the pressurizer surge line. The program required both visual inspection of the surge line and demonstration that the design requirements of the surge line are satisfied, including the consideration of stratification effects.

The Pressurizer Surge Line piping and nozzles were previously evaluated for the effects of thermal stratification and plant-specific transients (1990) and it was determined that the surge line will remain within the ASME Code requirements for the design life of the unit. The controlling fatigue location was the hot leg surge line nozzle safe-end. In later evaluations, plant-specific ASME Section III, Class 1 evaluations were performed for the hot leg surge line nozzle and pressurizer surge nozzle.

***Hot Leg Surge Line Nozzle***

The hot leg surge line nozzle was evaluated for the effects of pressurizer insurge and outsurge transients and surge line stratification. This evaluation was part of an evaluation of reactor water environmental effects on the surge line. The model from that analysis was also used to evaluate the design cycles for the NSSS transients and projected 60-year cycles of surge line stratification and insurge and outsurge transients, to calculate CUF at the hot leg surge line nozzle, without environmentally-assisted fatigue effects. Since the analysis envelopes the 60-year cycles, it remains valid for 60 years. The evaluation incorporating these models in the evaluation of environmentally-assisted fatigue is described further in section 4.3.4 Environmentally-Assisted Fatigue Analyses

*Pressurizer Surge Nozzle*

The pressurizer surge nozzle was evaluated for the effects of pressurizer insurge and outsurge transients and surge line stratification. This evaluation was part of an evaluation of the structural weld overlay applied to the pressurizer surge nozzle. The model from that analysis was also used to evaluate the design cycles for the NSSS transients and design cycles of surge line stratification and insurge and outsurge transients, to calculate CUF at the pressurizer surge nozzle. Since the analysis envelopes the 60-year cycles, it remains valid for 60 years.

## Analysis

*Hot Leg Surge Line Nozzle*

The hot leg surge line nozzle has been evaluated using an ASME Section III, Class 1 fatigue analysis. This analysis was part of the evaluation of the environmental effects of reactor coolant (Section 4.3.4). In addition, that analysis was used to evaluate CUF without environmental effects. The analysis performed to demonstrate compliance with design requirements considered ASME Code requirements and utilized the design set of NSSS transients. Pressurizer surge line stratification sub-transients were developed based on NSSS-vendor Seabrook Station-specific evaluations for pre-MOP (Modified Operating Procedure) plant operating procedures and NSSS-vendor evaluations of surge line monitoring data from similar units and historical records for Seabrook Station for post-MOP operating procedures. Projected 60-year cycles of surge line stratification and insurge and outsurge transients were used when these were greater than previously evaluated design cycles. These evaluations resulted in CUF less than 1.0 at the hot leg surge line nozzle.

*Pressurizer Surge Nozzle*

The pressurizer surge nozzle has been evaluated using an ASME Section III, Class 1 fatigue analysis. This analysis was part of the evaluation of the structural weld overlay applied to the pressurizer surge nozzle. The analysis performed to demonstrate compliance with design requirements considered ASME Code requirements and utilized the design set of NSSS transients and surge line stratification sub-transients were developed based on NSSS-vendor Seabrook Station specific evaluations for pre-MOP operating plant operating procedures and NSSS-vendor evaluations of surge line monitoring data from similar units and historical records for Seabrook for post-MOP operating procedures. This evaluation included an elastic-plastic formulation and resulted in CUF less than 1.0 at the pressurizer surge nozzle.

### Disposition

**Validation**, 10 CFR 54.21(c)(1)(i) – The analyses remain valid for the period of extended operation for the Pressurizer Surge Line, Pressurizer Surge Nozzle and Hot Leg Surge Line Nozzle

## 4.3.3 REACTOR VESSEL INTERNALS AGING MANAGEMENT

### Summary Description

The Seabrook Station Reactor Vessel Internals were designed and constructed prior to the development of ASME Code requirements for core support structures, but the reactor coolant system functional design requirements were considered in the design. The Reactor Vessel Internals were further analyzed for fatigue as part of the Seabrook Station power uprate and determined that cumulative usage factors would remain less than 1.0.

Demonstration that the effects of aging are adequately managed is essential for assuring continued functionality of the reactor internals during the desired plant operating period, including license renewal. The recently-published EPRI Materials Reliability Program (MRP) Reactor Internals Inspection & Evaluation (I&E) Guidelines, MRP-227, are intended to support that demonstration, with requirements for inspection to detect the effects of aging degradation.

### Analysis

The mechanisms of aging of PWR internals are described below:

***Stress Corrosion Cracking***

Stress Corrosion Cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

***Irradiation-Assisted Stress Corrosion Cracking***

Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly-irradiated components. The aging effect is cracking.

### *Wear*

Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

### *Fatigue*

Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive. Fatigue crack initiation and growth resistance is governed by a number of material, structural and environmental factors, such as stress range, loading frequency, surface condition and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations, such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

### *Thermal Aging Embrittlement*

Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardenable stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and precipitation-hardenable stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

***Irradiation Embrittlement***

Irradiation embrittlement is also referred to as neutron embrittlement. When exposed to high energy neutrons, the mechanical properties of stainless steel and nickel-base alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

***Void Swelling and Irradiation Growth***

Void swelling is a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5% by volume) has been correlated with extremely low fracture toughness values. Also included in this description is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes in in-core instrumentation tubes, fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

***Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep***

The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or, primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, such as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 100 hours) at PWR internals temperatures.

Creep (or more precisely, secondary creep) is a slow, time and temperature dependent, plastic deformation of materials that can occur when subjected to stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress; and, it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading which, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

### Disposition

**Aging Management, 10 CFR 54.21(c)(1)(iii)** The PWR Vessel Internals Program, B.2.1.7 will manage the aging effects including changes in dimensions, cracking, loss of fracture toughness, and loss of preload of the Reactor Vessel Internals components for the period of extended operation per 10 CFR 54.21(c)(1)(iii).

## 4.3.4 ENVIRONMENTALLY-ASSISTED FATIGUE ANALYSES

### Summary Description

Environmentally-assisted fatigue analyses do not meet the definition of time-limited aging analyses under 10 CFR 54.3 because they are not contained or incorporated by reference in Seabrook Station's current licensing basis. This subsection is included in response to the request in NUREG-1800 that license renewal applicants address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. This discussion of environmentally-assisted fatigue calculations is provided, in accordance with NUREG-1800, to assist with the formulation of aging management programs.

NUREG-1801, Revision 1, Generic Aging Lessons Learned, contains recommendations on specific areas for which existing programs should be augmented for license renewal. The program description for Aging Management Program X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary Program, provides guidance for addressing environmental fatigue for license renewal. It states that an acceptable program addresses the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant. Examples of these components are identified in NUREG/CR-6260, "*Application of* NUREG/CR-5999 *Interim Fatigue Curves to Selected Nuclear Power Plant Components*".

This sample of components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses using formulae contained in NUREG/CR-6583, "*Effects of LWR Coolant Environments on*

*Fatigue Design Curves of Carbon and Low Alloy Steels*" and in NUREG/CR-5704, "*Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels*". Demonstrating that these components have an environmentally adjusted cumulative usage factor less than or equal to the design limit of 1.0 for the 60-year period of extended operation is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary. Another acceptable option is to manage fatigue of these components in the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1 by tracking the number and severity of plant transients to compare to the established trigger number for each plant transient. Under the B.2.3.1 option, when the counted number of a specific plant transient reaches the trigger number associated with that specific plant transient, a preemptive remedial action will be undertaken appropriate to the plant transient and the components affected by it. This remedial action may encompass one of several activities, as described in B.2.3.1:

1. Reanalyze affected components for an increase in the number of that specific transient while accounting for other component-affecting plant transients that may be projected not to achieve their analyzed levels.

2. Perform a fracture mechanics evaluation of a postulated flaw in affected plant components, which when coupled with an inservice inspection program, will serve to demonstrate flaw tolerant behavior.

3. Repair the affected component.

4. Replace the affected component.

NUREG/CR-6260 provides environmental fatigue calculations for a newer vintage Westinghouse plant (like Seabrook Station) using the interim fatigue curves from NUREG/CR-5999 for the locations of highest design CUF for the components listed below:

- Reactor Vessel Shell and Lower Head
- Reactor Vessel Inlet and Outlet Nozzles
- RCS Pressurizer Surge Line
- RCS Charging Nozzle
- RCS Safety Injection Nozzle
- RCS Residual Heat Removal System Class 1 Piping

## Analysis

For the NUREG/CR-6260 locations identified above, the plant-specific components were identified and the design ASME fatigue usage factors were adjusted by the environmentally-assisted fatigue penalty factors ($F_{en}$) to obtain the environmentally-assisted fatigue (EAF) results for the RV Inlet and Outlet Nozzles, RV Shell and Lower Head and RHR Hot Leg Nozzle.

Table 4.3.4-1 summarizes the locations where ASME 60-Year air-curve and EAF results were calculated as well as the results. All locations were shown to achieve air-curve cumulative usage factors less than 1.0 for the 60 years of service. The evaluations show that 60-year EAFs exceed 1.0 for 60 years of service for the hot leg surge line nozzle and charging nozzle.

In the EAF evaluations for Hot Leg Surge Nozzle, Charging Nozzle and Safety Injection Nozzle, $F_{en}$ factors for each transient pair were calculated using the integrated strain rate method from MRP-47, Rev. 1.

The Effective $F_{en}$ Multipliers shown in Table 4.3.4-1 for Hot Leg Surge Nozzle, Charging Nozzle and Safety Injection Nozzle, were computed as (Environmentally-Assisted Fatigue CUF) / (Air-Curve CUF).

**Table 4.3.4-1**

**60 Year Air-Curve and Environmentally-Assisted Fatigue Results**

| Component | 60 Year ASME Air-Curve CUF | Effective $F_{en}$ Multiplier | 60-Year EAF-adjusted CUF | Material Type [(6)] | DO [(7)] | T [(8)] |
|---|---|---|---|---|---|---|
| **1. Reactor Vessel Shell and Lower Head** [(1)] | | | | | | |
| Inside surface of lower head near shell-to-head junction [(2)(4)] | 0.007 | 2.455[(9)] | 0.0172 | LAS | < 50 ppb | > 200°C |
| **2. Reactor Vessel Inlet and Outlet Nozzles** [(1)] | | | | | | |
| Reactor vessel inlet nozzle [(2)(4)] | 0.0795 | 2.455[(9)] | 0.195 | LAS | < 50 ppb | > 200°C |
| Reactor vessel outlet nozzle [(2)(4)] | 0.1077 | 2.455[(9)] | 0.264 | LAS | < 50 ppb | > 200°C |
| **3. Pressurizer Surge Line** [(1)] | | | | | | |
| Hot leg surge nozzle-to-pipe weld [(2)(3)(5)] | 0.2844 | 12.05 | 3.428 | SS | < 50 ppb | > 200°C |
| **4. Charging Nozzle** [(1)] | | | | | | |
| Charging nozzle near blend radius [(2)(3)(5)] | 0.9671 | 5.66 | 5.471 | SS | < 50 ppb | > 200°C |
| **5. Safety Injection Nozzle** [(1)] | | | | | | |
| BIT nozzle near blend radius [(2)(3)(5)] | 0.112 | 3.49 | 0.390 | SS | < 50 ppb | > 200°C |
| **6. RHR System Class 1 Piping** [(1)] | | | | | | |
| RHR hot leg nozzle-to-pipe weld [(2)(4)] | 0.0407 | 15.35[(10)] | 0.625 | SS | < 50 ppb | > 200°C |

(1) NUREG/CR-6260 Component Location for a Newer Vintage Westinghouse Plant

(2) Plant-specific limiting location within the boundary of the applicable NUREG/CR-6260 Component Location

(3) Analysis performed using 60-year projected cycles

(4) Analysis performed using design number of design-severity cycles

(5) The Effective $F_{en}$ Multiplier is computed to be (Environmentally-Assisted Fatigue CUF) / (Air-Curve CUF)

(6) LAS = Low Alloy Steel; SS = Stainless Steel

(7) DO – dissolved oxygen for EAF computation

(8) T – maximum service temperature for EAF computation

(9) The Effective $F_{en}$ Multiplier is computed based on the maximum value from NUREG/CR-6583

(10) The Effective $F_{en}$ Multiplier is computed based on the maximum value from NUREG/CR-5704

## Disposition

**Revision 10 CFR 54.21(c)(1)(ii)** The evaluation of environmental fatigue effects for the Reactor Vessel Shell and Lower Head and Reactor Vessel Inlet and Outlet Nozzles determined that the CUF will remain below the ASME code allowable fatigue limit of 1.0 using the maximum applicable $F_{en}$, applied to CUF based on the design number of transients for these locations, when extended to 60 years. The evaluation of fatigue effects for these locations has thereby been validated for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i), including effects of the reactor coolant environment. Therefore, no aging management program is necessary to address environmentally-assisted fatigue for these components.

**Aging Management 10 CFR 54.21(c)(1)(iii)** The remainder of these locations, RCS Pressurizer Surge Line Nozzle, RCS Charging Nozzle, RCS Safety Injection Nozzle, and RCS Residual Heat Removal System Class 1 Piping, were analyzed in accordance with ASME Code Section III, Subarticle NB-3200 using all six stress components. These analyses were based on Seabrook Station Specific conditions and these locations will be monitored for fatigue usage including environmental effects by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1. Specifically, this program will monitor critical transients to verify cycle limits are maintained below limits specified in the UFSAR. Pre-established action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor, including environmental effects, exceeds the AMSE Code limit of 1.0.

At least 2 years prior to entering the period of extended operation, Seabrook Station will implement the following aging management program for the plant-specific locations listed in NUREG/CR-6260 for the newer vintage Westinghouse plants.

(1) Consistent with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1 Seabrook Station will update the fatigue usage calculations using refined fatigue analyses, if necessary, to determine acceptable CUFs (i.e., less than 1.0) when accounting for the effects of the reactor water environment. This includes applying the appropriate $F_{en}$ factors to valid CUFs determined from an existing fatigue analysis valid for the period of extended operation or from an analysis using an NRC-approved version of the ASME Code or NRC-approved alternative (e.g., NRC-approved code case). Formulas for calculating the environmental life correction factors for carbon and low alloy steels are contained in NUREG/CR-6583 and those for austenitic stainless steels are contained in NUREG/CR-5704. NUREG/CR-6909 includes alternate formulas for calculating environmental life correction factors, in addition to updated fatigue design curves.

(2) If acceptable CUFs cannot be demonstrated for all the selected locations, then additional plant-specific locations will be evaluated. For the additional plant-specific locations, if CUF, including environmental effects is greater than 1.0, then Corrective Actions will be initiated, in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1. Corrective Actions will include inspection, repair, or replacement of the affected locations before exceeding a CUF of 1.0 or the effects of fatigue will be managed by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Therefore, the effects of the reactor coolant environment on fatigue usage factors in the remaining locations will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

### 4.3.5 STEAM GENERATOR TUBE, LOSS OF MATERIAL AND FATIGUE USAGE FROM FLOW-INDUCED VIBRATION

**Summary Description**

The Seabrook Station Model F steam generators were evaluated with respect to flow induced vibration (tube wear and fatigue usage) for the power increases that were implemented as part of the Seabrook Station Power Uprates. The analysis of the effects of steam generator flow-induced vibration on tube wear and fatigue usage assumed 40 years of operation.

**Analysis**

The maximum predicted tube wall wear for a 40-year operating life was 0.0032 inch for the pre-power uprate conditions. As a result of the 56% increase in the tube wear rate as a result of the power uprates, the maximum 40-year tube wall wear is less than 0.0050 inch. The maximum 60-year tube wall wear is 0.0075 inch (~20% through-wall wear). This amount of tube wall wear is less than the limit of acceptability of 40% of wall thickness and is deemed not to significantly affect tube integrity.

The evaluation showed that significant levels of tube vibration will not occur from either the fluidelastic or turbulent mechanisms above those associated with the pre-uprated condition.

Low-cycle fatigue usage for the most limiting tube in the most limiting power-uprated operating condition resulting from the flow-induced vibration tube bending stress is 0.2 ksi. This value is well below the fatigue endurance limit of 20 ksi at 1E+11 cycles, resulting in a computed fatigue usage of 0.0. High-cycle fatigue usage of U-bend tubes was evaluated. One of the prerequisites for high-cycle U-bend fatigue is a dented support condition at the upper plate. Seabrook Station steam generator tube support plates are manufactured from stainless steel therefore there is no potential for the necessary conditions to occur. It was concluded that the support condition leading to a dented support condition necessary for high-cycle fatigue cannot occur in the Model F steam generators.

**Disposition**

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation.

### 4.3.6 ABSENCE OF TLAAS FOR FATIGUE CRACK GROWTH, FRACTURE MECHANICS STABILITY, OR CORROSION ANALYSES SUPPORTING REPAIR OF ALLOY 600 MATERIALS

**Summary Description**

Both Alloy 600 base material and Alloy 82/182 weld material have exhibited susceptibility to primary water stress corrosion cracking (PWSCC). Evaluations of these effects, or analyses in support of repairs to affected locations, can be TLAAs.

**Analysis**

***Pressurizer***

The pressurizer contains Alloy 600 material only as Alloy 82/182 welds attaching the surge, spray, and relief valve nozzles to the safe ends, and the safe ends to the connecting piping. Complete Alloy 690 structural weld overlays were completed on all of these locations during Refueling Outage 12 (Spring 2008). The overlays were supported by fatigue crack growth analyses. These fatigue crack growth analyses were projected for a 60-year life, to the end of the period of extended operation, and are therefore not TLAAs.

No base-metal corrosion analyses exist for the pressurizer, since no half-nozzle or similar repairs have exposed the base metal to reactor coolant.

***Reactor Vessel***

A reactor vessel hot leg nozzle Alloy 600 weld was mitigated through Mechanical Stress Improvement Process (MSIP) repair during Outage 13 (Fall 2009). The MSIP repair was supported by fatigue crack growth analysis. This fatigue crack growth analysis was projected, to the end of the period of extended operation, and is therefore not a TLAA.

There have been no other MSIP, Mechanical Nozzle Seal Assembly (MNSA), half-nozzle, or weld overlay repairs to reactor vessel Alloy 600 nozzle locations. Since there have been no MSIP, MNSA, half-nozzle, or weld overlay repairs to reactor vessel Alloy 600 nozzle locations, no other TLAA exists supporting their installation.

***Steam Generators***

The steam generator channel head drains (a/k/a bowl drains) contain Alloy 600 material. The channel head drains will be inspected periodically until the susceptible material is mitigated by replacing the alloy 600 welds.

**Conclusion**

No TLAAs for Fatigue Crack Growth, Fracture Mechanics Stability, or Corrosion Analyses Supporting Repair of Alloy 600 Materials exist for Seabrook Station. Components containing Alloy 600 material are monitored in accordance with B.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program.

## 4.3.7 NON-CLASS 1 COMPONENT FATIGUE ANALYSES

This section describes fatigue-related TLAAs arising within design analyses of the Non-Class 1 piping and components. These piping and tubing components can be designed in accordance with ASME Section III Class 2 and 3.

**Summary Description**

Non-Class 1 piping designed in accordance with ASME Code Section III Class 2 and 3 Piping is not required to have an analysis of cumulative fatigue usage, but cyclic loading is considered in a simplified manner in the design process. When the non-Class 1 Seabrook Station components were designed, the overall number of thermal and pressure cycles expected during the 40-year lifetime of these components was determined. The total number of cycles expected during 40 years was compared to cycle ranges specified in ASME Section III Class 2 and 3 design codes for consideration of allowable stress reduction. If the total number of cycles exceeded 7,000 cycles, a stress range reduction factor was applied to the allowable stress range for secondary stresses (expansion and displacement) to account for thermal cycling. This method is considered to be an implicit fatigue analysis because it is based upon a total number of cycles projected to occur in 40 years, but no explicit Cumulative Usage Factor (CUF) is computed. Because the overall number of cycles could potentially increase during the period of extended operation, which could potentially result in further reduction of the allowable stress range, these implicit fatigue analyses are also considered to be TLAAs requiring evaluation for the period of extended operation.

The following non-Class 1 Seabrook Station systems that are in scope for license renewal were designed in accordance with ASME Section III Class 2 and 3 requirements:

- Reactor Coolant System (including primary loop piping and pressurizer surge line piping)
- Chemical and Volume Control System
- Safety Injection System
- Primary Component Cooling Water
- Service Water
- Sample System
- Residual Heat Removal System
- Main Steam System
- Condensate and Feedwater Systems
- Steam Generator Blowdown System

**Analysis**

In order to evaluate these TLAAs for 60 years, the number of cycles expected to occur within the 60-year operational period should be compared to the numbers of cycles that were originally considered in the design of these components. If the number of expected cycles does not exceed 7,000 cycles, the minimum number of cycles required that would result in reduction of the allowable stress range, then there is no impact from the added years of service and the original analyses remain valid. If the total number of cycles exceeds 7,000 cycles, then additional evaluation is required.

The 60-year transient projection results shown in Table 4.3.1-3 for Seabrook Station show that even if all of the projected operational transients are added together, the total number of cycles projected for 60 years will not exceed 7,000 cycles. Therefore, there is no impact upon the implicit fatigue analyses used in the component design for the systems designed to ASME Section III Class 2 and 3 requirements.

The Sample System thermal cycles do not trend along with operational cycles because sampling is required on a periodic basis, as opposed to an operational basis. However, only the portion of the sampling lines that constitutes piping need be considered here. In this case that portion turns out to be a very short section of piping directly connected to the RCS loop piping. Since this section of piping has no isolation valve and no bends, it is assumed to always be exposed to primary loop temperature and pressure condition. Similarly since there are no other external piping connections (only the tubing connection exits), the line will not experience any other externally applied loads. Therefore, that section of the sampling line that constitutes ASME Section III Class 2 and 3 piping will only experience the RCS loop transients which have already been shown to be less than 7,000 cycles and the line is, therefore, acceptable.

### Disposition

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for 60 years of operation.

## 4.4 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS

### Summary Description

Thermal, radiation, and cyclical aging analyses of plant electrical and I&C components, developed to meet 10 CFR 50.49, "*Environmental qualification of electric equipment important to safety for nuclear power plants*" requirements, have been identified as time-limited aging analyses (TLAAs) for Seabrook Station. In accordance with 10 CFR 50.49, all electrical equipment important to safety located in a harsh environment and required to function in that environment must be environmentally qualified. In order for a component to have sufficient design margin to perform its important to safety function under harsh environment conditions, the component may need to be periodically rebuilt or replaced. For these EQ components, the EQ program insures that they are rebuilt, replaced or reevaluated at the necessary interval. All qualified lives of components within the scope of the EQ program are managed under the EQ Program.

The Seabrook Station Environmental Qualification (EQ) of Electric Components Program implements aging management activities which are credited for the management of aging in selected components within the scope of 10 CFR 54. The Seabrook Station EQ Program is an existing program that is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) report Section X.E1, "*Environmental Qualification (EQ) of Electric Components*". The program is administered in accordance with the Seabrook Station Environmental Qualification Manual, SSEQ.

As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for electrical components in the Seabrook Station EQ Program that specify a qualification of at least 40 years are TLAAs for license renewal because the criteria contained in 10 CFR 54.3 are met.

### Analysis

Under 10 CFR Part 54.21(c)(1)(iii), the Seabrook Station EQ Program, which implements the requirements of 10 CFR 50.49 (as further defined and clarified by NUREG-0588, and RG 1.89, Rev. 1), is viewed as an aging management program for License Renewal. Reanalysis of an aging evaluation to extend the qualifications of components is performed on a routine basis as part of the Seabrook Station EQ Program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). NUREG-

1800 states that the Staff evaluated the EQ program (10 CFR 50.49) and determined that it is an acceptable aging management program to address environmental qualification according to 10 CFR 54.21(c)(1)(iii). The evaluation referred to in the Standard Review Plan for License Renewal contains sections on "*EQ Component Reanalysis Attributes, Evaluation, and Technical Basis*" that is the basis of the description provided below.

### *EQ Component Reanalysis Attributes*

The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the Seabrook Station EQ Program. While a component life-limiting condition may be due to thermal, radiation or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters such as the assumed ambient temperature of the component, unnecessarily low activation energy, in the state of a component (de-energized versus energized), or equipment operating times. The reanalysis of an aging evaluation is documented according to Seabrook Station quality assurance program requirements, which require the verification of assumptions and conclusions. As already noted, important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed below.

### *Analytical Methods*

The Seabrook Station EQ Program uses the same analytical models in the reanalysis of an aging evaluation as those applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose, which is the normal radiation dose for the projected installed life plus accident radiation dose. For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis. Seabrook has a specific 60 year radiation dose calculation prepared as part of the power up-rate project.

### *Data Collection & Reduction Methods*

The chief method used for a reanalysis per the Seabrook Station EQ Program is reduction of excess conservatism in the component service conditions used in the prior aging evaluation, including temperature, radiation, and cycles. Temperature data used in an aging evaluation is conservative when based on plant design temperatures. Actual plant temperature data are obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors. A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data is used in an aging evaluation in different ways, such as: (a) directly applying the plant temperature data in the evaluation or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis must be justified and documented via the design control program. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.

### *Underlying Assumptions*

Seabrook Station EQ Program component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

### *Acceptance Criteria and Corrective Actions*

Under the Seabrook Station EQ Program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner such that sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful.

## Disposition

**Aging Management, 10 CFR 54.21(c)(1)(iii)** – The effects of aging on the intended functions will be adequately managed for the period of extended operation. The Seabrook Station EQ Program has been demonstrated to be capable of programmatically managing the qualified lives of the components within the scope of the program for License Renewal. The continued implementation of the Seabrook Station EQ Program insures that the aging effects will be managed and that EQ components will continue to perform their intended functions for the period of extended operation. This result meets the requirements of 10 CFR 54.21(c)(iii).

## 4.5 ABSENCE OF TLAA FOR CONCRETE CONTAINMENT TENDON PRESTRESS

### Summary Description

The Seabrook Station Containment and Containment Enclosure Building structures are designed without the use of prestressed tendons. Tendons in presetressed concrete containments lose their prestressing forces over time due to creep and shrinkage of concrete, and relaxation of the prestressed steel tendons. This relaxation is predicted over the period of operating life. Operating experience with the trend of prestressing forces indicates that the prestressed tendons lose their prestressing forces at a rate higher than predicted due to sustained higher temperature. Thus, it is necessary to ensure that the rate of loss of prestressed forces is addressed as TLAAs for the period of extended operation.

### Analysis

The Seabrook Station Unit 1 containment is a seismic Category I reinforced concrete dry structure, designed to function at atmospheric conditions. It consists of an upright cylinder topped with a hemispherical dome, supported on a reinforced concrete foundation mat keyed into the bedrock by the depression for the reactor pit and by continuous bearing around the periphery of the foundation mat.

Located outside the Containment Building and having a similar geometry is the Containment Enclosure Building. This structure provides leak protection for the containment and protects it from certain loads.

### Conclusion

The Containment and Containment Enclosure Building structures are designed without the use of prestressed tendons. Loss of prestressing forces is not applicable to this containment, and therefore not a TLAA.

## 4.6 CONTAINMENT LINER PLATE FATIGUE USAGE AND CONTAINMENT PENETRATION PRESSURIZATION CYCLES

### 4.6.1 CONTAINMENT LINER PLATE FATIGUE USAGE

#### Summary Description

The Containment structure is designed to contain the radioactive material released in the unlikely event of a Loss of Coolant Accident (LOCA). A welded carbon steel liner is attached to the inside face of the concrete shell to provide a leak-tight membrane. Fatigue of the containment liner plate was considered in the original design based on the assumed number of loading cycles that would occur during the life of the plant and is therefore considered a TLAA.

#### Analysis

The original design analysis for the Seabrook Station containment liner plate determined that all of the criteria specified in ASME Section III Article NE-3221.5(d) required for exemption from the requirement to perform a cyclic operation analysis were met. The six criteria to support a fatigue exemption are:

(1) Atmospheric-to-Service Pressure Cycles

(2) Normal Service Pressure Fluctuations

(3) Temperature Difference – Startup and Shutdown

(4) Temperature Difference – Similar Material

(5) Temperature Difference – Dissimilar Materials

(6) Mechanical Loads

The Seabrook Station analyses confirmed the 40-year anticipated stress cycles listed below would satisfy the exemption criteria of NE 3221.5(d).

- Atmospheric-to-service pressure cycles (120 cycles)
- Temperature difference from Startup to Shutdown (120 cycles)
- Operating Basis Earthquake (500 cycles)
- LOCA (10 cycles)

To address these 40-year cycles during the period of extended operation, a re-evaluation of the six fatigue exemption requirements utilizing anticipated 60-year stress cycles specified below was performed.

- Atmospheric-to-service pressure cycles (180 cycles)
- Temperature difference from Startup to Shutdown (180 cycles)
- Operating Basis Earthquake (750 cycles)
- LOCA (15 cycles)

The result of this analysis determined that the specified conditions through the period of extended operation continue to satisfy the requirement for exemption from analysis for cyclic operation in accordance with in ASME Section III Article NE-3221.5(d).

### Disposition

**Revision, 10 CFR 54.21(c)(1)(ii)** – The analysis has been projected to the end of the period of extended operation

## 4.6.2 PRESSURIZTAION CYCLES: PERSONNEL AIRLOCK, EQUIPMENT HATCH AND FUEL TRANSFER TUBE ASSEMBLY ABSENCE OF TLAA FOR CONTAINMENT PENETRATIONS

### Summary Description

Similar to the steel Containment Liner Plate discussed in Section 4.6.1, containment penetrations are designed to ensure a leak-tight membrane to contain the radioactive material released in the unlikely event of a Loss of Coolant Accident (LOCA). The design of the containment penetrations did not involve cyclic evaluations and therefore are not considered TLAAs. Specific cyclic evaluations are listed in the Seabrook Station UFSAR for the Personnel Airlock, Equipment Hatch and Fuel Transfer Tube therefore TLAAs are considered.

### Analysis

The design of the Seabrook Station containment penetrations was searched for evidence of cyclic evaluations. In each case, the design process was a comparison of design stresses to a stress limit independent of the number of load cycles and with no fatigue analysis. Comparison to other containment systems designed by United Engineers and Constructors (UE&C) resulted in similar design criteria.

UFSAR Section 3.8.2.3 lists the cyclic loads considered in the design of the personnel airlock and equipment hatch which include:

- 120 cycles of plant startup and shutdown
- 400 OBE cycles
- 100 SSE cycles
- 1 accident cycle (LOCA)
- 160 pressure test cycles

UFSAR section 3.8.2.3 lists the cyclic considered in the design of the fuel transfer tube assembly which include:

- 400 OBE cycles
- 1 accident cycle (LOCA)
- 160 pressure test cycles
- 1000 temperature cycles

The anticipated number of cycles for the Personnel Airlock, Equipment Hatch and Fuel Transfer Tube Assembly projected to occur during the period of extended operation is bounded by the original design.

**Disposition**

***Personnel Airlock, Equipment Hatch and Fuel Transfer Tube***

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses for the Personnel Airlock, Equipment Hatch and Fuel Transfer Tube remains valid for the period of extended operation as the anticipated number of cycles anticipated during the period of extended operation is bounded by the original design.

***Containment Penetrations***

**Conclusion,** Because the design of the containment penetrations did not involve cyclic analysis, and the plant will continue to operate within the design envelope in the period of extended operation, the analyses of the containment penetrations are not considered TLAAs.

## 4.7 PLANT-SPECIFIC TIME LIMITED AGING ANALYSES

### 4.7.1 ABSENCE OF A TLAA FOR REACTOR VESSEL UNDERCLAD CRACKING ANALYSES

#### Summary Description

Crack growth due to cyclic loading could occur in reactor vessel shell forgings clad with stainless steel by a high-heat-input welding process. Growth of intergranular separations (underclad cracks) in the heat-affected zone under austenitic stainless steel cladding is a TLAA to be evaluated for the period of extended operation for all SA-508 Class 2 forgings with cladding deposited by high-heat-input welding.

Growth of intergranular separations (underclad cracks) in the heat-affected zone under austenitic steel cladding is not an applicable aging effect. The supplementary criteria identified in Regulatory Guide 1.43 were implemented to give reasonable assurance that underclad cracking was avoided in production weld cladding. From Table 5.3-3 of UFSAR:

> Closure Head Flange, Vessel Flange, Inlet Nozzle and Outlet Nozzle – SA-508 Class 2 from UFSAR 5.3.1.2 Special Processes Used For Manufacturing and Fabrication:
>
> i. The procedure qualification for cladding low alloy steel (SA-508, Class 2) requires a special evaluation to assure freedom from underclad cracking.

#### Analysis

For the Seabrook Station reactor vessel, the cladding of the reactor vessel SA-508 Class 2 forgings did not use a high-heat-input welding process which could induce underclad cracking; therefore, the Seabrook Station reactor vessel is not susceptible to underclad cracking.

#### Conclusion

No TLAA is assigned for Reactor Vessel underclad cracking analyses at Seabrook Station. The application of cladding to the Reactor Vessel SA-508, Class 2 forgings is not susceptible to underclad cracking, because high-heat-input welding processes, which could induce underclad cracking, were not used.

## 4.7.2 REACTOR COOLANT PUMP FLYWHEEL FATIGUE CRACK GROWTH ANALYSES

### Summary Description

Westinghouse Report WCAP-14535-A, Rev. 0, "*Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination*" includes a fatigue crack growth analysis that has been identified as a TLAA. The report was submitted for NRC review and the NRC issued a Safety Evaluation Report in September 1996. The purpose of the report was to provide an engineering basis for elimination of reactor coolant pump (RCP) flywheel inservice inspection requirements for all operating Westinghouse plants and certain Babcock and Wilcox plants. The number of cycles (pump starts and stops) used in this report was 6,000 for a 60-year plant life. Crack growth was shown to be negligible from exposure to these 6,000 cycles.

### Analysis

The number of cycles (pump starts and stops) for a 60-year plant life was assumed to be 6,000 for this analysis. Crack growth was shown to be negligible from exposure to these 6,000 cycles. Table 4.4.2-1 provides the current and 60-year projected number of RCP start/stop cycles for Seabrook.

**Table 4.4.2-1 Seabrook Station 60-Year Projected Reactor Coolant Pump Start/Stop Cycles**

| RCP Identification | Current Cycle Count [(1)] | 60-Year Cycle Projection [(2)] |
|---|---|---|
| 1A | 152 | 536 |
| 1B | 152 | 536 |
| 1C | 152 | 536 |
| 1D | 152 | 536 |

(1) Data obtained from Seabrook Station Cycle Counting records.

(2) Based on 18.6 years of operations (Refer to Section 4.3.1).

From Table 4.4.2-1, the projected number of RCP start/stop cycles for the Seabrook Station RCP flywheels are much less than the analyzed 6,000 cycles.

### Disposition

**Validation, 10 CFR 54.21(c)(1)(i)** – Since the number of analyzed start/stop cycles exceeds the 60-year cycle projections, the reactor coolant pump flywheel analysis remains valid for the period of extended operation.

### 4.7.3 LEAK-BEFORE-BREAK ANALYSES

**Summary Description**

Title 10 Part 50 Appendix A, General Design "*Criteria for Nuclear Power Plants*" Criterion 4 of the Code of Federal Regulations allows for the use of leak-before-break (LBB) methodology for excluding the dynamic effects of postulated ruptures in reactor coolant system piping. The fundamental premise of the LBB methodology is that the materials used in nuclear power plant piping are sufficiently tough, that even a large through-wall crack would remain stable and would not result in a double-ended pipe rupture. Application of the LBB methodology is limited to those high-energy fluid systems not considered to be overly susceptible to failure from such mechanisms as corrosion, water hammer, fatigue, thermal aging or indirectly from such causes as missile damage or the failure of nearby components. The analyses involved with LBB are considered TLAAs.

**Analysis**

A LBB analysis was initially performed for Seabrook Station primary loop piping in 1984. To demonstrate the elimination of RCS primary loop pipe breaks, the following objectives had to be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate.
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.
- Demonstrate margin on the applied load.
- Demonstrate that fatigue crack growth is negligible.

The initial analysis was reviewed to demonstrate compliance with LBB technology for Seabrook Station. Review of the 40-year LBB analysis considered input from the Stretch Power Uprate (2005) and the Mechanical Stress Improvement Process (MSIP) application at one of the reactor vessel primary hot leg nozzle locations (2009). Review for the period of extended operation was based on the same set of design transients as the original analyses; therefore the conclusions of the original evaluation remain valid for the 60-year period.

Plant specific geometry, operating parameters, loading, and material properties were used in the fracture mechanics evaluation. The mechanical properties were determined at operating temperatures. Since the piping systems also include cast austenitic stainless steel (CASS) piping components, fracture toughness considering thermal aging was determined for each affected component's heat of material for the fully aged condition.

Based on loading, pipe geometry, and fully aged fracture toughness considerations, enveloping governing locations were determined at which LBB crack stability evaluations were made. Through-wall flaw sizes were found which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the reactor coolant system primary loop piping. The thermal transients used in the fatigue crack growth analysis were Seabrook Station design transients and projected cycles, which are reported in Table 4.3.1-3. The corresponding 60-year projected cycles, also shown in Table 4.3.1-3 are lower than the 40-year design values. Therefore, the numbers of design cycles assumed in the analysis bound the numbers of design cycles projected for 60 years of operation.

**Disposition:**

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation. The LBB review demonstrates that the previous LBB conclusions still remain valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the Seabrook Station design basis for the period of extended operation.

### 4.7.4 HIGH ENERGY LINE BREAK POSTULATION BASED ON CUMULATIVE USAGE FACTOR

**Summary Description**

The Seabrook Station High Energy Line Break (HELB) analysis used a screening criterion of CUF greater than 0.1 to identify areas of investigation. The Seabrook Station Updated Final Safety Analysis Report (UFSAR) Section 3.6(B).2.1(a) provided a basis to eliminate locations in each piping run or branch run from further consideration as high energy line break locations on the basis of low fatigue usage, including intermediate locations where the cumulative usage factor was less than 0.1. These locations are considered TLAAs as the HELB analysis is based on a set of anticipated design transients and must be evaluated for the period of extended operation.

Selection of pipe failure locations for evaluation of the consequences on nearby essential systems, components, and structures, except for the reactor coolant loop, is in accordance with Regulatory Guide 1.46, and NRC Branch Technical Positions ASB 3-1 and MEB 3-1. A revised stress analysis also permitted omission of the surge line intermediate breaks. A leak-before-break

(LBB) analysis eliminated large breaks in the main reactor coolant loops. See Section 4.7.3.

### Analysis

The citation of MEB 3-1 means that break locations in piping with ASME Section III Class 1 fatigue analyses are identified based on cumulative usage factor (with the stated exception of the reactor coolant system primary loops), and that these determinations are therefore TLAAs.

The surge line intermediate break locations were eliminated based on usage factor. The most recent piping analysis confirmed the elimination of these break locations. The analysis that justified the elimination of these intermediate locations in the surge line is therefore a TLAA.

The same would be true of other line sections with no intermediate locations with fatigue usage factors above 0.1, if this analysis result were used to eliminate intermediate breaks, that is, the determination that there are no intermediate breaks in these sections based on a low usage factor would, for the same reason, be a TLAA. However, no additional cases similar to the surge line occur in the Seabrook Station licensing basis.

The scope of these HELB location TLAAs is therefore limited to ASME Section III Class 1 piping connected to the RCS from the RCS primary coolant loops, to the ASME Class I/II piping interface. Since the 60 year projected cycles are bounded by the original design cycles, the present intermediate locations with CUF less than 0.1 remain valid for the period of extended operation.

Seabrook Station has containment penetration break exclusion regions. However, these break exclusion regions do not contain any ASME Section III Class 1 piping with fatigue analyses, and their qualification is therefore based only on calculated stress. The break locations in these no break zones are therefore independent of time and are not supported by a TLAA.

### Disposition

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation.

## 4.7.5 FUEL TRANSFER TUBE BELLOWS DESIGN CYCLES

### Summary Description

The fuel transfer tube assembly connects the fuel transfer canal (inside the containment structure) to the transfer pool (inside the spent fuel handling building). The fuel transfer tube assembly passes through the containment wall and through the exterior wall of the spent fuel handling building. The fuel transfer tube assembly is comprised of a 24-inch diameter penetration sleeve penetrating through the containment and spent fuel building walls and three (3) sets of expansion joints (bellows). The penetration sleeve and the three bellows perform a water-retaining intended function, and are within the scope of license renewal.

The fatigue analysis for each of the three bellows is based on the consideration of 20 occurrences of the Operating Basis Earthquake, each occurrence having 20 cycles of maximum response therefore, this design analysis is a TLAA requiring evaluation for the period of extended operation.

### Analysis

In order to determine if the design analyses remain valid for 60 years of operation, the number of seismic cycles for 60 years has been projected. As of January 2010, the Seabrook transfer tube bellows will have been exposed to zero (0) Operating Basis Earthquake (OBE) cycles. It is projected that 1 OBE would occur for Seabrook in 60 years of operation. Therefore, since the number of cycles in 60 years is well below the 20 seismic movement cycles analyzed for these bellows, these design analyses remain valid for the period of extended operation.

### Disposition

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation.

## 4.7.6 CRANE LOAD CYCLE LIMITS

## 4.7.6.1 POLAR GANTRY CRANE

### Summary Description

The design specification for the 420/50-ton Polar Crane in the containment structure at Seabrook Station required that the crane conform to the design requirements of Crane Manufacturers Association of America (CMAA) Specification 70, "*Specifications for Electric Overhead Traveling Cranes*". Service requirements specified for the design of this crane correspond to the cyclic loading requirements of CMAA 70, Class A. This evaluation of cycles over the 40 year life is the basis of a safety determination and is, therefore, a TLAA.

The Polar Crane was designed for up to 100,000 load cycles per criteria of CMAA Specification 70 for service Class A. This service class covers cranes which may be used in installations such as power houses, public utilities, turbine rooms, motor rooms and transformer stations where precise handling of equipment at slow speeds with long, idle periods between lifts are required. Capacity loads may be handled for initial installation of equipment and for infrequent maintenance..

**Analysis**

The estimated number of lifts for the Polar Crane over the remaining 40 years of service (which includes 20 years of Extended Operation) is 19,440 with most of the lifts being less than 2500 pounds. This rate is based on refueling outage use, therefore, the first 20 years of service life for the Polar Crane would include approximately 10,000 load cycles. Thus, the total service life load cycles will be approximately 30,000. Since the total number of lifts is at the low end of the allowable design value of up to 100,000 cycles, the Polar Crane load cycle fatigue analyses for Seabrook Station remains valid for 60 years of plant operation.

**Disposition**

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation.

**4.7.6.2 CASK HANDLING CRANE**

**Summary Description**

The original Seabrook Station Cask Handling Crane was replaced in 2008 by a single failure-proof crane rated for 130 tons (main hoist) and 5 tons (each of two auxiliary hoists). To meet single failure criteria, each of these cranes was designed to the requirements of ASME NOG-1-2004, NUREG-0554, and NUREG-0612. The cranes were also designed to Crane Manufacturers Association of America (CMAA) Specification 70, "*Specifications for Electric Overhead Traveling Cranes*", with an allowable design life cycle range of up to 100,000 cycles. This evaluation of cycles over the projected 40-year life is the basis of a safety determination and has been identified as a TLAA requiring evaluation for the period of extended operation.

**Analysis**

Although the new crane became operational in 2008, the bridge structure is original equipment. The period of extended operation expires in 2050, resulting in 60 years of operation. The projected number of lifts for the Cask Handling Crane is less than 500. This estimate is based upon the expected number of casks that must be handled during each cask loading campaign and the projected number of campaigns through the period of extended operation. Allowing for double that number for minor lifts, or 1000 cycles, the estimated number of lifts for the Cask Handling Crane, 1500 cycles, is much less than the maximum allowable design value of 100,000 cycles, the Cask

Handling Crane load cycle fatigue analyses remain valid for 60 years of plant operation.

**Disposition**

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation.

### 4.7.7 SERVICE LEVEL I COATINGS QUALIFICATION

**Summary Description**

Service Level 1 coatings used at Seabrook Station are in compliance with the applicable ANSI standards for coating systems inside containment. In a design base accident, the Emergency Core Cooling System (ECCS) at Seabrook Station pumps water from inside the containment sump to the reactor vessel to keep the core covered with water and make up losses from the pipe break location. These coatings could potentially detach during a design basis accident and the coating debris could contribute to flow blockage of ECCS suction strainers. The ECCS has suction piping located below the waterline inside the sump. Since it is assumed that the degree of radiation exposure used in the original qualification testing was intended to bound 40 years of operation, qualification of Service Level 1 coatings is considered a TLAA.

**Analyses**

Industry operating history has shown that undesirable degradation, detachment, and other types of failures of coatings have occurred. Detached coatings from the substrate that are transported to Emergency Core Cooling System (ECCS) suction strainers could make those systems unable to satisfy the requirement in 10 CFR 50.46(b)(5) to provide long-term cooling.

Significant quantities of coated surfaces inside containment that would be exposed to the post Loss of Coolant Accident environment are listed in UFSAR Table 6.1(B)-2. The coating systems for these surfaces are epoxy-based Keeler & Long coating systems designed for a 40-year life and are in compliance with the applicable ANSI standards for coating systems inside containment. The qualification of the coatings to withstand the effects of radiation and the design basis accident conditions assures these coatings will remain in place and not contribute to clogging of ECCS strainers beyond analyzed limits.

The coatings used for Service Level 1 applications at Seabrook Station were qualified and applied in accordance with the requirements of the following documents:

- NRC Regulatory Guide 1.54, *"Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,"* June, 1973
- ANSI N101.4-1972, *"Quality Assurance for Protective Coatings Applied to Nuclear Facilities"*
- ANSI N101.2 – 1972, *"Protective Coatings (Paints) for Light Water Nuclear Containment Facilities"*
- ANSI N512-1974, *"Protective Coatings (Paints) for the Nuclear Industry"*

The maintenance rule, 10 CFR 50.65, "*Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*," includes in its scope safety-related SSCs that are relied upon to remain functional during and following design basis events with respect to specified functions and non-safety-related SSCs

(1) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures,

(2) Whose failure could prevent safety-related SSCs from fulfilling their safety-related function, and

(3) Whose failure could cause a reactor scram or an actuation of a safety-related system.

To the extent that protective coatings meet these criteria, they are within the scope of the maintenance rule. The maintenance rule requires that Seabrook Station monitor the effectiveness of maintenance for protective coatings within its scope (as discrete systems or components or as part of any SSC), or demonstrate that their performance or condition is being effectively controlled through the performance of appropriate preventive maintenance, in accordance with 10 CFR 50.65(a)(1) or (a)(2), as appropriate.

**Disposition**

**Aging Management** - 10 CFR 54.21(c)(1)(iii) - Seabrook Station Service Level I Coatings are managed by the ASME Section XI, Inservice Inspection, Subsection IWE Program, B.2.1.27 and Procedure for Application of Service Level I Coatings. Seabrook Station periodically conducts condition assessments of Service Level I coatings inside containment. Coating inspections are performed at the beginning and at the end of each refueling outage. Inspections at the beginning of the refueling outage are performed by a NextEra Energy Coating Supervisor and a Design Engineer for peeling coatings that have the potential of falling into the reactor or Containment Building Spray recirculation sumps.

At the end of the refueling outage, the NextEra Energy Coating Supervisor notifies the design engineer and Nuclear Coating Specialist to perform the Containment Closeout Inspection. The inspection assesses the condition of the Containment coating. The inspection includes accessible coated areas of the Containment and equipment including, but not limited to, the following items: concrete and steel surfaces, liner plate wall to concrete floor joint, liner plate wall penetrations, personnel and equipment hatches, and personnel air locks. As localized areas of degraded coatings are identified, those areas are evaluated and scheduled for repair or replacement, as necessary. The periodic condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings which may be susceptible to detachment from the substrate during a LOCA event is minimized.

### 4.7.8 ABSENCE OF A TLAA FOR REACTOR COOLANT PUMP: CODE CASE N-481

#### Summary Description

ASME Boiler and Pressure Vessel Code, Section XI, specifies that a volumetric inspection of the reactor coolant pump casing welds and a visual inspection of pump casing internal surfaces be performed on a reactor coolant pump within each 10-year inspection period. These 10-year volumetric inspections are significant because the reactor coolant pumps have already been welded to the piping and the pumps must be disassembled in order to gain access to the inside surface of the cast stainless steel casings. In recognition of these difficulties, ASME Code Case N-481, "*Alternative Examination Requirements for Cast Austenitic Pump Casings*", was developed to allow for the replacement of volumetric examinations with fracture mechanics, based evaluation and supplemented by specific visual inspections.

#### Analyses

The Seabrook Station UFSAR Section 5.4.1.4 *"Tests and Inspections"* states, "The reactor coolant pumps can be inspected in accordance with the ASME Code, Section XI, for in-service inspection of nuclear reactor coolant systems." The Seabrook Station pump casings are cast in one piece, eliminating welds in the casing. In addition, support feet are cast integral with the casing to eliminate a weld region.

#### Conclusion

The Seabrook Station Reactor Coolant Pump (RCP) casings are single castings, which contain no welds. Therefore, Code Case N-481, "*Alternative Examination Requirements for Cast Austenitic Pump Casings*", is not applicable to Seabrook Station RCP.

### 4.7.9 CANOPY SEAL CLAMP ASSEMBLIES

**Summary Description**

The canopy seal clamp assemblies were designed for a 40 year design life on the basis of meeting stress limits. The original fatigue analysis considered the forces that would be applied to the center head adapter which maximized the moments on the J-Grove weld and moment along the length of the adapter. The fatigue analysis for the Canopy Seal Clamps is based on the consideration of 400 cycles consisting of 20 occurrences of the Operating Basis Earthquake, each occurrence having 20 cycles of maximum response. This design analysis is a TLAA requiring evaluation for the period of extended operation.

**Analysis**

In order to determine if the design analyses remain valid for 60 years of operation, the number of seismic cycles for 60 years has been projected. As of January 2010, the Seabrook Station Canopy Seal Clamps have been exposed to zero (0) Operating Basis Earthquake (OBE) cycles. It is projected that 1 OBE would occur for Seabrook Station in 60 years of operation. Therefore, since the number of cycles in 60 years is well below the 20 seismic movement cycles analyzed for these clamps, these design analyses remain valid for the period of extended operation.

**Disposition**

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation. The canopy seal clamp assemblies are projected to continue to operate within the same stress limits in the extended operating period and will remain acceptable for the period of extended operation.

### 4.7.10 HYDROGEN ANALYZER

**Summary Description**

The Seabrook Station Hydrogen Analyzer was evaluated with respect to radiation exposure. The UFSAR contains accumulated radiation dose limits for a 40 year operating period. The Hydrogen Analyzer is a safety related component. The analysis of the effects of radiation considered the annual accumulated dosage, and projected these values, to demonstrate that the maximum dose limits specified in the UFSAR will not be exceeded in a 60 year operating period.

### Analysis

The post accident Hydrogen Analyzer must perform a safety function following a Loss of Coolant Accident (LOCA). Excessive radiation exposure could jeopardize the ability to perform this safety function. The Seabrook Station UFSAR Table 6.2-84 defines the Hydrogen Analyzer design parameters and maximum radiation dose limits for forty years of normal operation.

The operational dose for 40 years is $5 \times 10^6$ rads.

The projected maximum 40 year exposure comes from three sources; the gas in the analyzers themselves, the gas in the piping in the room, and the shine from the containment atmosphere through the penetrations into the room.

The predicted one year total integrated dose for the Seabrook Station Hydrogen Analyzer is comprised as follows:

| | |
|---|---|
| From the analyzers, | $5.8\text{x}10^6$ mrads |
| From the piping, | $0.2\text{x}10^6$ mrads |
| From the penetrations, | $1.2\text{x}10^6$ mrads |
| Total | $7.2\text{x}10^6$ mrads = $7.2\text{x}10^3$ rads |

Thus, the maximum predicted dose for a 60 year plant operating period is 60-years x $7.2\text{x}10^3$ rads/year. = $4.32\text{x}10^5$ rads. This is an order of magnitude less than the specified UFSAR radiation dose limit. Therefore, the Seabrook Station Hydrogen Analyzer is shown to be acceptable for the period of extended operation.

### Disposition

**Validation, 10 CFR 54.21(c)(1)(i)** – The analysis remains valid for the period of extended operation.

## 4.7.11 MECHANICAL EQUIPMENT QUALIFICATION

### Summary Description

The Seabrook Station CLB commits to the review and evaluation of the environmental qualification of mechanical equipment to demonstrate compliance with General Design Criteria 4 of Appendix A to 10 CFR Part 50.

Results of this evaluation demonstrate safety-related active mechanical equipment located in harsh environments had been adequately addressed.

Since a period of 40 years was used to determine the normal service radiation exposure to the equipment, mechanical equipment qualification (MEQ) is considered a TLAA.

### Analysis

The design basis event conditions during the period of extended operation will remain the same as those in the current license period, which have been adjusted to account for previously approved power uprate conditions. Therefore, the design basis event parameters, including the temperature, pressure, and time profiles, do not require further evaluation as TLAAs for license renewal.

### Disposition

**Revision, 10 CFR 54.21(c)(1)(ii)** – The effects of aging on the intended function(s) of equipment included under Mechanical Equipment Qualification will be adequately addressed for the period of extended operation. Calculations for Mechanical Equipment Qualification will be revised prior to entering the period of extended operation. Revision of MEQ calculations will be accomplished using techniques currently used under the CLB for equipment qualification including analytical methods, replacement of radiation sensitive materials or equipment replacement.

## 4.7.12 ABSENCE OF A TLAA FOR METAL CORROSION ALLOWANCES AND CORROSION EFFECTS

### Summary Description

Nuclear plant components are commonly designed with corrosion allowances, and TLAAs of corrosion effects for the 40-year design life.

### Analysis

A review of the Seabrook Station licensing basis found no description of time-dependent corrosion allowances, rates, or corrosion-dependent design lives of pressure vessels, system components, piping, or metal containment components.

### Conclusion

There are no TLAAs for metal corrosion allowances and corrosion effects.

### 4.7.13 ABSENCE OF A TLAA FOR INSERVICE FLAW GROWTH ANALYSES THAT DEMONSTRATE STRUCTURAL STABILITY FOR 40 YEARS

**Summary Description**

Defects discovered by inservice inspection or component failures may be repaired or replaced to restore the basis of the original design analysis; may be repaired or replaced to a different configuration, or may be analyzed to confirm that the as-found condition is acceptable. For ASME components these activities are controlled by Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components."* A flaw analysis of a Class 1 component usually requires a fatigue crack growth analysis, which is a TLAA if it qualifies the component for the plant design life.

**Analysis**

A thorough review of the Seabrook Station licensing basis, supported by interviews with plant staff familiar with the history of Class 1 components, found the following fatigue crack growth analyses:

- Fatigue crack growth and fracture mechanics stability analyses in support of pressurizer nozzle overlays. The overlays were supported by fatigue crack growth analyses. These fatigue crack growth analyses were projected to the end of the period of extended operation, and are therefore not TLAAs. See Section 4.3.6.

- Fatigue crack growth assessments and fracture mechanics stability analyses in support of the leak-before-break (LBB) evaluation. Review for the period of extended operation was based on the same set of design transients as the 40-year analyses; therefore the conclusions of the original evaluation remain valid for the 60-year period and are therefore not TLAAs. See Section 4.7.3.

- Fatigue crack growth and fracture mechanics stability analyses of Mechanical Stress Improvement Process (MSIP) repairs to Alloy 600 material in reactor coolant hot legs. The MSIP repair was supported by fatigue crack growth analysis. This fatigue crack growth analysis was projected to the end of the period of extended operation, and is therefore not a TLAA. See Section 4.3.6.

**Conclusion**

These fatigue crack growth analyses are not TLAAs because they qualify the affected components for the period of extended operation.

## 4.7.14 DIESEL GENERATOR THERMAL CYCLE EVALUATION

### Summary Description

The Emergency Diesel Generators were designed for a 40 year design life on the basis of 5454 Total Equivalent Full Temperature Cycles.

### Analysis

The Emergency Diesel Generators provide Emergency Power to Buses 5 and 6. The Emergency Diesel Generators were analyzed for thermal cycling by the engine manufacturer for Environmental Qualification in accordance with IEEE 323. The manufacturer qualified the Diesel Generator for 5454 Full Temperature Cycles for the forty year design life of the plant. This would amount to starting the Diesel once per week since installation to the end of the period for extended operation. Under current plant operating practices, the Emergency Diesel Generators are operated only occasionally during periodic surveillance and maintenance testing. Monthly testing over 60 years would contribute 720 cycles assuming an equal number of starts for maintenance and actual events an additional 1440 cycles could occur. These actual and potential cycles combined equal slightly more than 2000 cycles for the Emergency Diesel Generators. It is, therefore, unlikely that the 5454 assumed cycles will be approached during the period of extended operation. Thus, the existing analysis is considered to remain valid for the period of extended operation, and there is reasonable assurance that the intended function will be maintained.

### Disposition

**Validation, 10 CFR 54.21(c)(1)(i)** – The analyses remain valid for the period of extended operation.

## 4.8 GENERAL REFERENCES


4-1. NEI 95-10, Revision 6, “Industry Guidance for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” June 2005.

4-2. NUREG-1800, Revision 1, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” September 2005.

4-3. USNRC Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” March 2001.

4-4. USNRC Regulatory Guide 1.99, Revision 2, “”Radiation Embrittlement of Reactor Materials,” May 1988.

4-5. 10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants, U. S. Nuclear Regulatory Commission.

4-6. USNRC Regulatory Guide 1.89, Rev. 1, 6/84, Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,

4-7. USNRC Regulatory Guide 1.97, Rev. 0, 3/76, Instrumentation of Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.

4-8. USNRC Generic Safety Issue-190, Fatigue Evaluation of Metal Components for 60-Year Plant Life.

4-9. 10 CFR 50.60, Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation, Code of Federal Regulations, U. S. Nuclear Regulatory Commission.

4-10. Appendix G, 10 CFR 50, Fracture Toughness Requirements, Code of Federal Regulations, U. S. Nuclear Regulatory Commission.

4-11. Appendix H, 10 CFR 50, Reactor Vessel Material Surveillance Requirements, Code of Federal Regulations, U. S. Nuclear Regulatory Commission.

4-12. Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Vessel Pressure Code, American Society of Mechanical Engineers, July 1986.

4-13. 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events, U. S. Nuclear Regulatory Commission.

4-14. USNRC Inspection and Enforcement Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems," June 1988.

4-15. USNRC Inspection and Enforcement Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," December 1988.

4-16. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

4-17. NUREG/CR-5704, "Effects of LWR Coolant Environments of Fatigue Design Curves of Austenitic Stainless Steels."

4-18. NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels."

4-19. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants", May 1979

4-20. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", July 1980

4-21. ANSI N45.2, 1971, "Quality Assurance Program Requirements for Nuclear Power Plants,"

4-22. USNRC Regulatory Guide 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants, June, 1973

4-23. ANSI N101.4-1972, Quality Assurance for Protective Coatings Applied to Nuclear Facilities

4-24. ANSI N101.2 – 1972, Protective Coatings (Paints) for Light Water Nuclear Containment Facilities

4-25. ANSI N512-1974, Protective Coatings (Paints) for the Nuclear Industry

4-26. WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," November 1996.

4-27. Technical Specifications NextEra Energy Seabrook, LLC, Et Al.* Docket No. 50-443 Seabrook Station, Unit No. 1.

4-28. Seabrook Station Updated Final Safety Analysis Report, Revision 13

4-29. Westinghouse. Report WCAP-14535-A, Rev. 0, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination

4-30. Westinghouse. Report WCAP-10567, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Seabrook Units 1 and 2," June 1984.

4-31. Westinghouse. Report WCAP 14040-NP-A Rev 2 " Methodology used to Develop Cold Overpressure Mitigating system Setpoints and RCS Heatup and Cooldown Limit Curves" January 1996

4-32. Westinghouse. Report WCAP 15745, Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation, December 2001

4-33. Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

4-34. Crane Manufacturers Association of America (CMAA) Specification 70. Specifications for Electric Overhead Traveling Cranes, 2004

4-35. North Atlantic Energy Service Company T. C. Feigenbaum (NAESC) to the USNRC Document Control Desk, : Revision to Limiting Reactor Vessel Beltline Material January 18, 1999

USNRC, "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," Generic Letter 98-04, July 14, 1998.2

4-36. Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants," IWE-1000.

# APPENDIX A

# UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

## APPENDIX A - Table of Contents

## A.1 INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a Updated Final Safety Analysis Report (UFSAR) Supplement. This appendix, which includes the following sections, comprises the UFSAR supplement:

| | |
|---|---|
| Section A.1.1 | contains a listing of the aging management programs that correspond to NUREG-1801 Chapter XI programs. |
| Section A.1.2 | contains a listing of the plant-specific aging management programs. |
| Section A.1.3 | contains a listing of aging management programs that correspond to NUREG-1801 Chapter X programs associated with Time- Limited Aging Analyses. |
| Section A.1.4 | contains a listing of the Time-Limited Aging Analyses (TLAA). |
| Section A.1.5 | contains a discussion of the Quality Assurance Program and Administrative Controls. |
| Section A.2.1 | contains a summarized description of the NUREG-1801 Chapter XI programs for managing the effects of aging. |
| Section A.2.2 | contains a summarized description of the plant-specific programs for managing the effects of aging. |
| Section A.2.3 | contains a summarized description of the NUREG-1801 Chapter X programs that support the TLAAs. |
| Section A.2.4 | contains a summarized description of the TLAAs applicable to the period of extended operation. |
| Section A.3 | contains the License Renewal Commitment List. |

The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that system, structures, and components (SSC) within the scope of license renewal will continue to perform their intended functions consistent with the Current Licensing Basis (CLB) for the period of extended operation.

## A.1.1 NUREG-1801 CHAPTER XI AGING MANAGEMENT PROGRAMS

The following list of aging management programs correspond to NUREG-1801 Chapter XI programs.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (A.2.1.1)
2. Water Chemistry (A.2.1.2)
3. Reactor Head Closure Studs (A.2.1.3)
4. Boric Acid Corrosion (A.2.1.4)
5. Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (A.2.1.5)
6. (Not Used)
7. PWR Vessel Internals (A.2.1.7)
8. Flow-Accelerated Corrosion (A.2.1.8)
9. Bolting Integrity (A.2.1.9)
10. Steam Generator Tube Integrity (A.2.1.10)
11. Open-Cycle Cooling Water System (A.2.1.11)
12. Closed-Cycle Cooling Water System (A.2.1.12)
13. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (A.2.1.13)
14. Compressed Air Monitoring (A.2.1.14)
15. Fire Protection (A.2.1.15)
16. Fire Water System (A.2.1.16)
17. Aboveground Steel Tanks (A.2.1.17)
18. Fuel Oil Chemistry (A.2.1.18)
19. Reactor Vessel Surveillance (A.2.1.19)
20. One-Time Inspection (A.2.1.20)
21. Selective Leaching of Materials (A.2.1.21)
22. Buried Piping and Tanks Inspection (A.2.1.22)
23. One-Time Inspection of ASME Code Class 1 Small Bore-Piping (A.2.1.23)
24. External Surfaces Monitoring (A.2.1.24)
25. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (A.2.1.25)

26. Lubricating Oil Analysis (A.2.1.26)
27. ASME Section XI, Subsection IWE (A.2.1.27)
28. ASME Section XI, Subsection IWL (A.2.1.28)
29. ASME Section XI, Subsection IWF (A.2.1.29)
30. 10 CFR 50, Appendix J (A.2.1.30)
31. Structures Monitoring Program (A.2.1.31)
32. Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements (A.2.1.32)
33. Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits (A.2.1.33)
34. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 EQ Requirements (A.2.1.34)
35. Metal Enclosed Bus (A.2.1.35)
36. Fuse Holders (A.2.1.36)
37. Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements (A.2.1.37)

## A.1.2 PLANT SPECIFIC AGING MANAGEMENT PROGRAMS

The plant-specific aging management programs are listed below.

1. 345 KV $SF_6$ Bus (A.2.2.1)
2. Boral Monitoring (A.2.2.2)
3. Nickel Alloy Nozzles and Penetrations (A.2.2.3.)

## A.1.3 NUREG-1801 CHAPTER X AGING MANAGEMENT PROGRAMS

The following list of aging management programs correspond to NUREG-1801 Chapter X programs associated with Time-Limited Aging Analysis .

1. Metal Fatigue of Reactor Coolant Pressure Boundary (A.2.3.1) ]
2. Environmental Qualification (EQ) of Electrical Components (A.2.3.2)]

## A.1.4 TIME-LIMITED AGING ANALYSIS SUMMARIES

Summaries of the Time-Limiting Aging Analyses applicable for the period of extended operation are listed below.

1. Neutron Embrittlement of the Reactor Vessel (A.2.4.1)
2. Metal Fatigue of Vessels and Piping (A.2.4.2)
3. Environmental Qualification (EQ) of Electrical Equipment (A.2.4.3)
4. Fatigue of the Containment Liner and Penetrations. (A.2.4.4)
5. Other Plant-Specific TLAAs (A.2.4.5)

### A.1.5 QUALITY ASSURANCE PROGRAM AND ADMINISTRATIVE CONTROLS

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2, "*Quality Assurance For Aging Management Programs (Branch Technical Position IQMB-1)*" of NUREG-1800 "*Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*". The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and these elements are applicable to the safety-related and non-safety related systems, structures, and components (SSC) that are subject to Aging Management Review (AMR).

## A.2 AGING MANAGEMENT PROGRAMS

### A.2.2.1 NUREG-1801 CHAPTER XI AGING MANAGEMENT PROGRAMS

#### A.2.1.1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

American Society of Mechanical Engineer (ASME) Section XI, Subsections IWB, IWC, IWD Inservice Inspection Program facilitates inspections to identify and correct degradation in Class 1, 2, and 3 piping, components, and integral attachments. The program includes periodic visual, surface and/or volumetric examinations of all Class 1, 2 and 3 pressure-retaining components, their supports and integral attachments (including welds, pump casings, valve bodies and pressure-retaining bolting) and leakage tests of pressure retaining components.

The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives.

#### A.2.1.2 WATER CHEMISTRY

The Water Chemistry Program includes periodic monitoring and control of detrimental contaminants below the levels known to cause cracking, loss of material, or reduction of heat transfer. The primary scope of this program consists of the Reactor Coolant system and related auxiliary systems containing treated water, reactor coolant, treated borated water and steam. The program is based on Electric Power Research Institute (EPRI) PWR primary water chemistry guidelines and Pressurized Water Reactor (PWR) secondary water chemistry guidelines.

#### A.2.1.3 REACTOR HEAD CLOSURE STUDS

The Reactor Head Closure Studs Program conducts inspections of reactor vessel flange stud hole threads, reactor head closure studs, nuts, and washers to manage cracking and loss of material per the requirements of ASME, Boiler and Pressure Vessel Code, Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components.*"

#### A.2.1.4 BORIC ACID CORROSION

The Boric Acid Corrosion Program implements the recommendations of Nuclear Regulatory Commission (NRC) Generic Letter 88-05 "*Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,*" and manages aging of structures and components resulting from borated water leakage. The program requires periodic visual inspection of all systems within the scope of license renewal that contain borated water for evidence of leakage, accumulations of dried boric acid, or boric acid damage. The program provides for visual inspections and early discovery of borated water leaks such that mechanical, electrical, and structural components that may be contacted by leaking borated water will not be adversely affected or their intended functions impaired. The program identifies components exhibiting boric acid accumulations or leakage, evaluates the acceptability for continued service of components exhibiting boric acid accumulations or leakage, trends and tracks previously identified leaks or boric acid accumulations and provides corrective actions for the observed leakage sources and any other affected structures and components.

### A.2.1.5 NICKEL-ALLOY PENETRATION NOZZLES WELDED TO THE UPPER REACTOR VESSEL CLOSURE HEADS OF PRESSSURIZED WATER

The Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program manages the aging effect of cracking due to primary water stress corrosion cracking of the nickel-alloy used in the fabrication of the upper vessel head penetration nozzles. The NRC has approved ASME Code Case N-729-1, *"Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division"* with conditions in accordance with the requirements of 10 CFR 50.55a (g)(6)(ii)(D). Repair, replacement, and mitigation activities are conducted in accordance with the Seabrook Station ASME Section XI Repair/Replacement program.

### A.2.1.6 (NOT USED)

### A.2.1.7 PWR VESSEL INTERNALS

The PWR Vessel internals Program manages aging effects in reactor vessel internals components. It is based on the EPRI-MRP-227, "*Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*" and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program.

The aging management approach for PWR vessel internals consists of four major elements:

(1) component categorization and aging management strategy development;

(2) selection of aging management methodologies for PWR vessel internals that are both appropriate and based on an adequate level of applicable experience;

(3) qualification of the recommended methodologies that is based on adequate technical justification; and

(4) implementation of the recommendations based on the Industry Initiative for the Management of Materials Issues, NEI 03-08 "*Guideline for the Management of Materials Issues*".

### A.2.1.8 FLOW-ACCELERATED CORROSION

The Flow-Accelerated Corrosion (FAC) Program manages aging effects of loss of material due to wall thinning on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, tees, expanders, and valve bodies containing high energy fluids (both single phase and two phase flow). The program is based on the EPRI guidelines in NSAC-202L, "*Recommendations for an Effective Flow Accelerated Corrosion Program*" and uses the Chexal Horowitz Engineering/ Corrosion Workstation (CHECWORKS) software program as a predictive tool. Included in the FAC program are:

(a) an analysis to determine FAC susceptible lines,

(b) performance of baseline inspections,

(c) follow-up inspections to confirm the predictions

(d) repairing or replacing components, as necessary.

### A.2.1.9 BOLTING INTEGRITY

The Bolting Integrity Program manages the aging effects associated with bolting through the performance of periodic inspections for indications of cracking and loss of material and loss of preload. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication and assembly of bolted joints. The program follows the guidelines and recommendations delineated in NUREG-1339,"*Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*," EPRI NP-5769," *Degradation and Failure of Bolting in Nuclear Power Plants," and* EPRI TR-104213," *Bolted Joint Maintenance & Application Guide*" for comprehensive bolting maintenance.

The Bolting Integrity Program credits other aging management programs for the inspection of bolting. Operator rounds and system walkdowns will also identify joint leakage.

The Bolting Integrity Program credits six separate aging management programs for the inspection of bolting.

1. ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD provides the requirements for inservice inspection of ASME Class 1, 2, and 3 piping, which includes pressure retaining bolting.
2. ASME Section XI, Subsection IWE Program, includes steel containment shells and their integral attachments.
3. ASME Section XI, Subsection IWF Program, provides the requirements for inservice inspection of ASME Class 1, 2, and 3 component supports.
4. Buried Piping and Tanks Program, provides the requirements for the periodic visual inspections of corrosion on buried piping and tanks, including bolting.

5. External Surfaces Monitoring Program provides the requirements for the inspection of bolting for steel components such as piping, piping components, ducting and other components within the scope of license renewal.
6. Structures Monitoring Program provides the requirements for the inspection of structural support bolting.

### A.2.1.10 STEAM GENERATOR TUBE INTEGRITY

The Steam Generator Tube Integrity Program manages the aging effects of cracking, loss of material, reduction of heat transfer and wall thinning from flow accelerated corrosion of the Steam Generator components. The program is based on NEI 97-06 Rev. 2, *"Steam Generator Program Guidelines"*, the response and commitment to Generic Letter 97-06, *"Steam Generator Program Guidelines"*, and Seabrook Station Technical Specification 3/4.4.5 *"Steam Generators"* which ensure that the performance criteria for structural integrity, accident-induced leakage, and operational leakage are not exceeded. Seabrook Station has implemented the operational leakage limits found in NUREG-1431, *"Standard Technical Specifications for Westinghouse Pressurized Water Reactors"*.

### A.2.1.11 OPEN-CYCLE COOLING WATER SYSTEM

The Open-Cycle Cooling Water System Program manages the aging effects of hardening and loss of strength, loss of material, and reduction of heat transfer. This program relies on the implementation of the recommendations of NRC Generic Letter 89-13,"*Service Water System Problems Affecting Safety-Related Equipment*". The program manages aging effects for components in the circulating water, primary component cooling water, service water, and diesel generator systems.

### A.2.1.12 CLOSED-CYCLE COOLING WATER SYSTEM

The Closed-Cycle Cooling Water Program manages aging effects of cracking, loss of material and reduction of heat transfer in closed cycle cooling water systems. Closed-Cycle Cooling Water (CCCW) systems are described as systems not subject to significant sources of contamination, in which water chemistry is controlled and in which heat is not directly rejected to the ultimate heat sink. The program scope includes activities to manage aging in the Primary Component Cooling Water system and Emergency Diesel Generator Jacket Water cooling systems. The program also includes fire pump diesel engine glycol coolant system, the Control Building Air Handling glycol coolant system (safety-related), and the Thermal Barrier Cooling Water system.

The program includes maintenance of system corrosion inhibitor concentrations to minimize degradation and inspections of opportunity to assess management of component aging. The program is based on the EPRI "*Closed Cycle Cooling Water Chemistry Guidelines*".

### A.2.1.13 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program manages loss of material of structural components and wear on the rails of lifting systems within the scope of license renewal.

The program employs the use of visual inspections to identify aging effects prior to loss of function. Preventive actions are not associated with these activities.

Only the structural portions of the in-scope cranes and monorails are in the scope of this program.

### A.2.1.14 COMPRESSED AIR MONITORING

The Compressed Air Monitoring Program manages aging effects of hardening and loss of strength, loss of material, and reduction of heat transfer and assures an oil free, dry air environment in the plant compressed air system, Diesel Generator compressed air subsystem and containment compressed air system components.

Seabrook Station committed to maintain instrument air quality in accordance with the Quality Standard for Instrument Air, ISA-S7.3; *"Quality Standard for Instrument Air".* Compliance with ISA-S7.3 is verified by continuous monitoring or periodic testing. In-line dew point monitors are used to verify that the dew point of instrument air at the outlet of the instrument air system dryers is at or below a calculated limit. In-line filters are installed which limit air system maximum entrained particle size. These in-line filters meet or exceed the requirements of the quality standard. Periodic replacement of filters is part of the preventative maintenance program for instrument air systems. Air samples are obtained at least annually and tested for to ensure compliance with air quality standards.

### A.2.1.15 FIRE PROTECTION

The Fire Protection Program manages aging effects to the fire protection and suppression components through detailed inspections. Age-related degradation of the diesel-driven fire pump's fuel oil supply line is managed through regularly scheduled fire pump performance tests.

The Fire Protection Program includes but is not limited to inspections of fire barrier penetration seals, fire barrier walls, ceilings, floors, fire doors and testing to prove functionality of the diesel driven fire pump fuel oil supply line.

### A.2.1.16 FIRE WATER SYSTEM

The Fire Water System Program manages the aging effects of loss of material and reduction of heat transfer due to fouling of the Fire Water System components through detailed inspections via the Seabrook Station Surveillance Test Procedures.

The Fire Water System Program is established in accordance with the applicable National Fire Protection Association (NFPA) codes and standards.

The chemistry program provides methods and directions for adding various chemicals to various plant systems including the Fire Water Tanks. These chemicals prevent microbiological growth, inhibit scale formation, dispense solids contained in water, improve chlorination efficiency and maintain pH level to prevent corrosion of piping and components.

### A.2.1.17 ABOVEGROUND STEEL TANKS

The Aboveground Steel Tanks Program manages aging effects through preventive measures to mitigate corrosion and through periodic inspections to manage any effects of corrosion on aboveground steel tanks within the scope of License Renewal.

The program utilizes the application of protective coatings on the exterior surfaces of the in-scope steel tanks to mitigate corrosion development due to environmental factors. To ensure that the exterior surfaces of the tanks remain protected, the protective coatings are visually inspected.

Inaccessible locations, such as the tank bottom, will be surveyed by ultrasonic thickness measurements from inside the tank to detect any material degradation. The ultrasonic thickness measurements of fuel oil tanks within the scope of this program will be performed in accordance with the Fuel Oil Chemistry Program.

### A.2.1.18 FUEL OIL CHEMISTRY

The Fuel Oil Chemistry Program manages loss of material in the diesel fuel oil systems for the emergency diesel generators, diesel driven fire water pumps and the Auxiliary Boiler fuel oil system through monitoring and maintenance of diesel fuel oil quality.

New fuel oil is sampled and verified to meet the requirements of applicable American Society for Testing and Materials (ASTM) standards prior to offloading to the storage tanks. The program monitors fuel oil quality and the levels of water in the fuel oil which may cause the loss of material of the tank internal surfaces. The program monitors water and sediment contamination in diesel fuel.

Fuel Oil storage tanks are periodically drained and inspected. This inspection includes ultrasonic thickness measurements of the tank bottom surface to ensure that significant degradation has not occurred.

### A.2.1.19 REACTOR VESSEL SURVEILLANCE

The Reactor Vessel Surveillance Program manages the aging effect of loss of fracture toughness due to neutron embrittlement of the low alloy steel Reactor Vessel. The extent of reactor vessel embrittlement for upper-shelf energy and pressure temperature limits for 60 years is projected in accordance with the

NRC Regulatory Guide 1.99, "*Radiation Embrittlement of Reactor Vessel Materials*". The program utilizes the methodology of projecting neutron embrittlement using surveillance data. Monitoring methods are in accordance with 10 CFR 50, Appendix H "*Reactor Vessel Material Surveillance Requirements*". Testing methods are in accordance with ASTM E 185-82 "*Radiation Embrittlement of Reactor Vessel Materials*".

### A.2.1.20 ONE-TIME INSPECTION

The One-Time Inspection Program addresses potentially long incubation periods for certain aging effects and provides a means of verifying that an aging effect is either not occurring or is progressing so slowly as to have negligible effect on the intended function of the structure or components. The One-Time Inspection Program provides measures for verifying that an aging management program is not needed, for verifying the effectiveness of an existing program, or for determining that degradation is occurring which will require evaluation and corrective action.

The One-Time Inspection Program includes determination of appropriate inspection sample size, identification of inspection locations, selection of examination technique, specification of acceptance criteria, and evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample includes locations where the most severe aging effect(s) would be expected to occur. Inspection methods may include visual (or remote visual), surface or volumetric examinations, or other established NDE techniques.

This Program:

- Verifies the effectiveness of the Plant Chemistry Program for managing the effects of aging in portions of piping and components exposed to a treated water environment.
- Verifies the effectiveness of the Fuel Oil Chemistry Program for managing the effects of aging of piping and components in systems that contain fuel oil.
- Verifies the effectiveness of the Lubricating Oil Analysis Program for managing the effects of aging of piping and components in systems that contain lube oil.

### A.2.1.21 SELECTIVE LEACHING OF MATERIALS

The Selective Leaching of Materials Program manages the aging effects of loss of material in components susceptible to selective leaching that are exposed to raw water, brackish water, treated water (including closed cycle cooling), or groundwater environment.

The Selective Leaching of Materials Program will include a one-time examination of selected components that may be susceptible to selective leaching. Visual inspection and mechanical examination techniques (Brinell hardness testing or other mechanical examination techniques such as destructive testing (when appropriate), scraping, chipping or other types of hardness testing), or additional examination methods that become available to the nuclear industry, will be used to determine if selective leaching is occurring on the surfaces of a selected set of components.

### A.2.1.22 BURIED PIPING AND TANKS INSPECTION

The Buried Piping and Tanks Inspection Program manages loss of material from the external surfaces of buried steel (including cast iron) and stainless steel components. The plant has no buried steel tanks in scope for license renewal. The program includes preventive measures to mitigate corrosion and periodic inspections that manage the aging effects of corrosion on the pressure-retaining capacity of buried piping in the scope for license renewal.

The program includes provisions for visual inspections of the protective wraps and coatings on buried steel and stainless steel piping when the pipes are exposed during maintenance. If damage to the protective wraps or coatings is found, the outer surface of the pipe is inspected for loss of material due to general, pitting, crevice or microbiologically-influenced corrosion.

This program requires that at least one opportunistic or focused inspection be performed within the ten year period prior to entering the period of extended operation. Upon entering the period of extended operation a planned inspection will be performed within ten years, unless an opportunistic inspection has occurred within that ten year period.

### A.2.1.23 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL BORE-PIPING

The One-Time Inspection of ASME Code Class 1 Small Bore Piping Program applies to small-bore ASME Code Class 1 piping less than 4 inches nominal pipe size (NPS), including pipe, fittings, and branch connections. While the ASME Boiler and Pressure Vessel Code, Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components"*, does not require volumetric examination of Class 1 small-bore piping, the Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will be used to identify cracking by performing volumetric examinations of selected piping.

The inspection sample determination will include both socket welds and butt welds. If non-destructive volumetric inspection techniques have not been qualified, Seabrook will have the option to remove the weld for destructive examination.

### A.2.1.24 EXTERNAL SURFACES MONITORING

The External Surfaces Monitoring Program manages aging effects through visual inspection of external surfaces for evidence of hardening and loss of strength, reduction of heat transfer and loss of material (galvanic, general, crevice and pitting corrosion, and wear). This program consists of periodic inspections of aluminum, Cast Austenitic Stainless Steel (CASS), copper alloy, copper alloy >15% zinc, elastomer, galvanized steel, gray cast iron, nickel alloy, stainless steel and steel components such as piping, piping components, ducting, pipe supports and other components to manage aging effects.

The External Surfaces Monitoring Program utilizes periodic plant system inspections and walkdowns to monitor for materials degradation and leakage. This program inspects components such as piping, piping components, ducting and other components, including bolting. Coatings deterioration is monitored as an indication of possible underlying degradation.

### A.2.1.25 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program manages the aging effects of cracking, loss of material, fouling, reduction of heat transfer and hardening and loss of strength. This program consists of inspections of the internal surfaces of aluminum, CASS, copper alloy, copper alloy >15% zinc, elastomer, galvanized steel, gray cast iron, nickel alloy, stainless steel, and steel piping, piping components, ducting and other components that are not covered by other aging management programs.

The program inspections are inspections of opportunity, performed during pre-planned periodic system and component surveillances or during maintenance activities when the systems are opened and the surfaces made accessible for visual inspection. This maintenance may occur during power operations or refueling outages when many systems are opened. The visual inspections assure that existing environmental conditions are not causing material degradation that could result in a loss of the component intended function.

### A.2.1.26 LUBRICATING OIL ANALYSIS

The Lubricating Oil Analysis Program obtains and analyzes lubricating oil samples from plant equipment to ensure that the oil quality is maintained within established limits. The program provides an early indication of adverse equipment condition in lubricating oil environments.

The Seabrook Station Lubricating Oil Analysis Program includes sampling and analysis of lubricating oil for components within the scope of license renewal and subject to aging management review, that are exposed to lubricating oil and for which pressure boundary integrity or heat transfer is required for the component to perform its intended function.

### A.2.1.27 ASME SECTION XI, SUBSECTION IWE

The ASME Section XI, Subsection IWE Program manages aging effects to the containment liner, electrical penetrations, mechanical penetrations (piping, ventilation, and spares), personnel lock, equipment hatch, recirculation sump, reactor pit, moisture barriers, seals, gaskets, and supports.

The program performs inspections using the same primary Inservice Inspection method as specified in ASME Section XI, Subsection IWE "*Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants* "; visual examination (general visual, VT-3, VT-1).

### A.2.1.28 ASME SECTION XI, SUBSECTION IWL

The ASME Section XI, Subsection IWL, Inservice Inspection Program manages aging effects to the steel reinforced concrete for the containment building and complies with the requirement of examination requirements of 10 CFR 50.55a in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL "*Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants*".

The primary inspection methods used are VT-1C visual examination, VT-3C visual examination and alternative examination methods (in accordance with IWA-2240). All accessible containment reinforced concrete components are within the scope of this program.

### A.2.1.29 ASME SECTION XI, SUBSECTION IWF

The ASME Section XI, Subsection IWF "*Requirements for Class 1,2,3, and MC Component Supports of Light-Water Cooled Power Plants*," Inservice Inspection Program (ISI) provides inspections of Class 1, 2, and 3 Component Supports. For supports other than piping supports, the supports of only one component of a group having similar design, function, and service must be examined. Supports of piping and other items exempted from volumetric or surface examination are also exempt.

The program uses VT-3 visual examination for detection of degradation. The performance requirements for VT-3 examination are conducted to determine the general mechanical and structural condition of components and their supports.

### A.2.1.30 10 CFR 50, APPENDIX J

The 10 CFR Part 50, Appendix J Program implements Title 10 Code of Federal Regulations Part 50 Appendix J, "*Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*" Option B. The test requirements of Appendix J provide for periodic verification by tests of the leak-tight integrity of the primary reactor containment. The purposes of the tests are to assure that 1) leakage through the containment or systems and components penetrating the containment does not exceed the allowable leakage rate specified in the Technical Specifications and Updated Final Safety Analysis Report, and 2) integrity of the containment structure is maintained during its service life.

### A.2.1.31 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program includes the Masonry Wall Program and the Inspection of Water Control Structures Associated with Nuclear Power Plants Program.

The Structures Monitoring Program is implemented through the plant Maintenance Rule Program, which is based on the guidance provided in NRC Regulatory Guide 1.160 "*Monitoring the Effectiveness of Maintenance at Nuclear power Plants*" and NUMARC 93-01 "*Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*". The Structures Monitoring Program was developed using the guidance of these two documents. The Program is implemented to monitor the condition of structures and structural components within the scope of the Maintenance Rule, such that there is no loss of structure or structural component intended function.

### A.2.1.32 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging of accessible non-EQ cables and connections. Accessible cables and connections located in adverse localized environments shall be visually inspected for indications of accelerated insulation aging such as embrittlement, discoloration, cracking, swelling, or surface contamination. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable or connection. Accessible cables and connections shall be inspected prior to entering the period of extended operation, and at least once every 10 years thereafter.

### A.2.1.33 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program manages the aging of in-scope cables and connections. The program applies to sensitive instrumentation cable and connection circuits with low-level signals that are in scope for license renewal and are located in areas where the cables and connections could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents.

The program shall perform insulation resistance tests on the in-core neutron flux monitoring cable and connections in the Nuclear Instrumentation System.

The frequency of the tests on these cables shall be based on engineering evaluation, but the test frequency shall be at least once every ten years. The first test shall be performed prior to entering the period of extended operation.

### A.2.1.34 INACCESSIBLE MEDIUM VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging of inaccessible medium voltage cables exposed to adverse localized environments caused by significant moisture while energized.

The program includes the following two components:

- **Periodic Inspections Of Manholes Containing In-Scope Medium Voltage Cables**

  In-scope manholes shall be periodically inspected for water collection. Water found in the manholes shall be drained.

  The frequency of manhole inspections shall be adjusted based on inspection results. However, the maximum time between inspections shall be no more than two years. The first inspections shall be performed prior to entering the period of extended operation.

- **Testing Of In-Scope Inaccessible Medium Voltage Cables**

  The specific type of test performed shall be determined prior to the initial test, and shall be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI guidelines for "*Effects of Moisture on the Life of Power Plant Cables*" or other testing that is state-of-the-art at the time the test is performed. Cable testing shall be performed prior to entering the period of extended operation and at least once every 10 years thereafter.

### A.2.1.35 METAL ENCLOSED BUS

The Metal Enclosed Bus Program manages the aging of in-scope metal enclosed buses. The internal portions of the in-scope metal enclosed bus enclosures are inspected for cracks, corrosion, foreign debris, excessive dust buildup and evidence of moisture intrusion. The bus insulation is visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The internal bus supports are visually inspected for structural integrity and signs of cracks. Accessible bolted bus connections are checked for looseness using thermography from outside the metal enclosed bus.

The inspections and tests shall be performed prior to entering the period of extended operation and at least once every 10 years thereafter.

Aging management of the Metal Enclosed Bus enclosures and elastomers is included in the Structures Monitoring Program.

### A.2.1.36 FUSE HOLDERS

The Fuse Holders Program manages the aging of in-scope metallic clamps of fuse holders.

The Fuse Holders Program performs tests on in-scope fuse holders (metallic clamps). The type of test is a proven test, such as thermography or contact resistance which detects thermal fatigue in the form of high resistance caused by corrosion or oxidation. The type of test performed is determined prior to the initial test. The first test shall be performed prior to entering the period of extended operation and at least once every 10 years thereafter.

### A.2.1.37 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a one-time testing program that shall be used to verify the absence of aging effects on the metallic portion of electrical cable connections. The aging effect and mechanism of concern is the loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation.

The scope of this sampling program considers application (medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc).

The specific type of test performed is a proven test for detecting loose connections, such as thermography or contact resistance measurement, as appropriate for the application.

The one-time test shall be performed prior to entering the period of extended operation.

## A.2.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

### A.2.2.1 345 kV $SF_6$ BUS

The 345kV $SF_6$ Bus Program manages the aging that could lead to loss of pressure boundary due to elastomer degradation, loss of material due to corrosion and loss of function due to unacceptable air, moisture or $SO_2$ levels. Sulfur Hexafluoride ($SF_6$) is an inert gas which is used to insulate the bus conductor.

The program inspects for corrosion on the exterior of the bus duct housing, tests for leaks and tests gas samples to determine air, moisture and $SO_2$ levels.

The presence of air or moisture may lead to the loss of intended function. $SO_2$ levels are an indication of partial discharge internal to the bus.

The tests and inspections shall be performed prior to entering the period of extended of operation and at least once every six months thereafter.

### A.2.2.2 BORAL MONITORING

The Boral Monitoring Program assures the Boral neutron absorbers in the spent fuel racks maintain the validity of the criticality analysis in support of the rack design. The program relies on representative coupon samples mounted in a coupon "train" located in the spent fuel pool to monitor performance of the absorber material without disrupting the integrity of the storage system. Coupon samples are removed from the spent fuel pool on a prescribed schedule and physical, chemical and neutronic absorptive properties are measured. From these data, the physical condition and neutron-absorbing capacity of the Boral in the storage cells are assessed.

### A.2.2.3 NICKEL-ALLOY NOZZLES AND PENETRATIONS

The Nickel-Alloy Nozzles and Penetrations Program manages the aging effect of cracking due to primary water stress corrosion cracking (PWSCC) of nickel-alloy pressure boundary and structural components exposed to primary coolant.

The Nickel-Alloy Nozzles and Penetrations Program ranked the Alloy 600/82/182 locations based on four main criteria: PWSCC susceptibility (e.g., operational time and temperature), failure consequence, leakage detection margin, and radiation dose rates. Additionally, material heat susceptibility and other industry experience were considered.

The program incorporates the inspection schedules and frequencies for the nickel-alloy components in accordance with the plant ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program and, where applicable, ASME Code Case N-722 "*Additional Examinations for PWR [pressurized water reactor (PWR)] Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1*," subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(E).

The program complies with applicable NRC Orders, and implements applicable NRC Bulletins, Generic Letters, and staff-accepted industry guidelines.

## A.2.3 NUREG-1801 CHAPTER X AGING MANAGEMENT PROGRAMS

### A.2.3.1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

The Metal Fatigue of Reactor Pressure Boundary Program is a preventive program that monitors and tracks the number of critical thermal and pressure transients to verify that the Cumulative Usage Fatigue (CUF) for reactor coolant system components remain less than 1.0 through the period of extended operation. The program determines the number of transients that occur and updates 60-year projections on a periodic basis.

The program is credited with monitoring reactor coolant system design transients. Cumulative Usage Fatigue of the Reactor Vessel, the pressurizer, the Steam Generators, Class 1 and non-Class 1 piping, and Class 1 components subject to the reactor coolant, treated borated water, and treated water environments. The program will use fatigue monitoring software to monitor the number of cycles a system or components endure. Pre-established limits will identify components approaching design limits. Components approaching design limits will be reanalyzed, repaired, replaced or inspected in accordance with applicable design codes.

### A.2.3.2 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS

The Environmental Qualification (EQ) of Electric Components Program provides a summary of the components that are managed for EQ aging. The program meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics and the environmental conditions to which the components could be subjected. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires replacement or refurbishment of components not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in NUREG-0588, "*Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment*," July 1981, and RG 1.89, Rev. 1, "*Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants, June 1984*". Seabrook Station conforms to NUREG-0588 Category 1 requirements.

Seabrook Station's compliance with 10 CFR 50.49 ensures that the component can perform its intended functions during accident conditions after experiencing the effects of in-service aging. The EQ Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for electrical components in the EQ Program that specify a qualification of at least 40 years are TLAAs for license renewal because the criteria contained in 10 CFR 54.3 are met.

Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met).

TLAA disposition option 10 CFR 54.2(c)(1)(iii), which states that the effects of aging will be adequately managed for the period of extended operation, is chosen and the EQ Program will manage the aging effects of the components.

## A.2.4 TIME-LIMITED AGING ANALYSES

### A.2.4.1 NEUTRON EMBRITTLEMENT OF THE REACTOR VESSEL

The current license period reactor vessel embrittlement analyses that evaluate reduction of fracture toughness of the Seabrook Station reactor vessel beltline materials are based on a 40-year End-of-Life (EOL) fluence values. The analyses associated with neutron embrittlement of reactor vessel materials due to neutron irradiation are Time-Limited Aging Analyses (TLAAs) as defined by 10 CFR 54.21(c) and must be evaluated for the increased neutron fluence associated with 60 years of operation.

The following Seabrook Station analyses are TLAAs that address the effects of neutron irradiation on the reactor vessel.

- Neutron Fluence
- Upper-Shelf Energy (USE)
- Pressurized Thermal Shock
- Pressure-Temperature (P-T) Limits

#### A.2.4.1.1 Neutron Fluence Analyses

The neutron fluence analysis is a TLAA as defined by 10 CFR 54.21(c) and must be evaluated for the increased neutron fluence associated with 60 years of operation. These neutron fluence projections are used as input to the analyses for fracture toughness, or Upper Shelf Energy (USE), Pressurized Thermal Shock (PTS) limits, Reverence Temperatures – Nil Ductility Transition ($RT_{NDT}$), Adjusted Reference Temperatures (ART), Low-Temperature Overpressure Protection (LTOP) limits, and Reactor Vessel Pressure-Temperature Limit (P-T limit) curves.

#### A.2.4.1.2 Upper Shelf Analyses

The current Charpy Upper Shelf Energy (USE) analyses were prepared for the reactor vessel beltline materials for Seabrook Station based upon projected neutron fluence values for 40 years of service. These are TLAAs requiring evaluation using the projected 60-year fluence values.

The Seabrook Station analyses have been projected to the end of the period of extended operation for reactor vessel materials with projected fluence exceeding $1 \times 10^{17}$ n/cm$^2$ (MeV > 1.0). The USE values for the beltline and extended beltline materials are projected to remain above the 50 ft-lb requirement through the period of extended operation for Seabrook Station in accordance with 10 CFR 54.21(c)(1)(ii).

### A.2.4.1.3 Pressurized Thermal Shock

10 CFR 50.61(b)(1) provides rules for the protection of pressurized water reactors against pressurized thermal shock. Licensees are required to assess the projected values of nil-ductility reference temperature whenever a significant change occurs in the projected values of Reference Temperature – Pressurized Thermal Shock ($RT_{PTS}$), or upon request for a change in the expiration date for the facility operating license. The current $RT_{PTS}$ analyses, evaluated for 32 Effective Full Power Years (EFPY) fluence values predicted for 40 years of operation, are TLAAs requiring evaluation for 60 years.

The margin is the difference between the maximum nil-ductility reference temperature ($RT_{PTS}$) in the limiting beltline material and the screening criteria established in accordance with 10 CFR 50.61(b)(2). The screening criteria for the limiting reactor vessel materials are 270°F for beltline plates, forgings, and axial weld materials, and 300°F for beltline circumferential weld materials. If the calculated value reference temperature is less than the specified screening criterion, then the vessel is acceptable with respect to reactor vessel during postulated transients during the period of extended operation.

The $RT_{PTS}$ analyses have been projected to the end of the period of extended operation and are shown to be within the maximum allowable PTS screening criteria limits in accordance with 10 CFR 54.21(c)(1)(ii).

### A.2.4.1.4 Reactor Vessel Pressure-Temperature Limits, Including Low Temperature Overpressure Protection Limits

Title 10 CFR Part 50, Appendix G requires that the reactor pressure vessel be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. The P-T limits must account for the anticipated reactor vessel fluence. The current minimum Low Temperature Overpressure Protection (LTOP) setpoint for Seabrook Station is 561 psig.

The current Seabrook Station P-T and LTOP limit calculations are effective through 20 EFPY. Heatup and cooldown P-T limit curves for 55 EFPY will be prepared using the most limiting value of $RT_{NDT}$ (reference nil ductility transition temperature) corresponding to the limiting material in the beltline region of the reactor vessel. This is determined by using the unirradiated reactor vessel material fracture toughness properties adjusted to account for the estimated irradiation-induced shift in the Reference Temperature – Nil Ductility Transition ($\Delta RT_{NDT}$).

The P-T and LTOP limit analyses will not be submitted at this time. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54(c)(1)(iii). Seabrook Station will submit updates to the P-T curves and LTOP limits to the NRC at the appropriate time to comply with 10 CFR 50 Appendix G.

## A.2.4.2 METAL FATIGUE OF VESSELS AND PIPING

Metal fatigue was evaluated in the design process for Seabrook Station pressure boundary components, including the reactor vessel, reactor coolant pumps, steam generators, pressurizer, piping, valves, and components of primary, secondary, auxiliary, steam, and other systems. The current design analyses for these components have been determined to be Time-Limited Aging Analyses (TLAAs) requiring evaluation for the period of extended operation. This section is divided into seven subsections that each addresses a specific grouping of components that were analyzed in accordance with the same design requirements.

These groupings are as follows:

- Nuclear Steam Supply System (NSSS) Pressure Vessel and Component Fatigue Analyses
- Supplementary ASME Section III, Class 1 Piping and Component Fatigue Analyses
- Reactor Vessel Internals Fatigue Analyses
- Environmentally-Assisted Fatigue Analyses
- Steam Generator Tube, Loss of Material and Fatigue Usage from Flow-induced Vibration
- Absence of TLAAs For Fatigue Crack Growth, Fracture Mechanics Stability, or Corrosion Analyses Supporting Repair of Alloy 600 Materials
- Non-class I Component Failure Analysis

### A.2.4.2.1 Nuclear Steam Supply System (NSSS) Pressure Vessel and Component Fatigue Analyses

Nuclear Steam Supply System (NSSS) pressure vessels and components for Seabrook were designed in accordance with ASME Section III, Class 1 requirements and were required to have explicit analyses of cumulative usage fatigue. The major components are the Reactor Vessel, Vessel Closure Head, Steam Generators and the Reactor Coolant Pump Casings. The applicable design codes for these components have been identified.

In order to determine if the ASME Section III, Class 1 fatigue analyses will remain valid for 60 years of service, a review of fatigue monitoring data was performed to determine the number of cumulative cycles of each transient type that have occurred during past plant operations. Then the average rate of occurrence was determined, and predictions of future transient occurrences were made. For each transient type, the 60-year projected number of occurrences was determined by adding the number of past occurrences to the number of predicted future occurrences. These 60-year projections were then compared to the numbers of design cycles used in the fatigue analyses to determine if the design cycles remain bounding for 60 years of operations. If the 60-year projected numbers of cycles is less than the numbers of cycles used in the design fatigue analyses, then the fatigue analyses based upon the design transients will remain valid for 60 years of operation if the design transient severity is also bounding of the actual transient severity.

The 40-year design transients bound the numbers of cycles projected to occur during 60 years of plant operations at Seabrook. Therefore, the NSSS Class 1 fatigue analyses that are based upon the 40-year design transients remain valid for the period of extended operation.

#### A.2.4.2.2 Supplementary ASME Section III, Class 1 Piping and Component Fatigue Analyses

In addition to the original design assumptions, the Seabrook Pressurizer fatigue evaluations were updated to include the added thermal stratification effects of insurge and outsurge events on the pressurizer lower head and surge nozzle.

Each of the Seabrook Station piping systems, including the Reactor Coolant System main loop piping, were originally designed in accordance with ASME Section III 1971 Edition with addenda through Winter 1972 . Since then, a number of updated fatigue analyses have been prepared for piping systems and components to address transients that have been identified in the industry that were not originally considered. These analyses have been performed in accordance with ASME Section III, Class 1 rules to enable these transients to be thoroughly evaluated. These transients included thermal stratification of the pressurizer surge line, as described in NRC Bulletin 88-11.

These analyses are separated from those evaluated in the previous sections because the transient definitions have been modified, or additional transients have been postulated for these components, in addition to those previously described. Therefore, the cycle projections for these components must address these revised transients or additional transient types to determine if they also remain bounded for 60 years of service.

##### A.2.4.2.2.1 NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification

NRC Bulletin 88-11, issued in December 1988, requested utilities to establish and implement a program to confirm the integrity of the pressurizer surge line. The program required both visual inspection of the surge line and demonstration that the design requirements of the surge line are satisfied, including the consideration of stratification effects.

The Pressurizer Surge Line piping and nozzles were previously evaluated for the effects of thermal stratification and plant-specific transients (1990) and determined that the surge line will remain within the ASME Code requirements for the design life of the unit. The controlling fatigue location was the surge line hot leg nozzle safe-end. In later evaluations, plant-specific ASME Section III, Class 1 evaluations were performed for the surge line hot leg nozzle and pressurizer surge nozzle.

The hot leg surge line nozzle was evaluated for the effects of pressurizer insurge and outsurge transients and surge line stratification. Projected 60-year cycles of surge line stratification and insurge and outsurge transients were used when these were greater than previously evaluated design cycles. These evaluations resulted in CUF less than 1.0 at the hot let surge line nozzle.

The pressurizer surge nozzle was evaluated for the effects of pressurizer insurge and outsurge transients and surge line stratification. The pressurizer surge nozzle has been evaluated using an ASME Section III, Class 1 fatigue analysis. This analysis was part of the evaluation of the structural weld overlay applied to the pressurizer surge nozzle. This evaluation resulted in a Cumulative Usage Factor (CUF) less than 1.0 at the pressurizer surge nozzle.

The analyses remain valid for the period of extended operation for the Pressurizer Surge Line, Pressurizer Surge Nozzle and Surge Line Hot Leg Nozzle in accordance with 10 CFR 54.21(c)(1)(i).

#### A.2.4.2.2.2 Reactor Vessel Internals Aging Management

The Seabrook Reactor Vessel Internals were designed and constructed prior to the development of ASME Code requirements for core support structures. Demonstration that the effects of aging degradation are adequately managed is essential for assuring continued functionality of the reactor internals during the desired plant operating period, including license renewal. The EPRI Materials Reliability Program (MRP) Reactor Internals Inspection & Evaluation (I&E) Guidelines (MRP-227) inspection requirements will manage aging effects.

In accordance with 10 CFR 54.21(c), the Aging Management Program for reactor vessel internals will provide assurance that the effects of aging will be adequately managed for the period of extended operation per 10 CFR 54.21(c)(1)(iii).

#### A.2.4.2.3 Environmentally-Assisted Fatigue Analyses

NUREG-1801, Revision 1, "*Generic Aging Lessons Learned*", contains recommendations on specific areas for which existing programs should be augmented for license renewal. The program description for Aging Management Program X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary Program, provides guidance for addressing environmental fatigue for license renewal. It states that an acceptable program addresses the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant. Examples of these components are identified in NUREG/CR-6260, "*Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*".

This sample of components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses using formulae contained in NUREG/CR-6583, *"Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels"* and in NUREG/CR-

5704, *"Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels"*. Demonstrating that these components have an environmentally adjusted cumulative usage factor less than or equal to the design limit of 1.0 is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary.

NUREG/CR-6260 provided environmental fatigue calculations for a newer vintage Westinghouse plant using the interim fatigue curves from NUREG/CR-5999 for the locations of highest design CUF for the components listed below:

1. Reactor Vessel Shell and Lower Head
2. Reactor Vessel Inlet and Outlet Nozzles
3. Pressurizer Surge Line
4. Charging Nozzle
5. Safety Injection Nozzle
6. Residual Heat Removal System Class 1 Piping

For the NUREG/CR-6260 locations identified above, the plant-specific components were identified and the design ASME fatigue usage factors were adjusted by the environmentally-assisted fatigue penalty factors ($F_{en}$) to obtain the environmentally-assisted fatigue (EAF) result.

All locations were shown to achieve air-curve cumulative usage factors less than 1.0 for the 60 years of service. The evaluation of environmental fatigue effects for the Reactor Vessel Shell and Lower Head and Reactor Vessel Inlet and Outlet Nozzles determined that the CUF will remain below the ASME code allowable fatigue limit of 1.0 using the maximum applicable Fen, applied to CUF based on the design number of transients for these locations, when extended to 60 years. The remainder of these locations, Reactor Coolant System (RCS) Pressurizer Surge Line Nozzle, RCS Charging Nozzle, RCS Safety Injection Nozzle, and RCS Residual Heat Removal System Class 1 Piping, were analyzed in accordance with ASME Code Section III, Subarticle NB-3200 using all six stress components. The evaluations show that EAFs exceed 1.0 for 60-years of service for the hot leg surge line nozzle and charging nozzle. These analyses were based on Seabrook Station specific conditions and these locations will be monitored for fatigue usage including environmental effects by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1. Specifically, this program will monitor critical transients to verify cycle limits are maintained below limits specified in the UFSAR. Pre-established action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor, including environmental effects, exceeds the AMSE Code limit of 1.0. At least 2 years prior to entering the period of extended operation, Seabrook Station will implement the following aging management program for the plant-specific locations listed in NUREG/CR-6260 for the newer vintage Westinghouse plants.

(1) Consistent with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1 Seabrook Station will update the fatigue usage calculations using refined fatigue analyses, if necessary, to determine acceptable CUFs (i.e., less than 1.0) when accounting for the effects of the reactor water environment. This includes applying the appropriate $F_{en}$ factors to valid CUFs determined from an existing fatigue analysis valid for the period of extended operation or from an analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). Formulas for calculating the environmental life correction factors for carbon and low alloy steels are contained in NUREG/CR-6583 and those for austenitic stainless steels are contained in NUREG/CR-5704. NUREG/CR-6909 includes alternate formulas for calculating environmental life correction factors, in addition to updated fatigue design curves.

(2) If acceptable CUFs cannot be demonstrated for all the selected locations, then additional plant-specific locations will be evaluated. For the additional plant-specific locations, if CUF, including environmental effects is greater than 1.0, then Corrective Actions will be initiated, in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1. Corrective Actions will include inspection, repair, or replacement of the affected locations before exceeding a CUF of 1.0 or the effects of fatigue will be managed by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Therefore, the effects of the reactor coolant environment on fatigue usage factors in the remaining locations will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

### A.2.4.2.4. Steam Generator Tube, Loss of Material and Fatigue Usage from Flow-Induced Vibration

The Seabrook Station Model F steam generators were evaluated with respect to flow induced vibration (tube wear and fatigue usage) for the power increases that were implemented as part of the Seabrook Station Power Uprates. The analysis of the effects of steam generator flow-induced vibration on tube wear and fatigue usage assumed 40 years of operation.

Low-cycle fatigue usage for the most limiting tube in the most limiting power-uprated operating condition resulting from the flow-induced vibration tube bending stress is 0.2 ksi. This value is well below the fatigue endurance limit of 20 ksi at 1E+11 cycles, resulting in a computed fatigue usage of 0.0. High-cycle fatigue usage of U-bend tubes was evaluated. One of the prerequisites for high-cycle U-bend fatigue is a dented support condition at the upper plate. Seabrook Station steam generator tube support plates are manufactured from stainless steel therefore there is no potential for the necessary conditions to occur. It was concluded that the support condition leading to a dented support condition necessary for high-cycle fatigue cannot occur in the Seabrook Station Model F steam generators.

### A.2.4.2.5 Non-Class 1 Component Fatigue Analyses

This section describes fatigue-related TLAAs arising within design analyses of the Non-Class 1 piping and components. These piping and tubing components can be designed in accordance with ASME Section III Class 2 and 3.

The following non-Class 1 Seabrook Station systems that are in scope for license renewal were designed in accordance with ASME Section III Class 2 and 3, requirements: Reactor Coolant System (including primary loop piping and pressurizer surge line piping), Chemical and Volume Control System, Safety Injection System, Primary Component Cooling Water, Service Water, Sample System, Residual Heat Removal System, Main Steam System, Main Condensate and Feedwater, and the Steam Generator Blowdown System.

In order to evaluate these TLAAs for 60 years, the number of cycles expected to occur within the 60-year operational period should be compared to the numbers of cycles that were originally considered in the design of these components. If this number does not exceed 7,000 cycles, the minimum number of cycles required that would result in reduction of the allowable stress range, then there is no impact from the added years of service and the original analyses remain valid. If the total number of cycles exceeds 7,000 cycles, then additional evaluation is required.

The 60-year transient projection results for Seabrook show that even if all of the projected operational transients are added together, the total number of cycles projected for 60 years will not exceed 7,000 cycles. Therefore, there is no impact upon the implicit fatigue analyses used in the component design for the systems designed to ASME Section III Class 2 and 3, requirements.

The Sample System thermal cycles do not trend along with operational cycles because sampling is required on a periodic basis, as opposed to an operational basis. However, only the portion of the sampling lines that constitutes piping need be considered here. In this case that portion turns out to be a very short section of piping directly connected to the Reactor Coolant System (RCS) loop piping. Since this section of piping has no isolation valve and no bends, it is assumed to always be exposed to primary loop temperature and pressure condition. Similarly since there are no other external piping connections (only the tubing connection exits), the line will not experience any other externally applied loads. Therefore, that section of the sampling line that constitutes ASME Section III Class 2 and will only experience the RCS loop transients which have already been shown to be less than 7,000 cycles and the line is, therefore, acceptable.

The analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

### A.2.4.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS

In accordance with 10CFR50.49, all electrical equipment important to safety located in a harsh environment and required to function in that environment must be environmentally qualified. In order for a component to have sufficient design margin to perform its important to safety function under harsh environment conditions, the component may need to be periodically rebuilt or replaced. For these EQ components, the EQ program insures that they are rebuilt, replaced or reevaluated at the necessary interval. All qualified lives within the scope of the EQ program are managed under the EQ Program.

The Seabrook Station Environmental Qualification (EQ) of Electric Components Program implements aging management activities which are credited for the management of aging in selected components within the scope of 10 CFR 54.

### A.2.4.4 FATIGUE OF THE CONTAINMENT LINER AND PENETRATIONS

The original design analysis for the Seabrook Station containment liner plate determined that all of the criteria specified in ASME Section III Article NE-3221.5(d) required for exemption from the requirement to perform a cyclic operation analysis were met. To address these 40-year cycles during the period of extended operation, a re-evaluation of the six fatigue exemption requirements utilizing anticipated 60-year stress cycles was performed. The result of this analysis determined that the specified conditions through the period of extended operation continue to satisfy the requirement for exemption from analysis for cyclic operation in accordance with in ASME Section III Article NE-3221.5(d). The analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

### A.2.4.5 OTHER PLANT-SPECIFIC TLAAS

#### A.2.4.5.1 Reactor Coolant Pump Flywheel Fatigue Crack Growth Analyses

Westinghouse Report WCAP-14535-A, Rev. 0, "*Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination*" includes a fatigue crack growth analysis that has been identified as a TLAA. The report was submitted for NRC review and the NRC issued a Safety Evaluation Report in September 1996. The purpose of the report was to provide an engineering basis for elimination of reactor coolant pump (RCP) flywheel inservice inspection requirements for all operating Westinghouse plants and certain Babcock and Wilcox plants. The number of cycles (pump starts and stops) used in this report was 6,000 for a 60-year plant life. Crack growth was shown to be negligible from exposure to these 6,000 cycles.

Based on the current cycle count projected to 60 years, the projected cycle count is much less than the analyzed cycle counts of 6,000 cycles. The reactor coolant pump flywheel analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

#### A.2.4.5.2 Leak-Before-Break Analyses

Title 10 Code of Federal Regulations Part 50 Appendix A, Criterion 4 allows for the use of leak-before-break (LBB) methodology for excluding the dynamic effects of postulated ruptures in reactor coolant system piping. The fundamental premise of the LBB methodology is that the materials used in nuclear power plant piping are sufficiently tough that even a large through-wall crack would remain stable and would not result in a double-ended pipe rupture. Application of the LBB methodology is limited to those high-energy fluid systems not considered to be overly susceptible to failure from such mechanisms as corrosion, water hammer, fatigue, thermal aging or indirectly from such causes as missile damage or the failure of nearby components. The analyses involved with LBB are considered TLAAs.

Based on loading, pipe geometry, and fracture toughness considerations, enveloping governing locations were determined at which LBB crack stability evaluations were made. Through-wall flaw sizes were found which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the reactor coolant system primary loop piping. The thermal transients used in the fatigue crack growth analysis were the design transients listed in the NSSS Design Limits for 40 years at Seabrook Station. The corresponding 60-year projected cycles are lower than the 40-year design values. Therefore, the numbers of design cycles assumed in the analysis bound the numbers of design cycles projected for the period of extended operation.

The analyses remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

#### A.2.4.5.3 High Energy Line Break Postulation Based on Cumulative Usage Factor

The Seabrook Station High Energy Line Break (HELB) analysis used a screening criterion of CUF greater than 0.1 to identify areas of investigation. The Seabrook Station Updated Final Safety Analysis Report (UFSAR) Section 3.6(B).2.1(a) provided a basis to eliminate locations in each piping run or branch run from further consideration as high energy line break locations on the basis of low fatigue including intermediate location when the CUF was less than 0.1.

Selection of pipe failure locations for evaluation of the consequences on nearby essential systems, components, and structures, except for the reactor coolant loop, is in accordance with Regulatory Guide 1.46, and NRC Branch Technical Positions ASB 3-1 and MEB 3-1. A revised stress analysis also permitted omission of the surge line intermediate breaks. A leak-before-break (LBB) analysis eliminated large breaks in the main reactor coolant loops.

The surge line intermediate break locations were eliminated based on usage factor. The most recent piping analysis confirmed the elimination of these break locations. The analysis that justified the elimination of these intermediate locations in the surge line is therefore a TLAA.

Since the 60 year projected cycles are bounded by the original design cycles, the present intermediate locations with CUF less than 0.1 remain valid for the period of extended operation.

The analyses remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

### A.2.4.5.4 Fuel Transfer Tube Bellows Design Cycles

The fuel transfer tube assembly connects the fuel transfer canal (inside the containment structure) to the transfer pool (inside the spent fuel handling building). The fuel transfer tube assembly passes through the containment wall and through the exterior wall of the spent fuel handling building. The fuel transfer tube assembly is comprised of a 24-inch diameter penetration sleeve penetrating through the containment and spent fuel building walls and three (3) sets of expansion joints (bellows). The penetration sleeve and the three bellows perform a water-retaining intended function, and are within the scope of license renewal.

The fatigue analysis for each of the three bellows is based on the consideration of 20 occurrences of the Operating Basis Earthquake, each occurrence having 20 cycles of maximum response therefore, this design analysis is a TLAA requiring evaluation for the period of extended operation.

It is projected that 1 OBE would occur for Seabrook Station in 60 years of operation. Since the number of occurrences projected for 60 years is below the design limit of 5 occurrences of 10 cycles the design analysis remains valid for the period of extended operation.

The analyses remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

### A.2.4.5.5 Crane Load Cycle Limits

#### A.2.4.5.5.1 Polar Gantry Crane

The design specification for the 420/50-ton Polar Crane in the containment structure at Seabrook Station required that the crane conform to the design requirements of Crane Manufacturers Association of America (CMAA)

Specification 70, "*Specifications for Electric Overhead Traveling Cranes*". Service requirements specified for the design of this crane correspond to the cyclic loading requirements of CMAA 70, Class A. This evaluation of cycles over the 40 year life is the basis of a safety determination and is, therefore, a TLAA.

The estimated number of lifts for the Polar Crane over the remaining 40 years of service (which includes 20 years of Extended Operation) is 19,440 with most of the lifts being less than 2500 pounds. This rate is based on refueling outage use, therefore, the first 20 years of service life for the Polar Crane would include approximately 10,000 load cycles. Thus, the total service life load cycles will be approximately 30,000. Since the total number of lifts is less than the allowable design value of up to 100,000 cycles, the Polar Crane load cycle fatigue analyses for Seabrook Station remains valid for 60 years of plant operation.

The analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

### A.2.4.5.5.2 Cask Handling Crane

The original Seabrook Station Cask Handling Crane was replaced in 2008 by a single failure-proof crane rated for 130 tons (main hoist) and 5 tons (each of two auxiliary hoists). To meet single failure criteria, each of these cranes was designed to the requirement of ASME NOG-1-2004, NUREG-0554, and NUREG-0612. The cranes were also designed to Crane Manufacturers Association of America (CMAA) Specification 70, "*Specifications for Electric Overhead Traveling Cranes*", with an allowable design life cycle range of up to 100,000 cycles. This evaluation of cycles over the projected 40-year life is the basis of a safety determination and has been identified as a TLAA requiring evaluation for the period of extended operation.

The projected number of major lifts for the Cask Handling Crane is less than 500 cycles. This estimate is based upon the expected number of casks that must be handled during each cask loading campaign and the projected number of campaigns through the period of extended operation. Allowing for double that number for minor lifts, or 1000 cycles, the estimated number of lifts for the Cask Handling Crane, 1500 cycles, is much less than the maximum allowable design value of 100,000 cycles, the Cask Handling Crane load cycle fatigue analyses remain valid for 60 years of plant operation.

The analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(i).

### A.2.4.5.6 Service Level I Coatings Qualification

Service Level 1 coatings used at Seabrook Station are in compliance with the applicable ANSI standards for coating systems inside containment. In a design basis accident, the Emergency Core Cooling System (ECCS) at Seabrook Station pumps water from inside the containment sump to the reactor vessel to keep the core covered with water and make up losses from the pipe break location. These coatings could potentially detach during a design basis accident and the coating debris could contribute to flow blockage of ECCS suction strainers. The ECCS has suction piping located below the waterline inside the sump. Since it is assumed that the degree of radiation exposure used in the original qualification testing was intended to bound 40 years of operation, qualification of Service Level 1 coatings is considered a TLAA. Seabrook Station Service Level I Coatings are managed by the ASME Section XI, Inservice Inspection, Subsection IWE Program, B.2.1.27 and Procedure for Application of Service Level I Coatings. Seabrook Station periodically conducts condition assessments of Service Level I coatings inside containment.

The periodic condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings which may be susceptible to detachment from the substrate during a LOCA event is minimized. The program provides for maintenance of coatings for the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(iii).

#### A.2.4.5.7 Canopy Seal Clamp Assemblies

The canopy seal clamp assemblies were designed for a 40 year design life on the basis of meeting stress limits. The original fatigue analysis considered the forces that would be applied to the center head adapter which maximized the moments on the J-Grove weld and moment along the length of the adapter. The fatigue analysis for the Canopy Seal Clamps is based on the consideration of 400 cycles consisting of 20 occurrences of the Operating Basis Earthquake, each occurrence having 20 cycles of maximum response. This design analysis is a TLAA requiring evaluation for the period of extended operation.

It is projected that 1 OBE would occur for Seabrook Station in 60-years of operation. Since the number of occurrences projected for 60-years is below the design limit of 5 occurrences of 10 cycles the design analysis remains valid for the period of extended operation.

The analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

#### A.2.4.5.8 Hydrogen Analyzer

The Seabrook Station Hydrogen Analyzer was evaluated with respect to radiation exposure. The UFSAR contains accumulated radiation dose limits for a 40-year operating period.

The operational dose for 40-year is $5 \times 10^6$ rads.

The projected maximum 40-year exposure comes from three sources; the gas in the analyzers themselves; the gas in the piping in the room, and the shine from the containment atmosphere through the penetrations into the room. The dose to the recombiner from these three sources is $7.2 \times 10^3$ rads annually. This leads to a projected 60-year dose of $4.32 \times 10^5$ rads which is less than the 40-year design dose.

The analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

#### A.2.4.5.9 Mechanical Equipment Qualification

The Seabrook Station CLB commits to the review and evaluation of the environmental qualification of mechanical equipment to demonstrate compliance with 10 CPR Part 50 General Design Criteria Appendix A.

Results of this evaluation demonstrate safety-related active mechanical equipment located in harsh environments had been adequately addressed.

Since a period of 40 years was used to determine the normal service radiation exposure to the equipment, mechanical equipment qualification (MEQ) is considered a TLAA.

The design basis event conditions during the period of extended operation will remain the same as those in the current license period which have been adjusted to account for previously approved power uprate conditions. Therefore, the design basis event parameters, including the temperature, pressure and time profiles, do not require further evaluation as TLAAs for license renewal.

The effects of aging on the intended function(s) of equipment included under Mechanical Equipment Qualification will be adequately addressed for the period of extended operation. Calculations for Mechanical Equipment Qualification will be revised prior to entering the period of extended operation. Revision of MEQ calculations will be accomplished using techniques currently used under the CLB for equipment qualification including analytical methods, replacement of radiation sensitive materials or equipment replacement, in accordance with 10 CFR 54.21(c)(1)(ii).

#### A.2.4.5.10 Diesel Generator Thermal Cycle Evaluation

The Emergency Diesel Generators provide Emergency Power to Buses 5 and 6. The Emergency Diesel Generators were analyzed for thermal cycling by the engine manufacturer for Environmental Qualification in accordance with IEEE-323. The manufacturer qualified the Diesel Generator for 5454 Full-Temperature Cycles for the forty year design life of the plant. Under current plant operating practices, the Emergency Diesel Generators are operated only occasionally during periodic surveillance and maintenance testing. Monthly testing over 60 years would contribute 720 cycles. Assuming an equal number of starts for maintenance and actual events an additional 1440 cycles could occur. These actual and potential cycles combined equal slightly more than 2160 cycles for the Emergency Diesel Generators. The projected 60 year cycles is much less than the design basis thermal cycling for 40 years.

The analyses will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

## A.3 LICENSE RENEWAL COMMITMENT LIST

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| 1. | PWR Vessel Internals | An inspection plan for Reactor Vessel Internals will be submitted for NRC review and approval at least twenty-four months prior to entering the period of extended operation. | A.2.1.7 | Program to be implemented prior to the period of extended operation. Inspection plan to be submitted to NRC not less than 24 months prior to the period of extended operation. |
| 2. | Closed-Cycle Cooling Water | Enhance the program to include visual inspection for cracking, loss of material and fouling when the in-scope systems are opened for maintenance. | A.2.1.12 | Prior to the period of extended operation |
| 3. | Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems | Enhance the program to monitor general corrosion on the crane and trolley structural components and the effects of wear on the rails in the rail system. | A.2.1.13 | Prior to the period of extended operation |
| 4. | Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems | Enhance the program to list additional cranes for monitoring. | A.2.1.13 | Prior to the period of extended operation |
| 5. | Compressed Air Monitoring | Enhance the program to include an annual air quality test requirement for the Diesel Generator compressed air sub system. | A.2.1.14 | Prior to the period of extended operation |
| 6. | Fire Protection | Enhance the program to perform visual inspection of penetration seals by a fire protection qualified inspector. | A.2.1.15 | Prior to the period of extended operation. |

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| 7. | Fire Protection | Enhance the program to add inspection requirements such as spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates by qualified inspector. | A.2.1.15 | Prior to the period of extended operation. |
| 8. | Fire Protection | Enhance the program to include the performance of visual inspection of fire-rated doors by a fire protection qualified inspector. | A.2.1.15 | Prior to the period of extended operation. |
| 9. | Fire Water System | Enhance the program to include NFPA 25 guidance for "where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing". | A.2.1.16 | Prior to the period of extended operation. |
| 10. | Fire Water System | Enhance the program to include the performance of periodic flow testing of the fire water system in accordance with the guidance of NFPA 25. | A.2.1.16 | Within ten years of entering the period of extended operation. |
| 11. | Fire Water System | Enhance the program to include the performance of periodic visual inspection of the internal surface of the fire protection system upon each entry to the system for routine or corrective maintenance. This inspection will be performed no earlier than 10 years before the period of extended operation. | A.2.1.16 | Prior to the period of extended operation. |
| 12. | Aboveground Steel Tanks | Enhance the program to include components and aging effects required by the Aboveground Steel Tanks. | A.2.1.17 | Prior to the period of extended operation. |
| 13. | Aboveground Steel Tanks | Enhance the program to include an ultrasonic inspection and evaluation of the internal bottom surface of the two Fire Protection Water Storage Tanks. | A.2.1.17 | Within ten years of entering the period of extended operation. |

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| 14. | Fuel Oil Chemistry | Enhance program to add requirements to 1) sample and analyze new fuel deliveries for biodiesel prior to offloading to the Auxiliary Boiler fuel oil storage tank and 2) periodically sample stored fuel in the Auxiliary Boiler fuel oil storage tank. | A.2.1.18 | Prior to the period of extended operation. |
| 15. | Fuel Oil Chemistry | Enhance the program to add requirements to check for the presence of water in the Auxiliary Boiler fuel oil storage tank at least once per quarter and to remove water as necessary. | A.2.1.18 | Prior to the period of extended operation. |
| 16. | Fuel Oil Chemistry | Enhance the program to require draining, cleaning and inspection of the diesel fire pump fuel oil day tanks on a frequency of at least once every ten years. | A.2.1.18 | Prior to the period of extended operation. |
| 17. | Fuel Oil Chemistry | Enhance the program to require ultrasonic thickness measurement of the tank bottom during the 10-year draining, cleaning and inspection of the Diesel Generator fuel oil storage tanks, Diesel Generator fuel oil day tanks, diesel fire pump fuel oil day tanks and auxiliary boiler fuel oil storage tank. | A.2.1.18 | Prior to the period of extended operation. |
| 18. | Reactor Vessel Surveillance | Enhance the program to specify that all pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage. | A.2.1.19 | Prior to the period of extended operation. |
| 19. | Reactor Vessel Surveillance | Enhance the program to specify that if plant operations exceed the limitations or bounds defined by the Reactor Vessel Surveillance Program, such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes on the extent of Reactor Vessel embrittlement will be evaluated and the NRC will be notified. | A.2.1.19 | Prior to the period of extended operation. |

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| 20. | Reactor Vessel Surveillance | Enhance the program as necessary to ensure the appropriate withdrawal schedule for capsules remaining in the vessel such that one capsule will be withdrawn at an outage in which the capsule receives a neutron fluence that meets the schedule requirements of 10 CFR 50 Appendix H and ASTM E185-82 and that bounds the 60-year fluence, and the remaining capsule(s) will be removed from the vessel unless determined to provide meaningful metallurgical data. | A.2.1.19 | Prior to the period of extended operation. |
| 21. | Reactor Vessel Surveillance | Enhance the program to ensure that any capsule removed, without the intent to test it, is stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation. | A.2.1.19 | Prior to the period of extended operation. |
| 22. | One-Time Inspection | Implement the One Time Inspection Program. | A.2.1.20 | Within ten years of entering the period of extended operation. |
| 23. | Selective Leaching of Materials | Implement the Selective Leaching of Materials Program. | A.2.1.21 | Within five years of entering the period of extended operation. |
| 24. | Buried Piping Inspection | Implement the Buried Piping And Tanks Inspection Program. | A.2.1.22 | Within ten years of entering the period of extended operation. |
| 25. | One-Time Inspection of ASME Code Class 1 Small Bore-Piping | Implement the One-Time Inspection of ASME Code Class 1 Small Bore-Piping Program. | A.2.1.23 | Within ten years of entering the period of extended operation. |
| 26. | External Surfaces Monitoring | Enhance the program to specifically address the scope of the program, relevant degradation mechanisms and effects of interest, the refueling outage inspection frequency, the inspections of opportunity for possible corrosion under insulation, the training requirements for inspectors and the required periodic reviews to determine | A.2.1.24 | Prior to the period of extended operation. |

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| | | program effectiveness. | | |
| 27. | Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components | Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. | A.2.1.25 | Prior to the period of extended operation. |
| 28. | Lubricating Oil Analysis | Enhance the program to add required equipment, lube oil analysis required, sampling frequency, and periodic oil changes. | A.2.1.26 | Prior to the period of extended operation. |
| 29. | Lubricating Oil Analysis | Enhance the program to sample the oil for the Switchyard SF6 compressors and the Reactor Coolant pump oil collection tanks. | A.2.1.26 | Prior to the period of extended operation. |
| 30. | Lubricating Oil Analysis | Enhance the program to require the performance of a one-time ultrasonic thickness measurement of the lower portion of the Reactor Coolant pump oil collection tanks prior to the period of extended operation. | A.2.1.26 | Prior to the period of extended operation. |
| 31. | ASME Section XI, Subsection IWL | Enhance procedure to include the definition of "Responsible Engineer". | A.2.1.28 | Prior to the period of extended operation. |
| 32. | Structures Monitoring Program | Enhance procedure to add the aging effects, additional locations, inspection frequency and ultrasonic test requirements. | A.2.1.31 | Prior to the period of extended operation. |
| 33. | Structures Monitoring Program | Enhance procedure to include inspection of opportunity when planning excavation work that would expose inaccessible concrete. | A.2.1.31 | Prior to the period of extended operation. |
| 34. | Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification | Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program. | A.2.1.32 | Prior to the period of extended operation. |

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| | Requirements | | | |
| 35. | Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits | Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits program. | A.2.1.33 | Prior to the period of extended operation. |
| 36. | Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | Implement the Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program. | A.2.1.34 | Prior to the period of extended operation. |
| 37. | Metal Enclosed Bus | Implement the Metal Enclosed Bus program. | A.2.1.35 | Prior to the period of extended operation. |
| 38. | Fuse Holders | Implement the Fuse Holders program. | A.2.1.36 | Prior to the period of extended operation. |
| 39. | Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program. | A.2.1.37 | Prior to the period of extended operation. |
| 40. | 345 KV $SF_6$ Bus | Implement the 345 KV $SF_6$ Bus program. | A.2.2.1 | Prior to the period of extended operation. |
| 41. | Metal Fatigue of Reactor Coolant Pressure Boundary | Enhance the program to include additional transients beyond those defined in the Technical Specifications and UFSAR. | A.2.3.1 | Prior to the period of extended operation. |

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| 42. | Metal Fatigue of Reactor Coolant Pressure Boundary | Enhance the program to implement a software program, to count transients to monitor cumulative usage on selected components. | A.2.3.1 | Prior to the period of extended operation. |
| 43. | Pressure –Temperature Limits, including Low Temperature Overpressure Protection Limits | Seabrook Station will submit updates to the P-T curves and LTOP limits to the NRC at the appropriate time to comply with 10 CFR 50 Appendix G. | A.2.4.1.4 | The updated analyses will be submitted at the appropriate time to comply with 10 CFR 50 Appendix G, Fracture Toughness Requirements. |
| 44. | Environmentally-Assisted Fatigue Analyses (TLAA) | (1) Consistent with the Metal Fatigue of Reactor Coolant Pressure Boundary Program Seabrook Station will update the fatigue usage calculations using refined fatigue analyses, if necessary, to determine acceptable CUFs (i.e., less than 1.0) when accounting for the effects of the reactor water environment. This includes applying the appropriate $F_{en}$ factors to valid CUFs determined from an existing fatigue analysis valid for the period of extended operation or from an analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case).<br><br>(2) If acceptable CUFs cannot be demonstrated for all the selected locations, then additional plant-specific locations will be evaluated. For the additional plant-specific locations, if CUF, including environmental effects is greater than 1.0, then Corrective Actions will be initiated, in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1. Corrective Actions will include inspection, repair, or replacement of the affected locations before exceeding a CUF of 1.0 or the effects of fatigue will be managed by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined | A.2.4.2.3 | At least two years prior to entering the period of extended operation. |

| No. | PROGRAM or TOPIC | COMMITMENT | UFSAR LOCATION | SCHEDULE |
|---|---|---|---|---|
| | | by a method accepted by the NRC). | | |
| 45. | Mechanical Equipment Qualification | Revise Mechanical Equipment Qualification Files. | A.2.4.5.9 | Prior to the period of extended operation. |

# APPENDIX B

# AGING MANAGEMENT PROGRAMS

## APPENDIX B – TABLE OF CONTENTS

# B.1 INTRODUCTION

## B.1.1 OVERVIEW

The license renewal Aging Management Program descriptions are provided in this appendix for each program credited for managing aging effects.

The demonstration that aging effects will be adequately managed is performed by evaluating the aging management programs and associated activities against certain required attributes. Each of the aging management programs described in this appendix has ten elements which are consistent with the attributes described in NUREG-1800 Appendix A.1, *"Aging Management Review – Generic (Branch Technical Position RLSB-1)"* and in NUREG-1800 Table A.1-1 *"Elements of An Aging Management Program for License Renewal"*. The ten element detail is only provided when the program is plant specific.

Credit for existing plant programs was considered and taken where appropriate. The program applicability to Systems, Structure and Components (SSCs) and commodities was considered and a determination made with respect to the effectiveness of the program in managing the aging effects.

Plant existing programs applied to age management are frequently associated with regulatory commitments or requirements. Upon evaluation many of the existing programs meet the license renewal 10 element attributes. If an existing program did not adequately manage the identified aging effect, the program was enhanced as necessary. New programs were created when no program existed.

## B.1.2 PROGRAM PRESENTATION

For those aging management programs that are comparable to the programs described in Sections X and XI of NUREG-1801, or are consistent with exceptions, each program discussion is presented in the following format:

1. Program Description – Summary description of the Seabrook Station program

2. NUREG-1801 Consistency – A statement of the consistency of the program with respect to the NUREG-1801 program

3. Exceptions to NUREG-1801 – Statement of exception(s) to the NUREG-1801 program with justification

4. Enhancements – Summary of enhancements, if needed, to attain consistency with NUREG-1801

5. Operating Experience – Relevant operating experience pertaining to the program

6. Conclusion – A statement attesting to the program adequacy for managing associated aging effects

The statement of consistency of the Seabrook Station Aging Management Program with respect to the NUREG-1801 program will take one of the three following forms:

1. The Aging Management Program states that the plant program is consistent with the recommendations of NUREG-1801, identifies no exceptions to NUREG-1801, and identifies no enhancements. This statement affirms that

   a. the plant program corresponds to and contains the elements of the referenced NUREG-1801 program;

   b. the conditions at the plant are bounded by the conditions for which the NUREG-1801 program was evaluated to the extent such conditions are specified in the NUREG-1801 program description; and

   c. verifications have been completed and documented.

   Therefore, the Aging Management Program identified in NUREG-1801 is being used.

2. The Aging Management Program states that the plant program is consistent with the recommendations of NUREG-1801 with exception(s), and either identifies enhancements or identifies no enhancements. This statement affirms that

   a. with the exception(s) identified, and enhancements, if any, the plant program corresponds to and contains the elements of the referenced NUREG-1801 program;

   b. the conditions at the plant are bounded by the conditions for which the NUREG-1801 program was evaluated to the extent such conditions are specified in the NUREG-1801 program description; and

   c. verifications have been completed and documented.

Therefore the Aging Management Program identified in NUREG-1801 is being used, as modified by the exceptions. A justification for each identified exception is provided.

3. The Aging Management Program states that the plant program is consistent with the recommendations of NUREG-1801, identifies no exceptions to NUREG-1801, but identifies enhancements. This statement affirms that

   a. with those enhancements, the plant program corresponds to and contains the elements of the referenced NUREG-1801 program;

   b. the conditions at the plant are bounded by the conditions for which the NUREG-1801 program was evaluated to the extent such conditions are specified in the NUREG-1801 description; and

   c. verifications have been completed and are documented.

   Therefore the Aging Management Program identified in NUREG-1801 is being used.

The plant specific aging management programs are described in terms of the 10 program elements in NUREG-1800, Section A.1.2.3 "Aging Management program Elements".

## B.1.3 QUALITY ASSURANCE PROGRAM AND ADMINISTRATIVE CONTROLS

The FPL/NextEra Energy Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A of NUREG-1801. The elements of corrective action, confirmation process, and administrative controls in the Quality Assurance Program are applicable to both safety related and non-safety related systems, structures, and components that are subject to an Aging Management Review. Each element will be implemented as follows:

**Corrective Actions**

FPL/NextEra Energy Quality Assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. Conditions adverse to quality, such as failures, malfunctions, deviations, defective material and equipment, and non-conformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, measures are implemented to ensure that the cause of the nonconformance is determined and that corrective action is taken to preclude recurrence. In addition, the root cause of the significant condition adverse to quality and the corrective action implemented are documented and reported to appropriate levels of management.

**Confirmation Process**

FPL/NextEra Energy Quality Assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. Corrective actions and administrative controls are accomplished by the implementation of the Corrective Action Program and Nuclear Fleet procedures. The confirmation process is part of the Corrective Action Program and includes:

1. review to assure that proposed actions are adequate
2. tracking and reporting of open corrective actions
3. review of corrective action effectiveness

Any follow-up activities required by the confirmation process are documented in accordance with the Corrective Action Program. The Corrective Action Program encompasses the confirmation process for Aging Management Programs and activities.

**Administrative Controls**

FPL/NextEra Energy Quality Assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. These administrative controls include the provisions related to organization and management, procedures, record keeping, review and audit, and reporting.

## B.1.4 OPERATING EXPERIENCE

Operating experience is an important resource in identifying aging effects and evaluating the effectiveness of Aging Management Programs. The Corrective Action Program, system health reports, self assessments, Nuclear Oversight Audits, and interviews with site personnel were the primary sources of plant-specific operating experience related to these Aging Management Programs.

Since the materials used for structures and components at Seabrook Station are common to most nuclear power plants and many non-nuclear power plants that have long operating histories, industry-wide operating experience is also valuable. Screening of a large body of operating data yielded much useful data relating to aging of plant structures and components.

The Seabrook Station plant-specific data and the industry-wide operating data were valuable in:

1. Providing bases for determining which aging effects require management.

2. Demonstrating that existing programs are adequately managing the aging effects.

3. Pointing out the need to enhance existing programs or the need for entirely new programs.

The effects and mechanisms of age related degradation for SSCs at Seabrook Station were developed from several sources. They include plant-specific and industry operating experience and interviews with site personnel. No new aging effects were identified. Known aging effects and mechanisms for a given environment and material were incorporated into NUREG-1801, up to the time of its publication in Sept. 2005.

With respect to Aging Management Programs, existing programs/ activities must demonstrate, with objective evidence, that they are effective in managing the aging effects if credited. Operating experience related to the program/activity, including past corrective actions resulting in program enhancements, provides objective evidence the program adequately manages the aging effects.

The FPL/NextEra Energy Operating Experience Program provides guidance for using, sharing, and evaluating operating experience information at FPL/NextEra Energy Nuclear Division sites. The procedure governing this program provides guidance on the effective and efficient use of operating experience information. The primary objectives of the FPL/NextEra Energy Operating Experience Program are:

1. Systematic evaluation of significant nuclear plant operating experiences.

2. Incorporation of lessons learned into appropriate plant practices, policies, programs, and procedures with the objective of preventing similar issues.

3. Sharing of lessons learned internally and with other utilities to promote industry-wide safety and reliability.

By increasing awareness of previous FPL/NextEra Energy Nuclear Division and industry events and issues, the FPL/NextEra Energy Operating Experience Program expects to prevent similar events from occurring at FPL/NextEra Energy Nuclear Division sites. The FPL/NextEra Energy Operating Experience

Program ensures that information that has the potential to affect safe and reliable station operation is properly screened and addressed to ensure timely response. This program promotes the identification and transfer of lessons learned from industry, and internal events, such that these lessons are shared between the FPL/NextEra Energy Nuclear Division and the nuclear industry. This program describes the methodology for receiving, processing, status reporting, screening, reviewing, evaluating, and taking preventive/corrective actions in response to operating experience information. This program satisfies the requirements of NUREG-0737, I.C.5, *"Procedures for Feedback of Operating Experience to Plant Staff"* and 10 CFR 50.65, *"Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"*, Sections (a)(1) and (a)(2).

## B.1.5 NUREG-1801 CHAPTER XI AGING MANAGEMENT PROGRAMS

The following Aging Management Programs are described in this appendix. The programs are either generic in nature as discussed in NUREG-1801, Chapter XI, *"Aging Management Programs (AMPs)"* or are plant-specific. NUREG-1801 Chapter XI programs are listed in Section B.2.1. Plant-specific programs are listed in Section B.2.2. All generic programs are either fully consistent with or are consistent with some exceptions with programs discussed in NUREG-1801. Programs are identified as either existing or new.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.1.1) [Existing]
2. Water Chemistry (B.2.1.2) [Existing]
3. Reactor Head Closure Studs (B.2.1.3) [Existing]
4. Boric Acid Corrosion (B.2.1.4) [Existing]
5. Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (B.2.1.5) [Existing]
6. Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (B.2.1.6) [Not Used]
7. PWR Vessel Internals (B.2.1.7) [New]
8. Flow-Accelerated Corrosion (B.2.1.8) [Existing]
9. Bolting Integrity (B.2.1.9) [Existing]
10. Steam Generator Tube Integrity (B.2.1.10) [Existing]
11. Open-Cycle Cooling Water System (B.2.1.11) [Existing]
12. Closed-Cycle Cooling Water System (B.2.1.12) [Existing]

13. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (B.2.1.13) [Existing]
14. Compressed Air Monitoring (B.2.1.14) [Existing]
15. Fire Protection (B.2.1.15) [Existing]
16. Fire Water System (B.2.1.16) [Existing]
17. Aboveground Steel Tanks (B.2.1.17) [Existing]
18. Fuel Oil Chemistry (B.2.1.18) [Existing]
19. Reactor Vessel Surveillance (B.2.1.19) [Existing]
20. One-Time Inspection (B.2.1.20) [New]
21. Selective Leaching of Materials (B.2.1.21) [New]
22. Buried Piping and Tanks Inspection (B.2.1.22) [New]
23. One-Time Inspection of ASME Code Class 1 Small Bore-Piping (B.2.1.23) [New]
24. External Surfaces Monitoring (B.2.1.24) [Existing]
25. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.25) [New]
26. Lubricating Oil Analysis (B.2.1.26) [Existing]
27. ASME Section XI, Subsection IWE (B.2.1.27) [Existing]
28. ASME Section XI, Subsection IWL (B.2.1.28) [Existing]
29. ASME Section XI, Subsection IWF (B.2.1.29) [Existing]
30. 10 CFR 50, Appendix J (B.2.1.30) [Existing]
31. Structures Monitoring Program (B.2.1.31) [Existing]
32. Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements (B.2.1.32) [New]
33. Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits (B.2.1.33) [New]
34. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 EQ Requirements (B.2.1.34) [New]
35. Metal Enclosed Bus (B.2.1.35) [New]
36. Fuse Holders (B.2.1.36) [New]

37. Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements (B.2.1.37) [New]

38. 345 KV SF6 Bus (B.2.2.1) [New]

39. Boral Monitoring (B.2.2.2) [Existing]

40. Nickel-Alloy Nozzles and Penetrations (B.2.2.3) [Existing]

## B.1.6 NUREG-1801 CHAPTER X AGING MANAGEMENT PROGRAMS

The following NUREG-1801 Chapter X, "*Time-Limited Aging Analyses Evaluation of Aging Management Programs Under 10 CFR 54.21 (c)(1)(iii)*": Aging Management Programs are described in Section B.2.3 of this appendix as indicated. Programs are identified as either existing or new.

1. Metal Fatigue of Reactor Coolant Pressure Boundary (B.2.3.1) [Existing]

2. Environmental Qualification (EQ) of Electrical Components (Section B.2.3.2) [Existing]

## B.2 AGING MANAGEMENT PROGRAMS

### B.2.0 AGING MANAGEMENT CORRELATION CHART- NUREG-1801 TO SEABROOK STATION PROGRAMS

The following Aging Management Programs are discussed in this appendix. The programs are as discussed in NUREG-1801 or are specific to Seabrook Station. All programs are fully consistent with NUREG-1801 or are consistent with exceptions to the programs discussed in NUREG-1801.

| NUREG-1801 Number | NUREG-1801 Program | Seabrook Station Program |
|---|---|---|
| XI.M1 | ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD | ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD |
| XI.M2 | Water Chemistry | Water Chemistry |
| XI.M3 | Reactor Head Closure Studs | Reactor Head Closure Studs |
| XI.M4 | BWR Vessel ID Attachment Welds | Not Applicable (Seabrook Station is a PWR) |
| XI.M5 | BWR Feedwater Nozzle | Not Applicable (Seabrook Station is a PWR) |
| XI.M6 | BWR Control Rod Drive Return Line Nozzle | Not Applicable (Seabrook Station is a PWR) |
| XI.M7 | BWR Stress Corrosion Cracking | Not Applicable (Seabrook Station is a PWR) |
| XI.M8 | BWR Penetrations | Not Applicable (Seabrook Station is a PWR) |
| XI.M9 | BWR Vessel Internals | Not Applicable (Seabrook Station is a PWR) |
| XI.M10 | Boric Acid Corrosion | Boric Acid Corrosion |
| XI.M11A | Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors | Nickel Alloy Penetration Nozzles Welded to the Upper RV Closure Heads of PWRS |
| XI.M12 | Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) | Not Used. Not credited for aging management. |

| NUREG-1801 Number | NUREG-1801 Program | Seabrook Station Program |
|---|---|---|
| XI.M13 | Thermal Aging and neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) | Not Used. Not credited for aging management. |
| XI.M14 | Loose Part Monitoring | Not Used. Not credited for aging management. |
| XI.M15 | Neutron Noise Monitoring | Not Used. Not credited for aging management. |
| XI.M16 | PWR Vessel Internals | PWR Vessel Internals |
| XI.M17 | Flow-Accelerated Corrosion | Flow Accelerated Corrosion |
| XI.M18 | Bolting Integrity | Bolting Integrity |
| XI.M19 | Steam Generator Tube Integrity | Steam Generator Tube Integrity |
| XI.M20 | Open-Cycle Cooling Water System | Open-Cycle Cooling Water System |
| XI.M21 | Closed-Cycle Cooling Water System | Closed-Cycle Cooling Water System |
| XI.M22 | Boraflex Monitoring | Not Used. Not credited for aging management. This material is not credited in the Spent Fuel Pool Criticality Analysis |
| XI.M23 | Inspection of Overhead Heavy Load and Light Load (Related to Refueling)Handling Systems | Inspection of Overhead Heavy Load And Light Load (Related to Refueling) Handling Systems |
| XI.M24 | Compressed Air Monitoring | Compressed Air Monitoring |
| XI.M25 | BWR Reactor Water Cleanup System | Not Applicable (Seabrook Station is a PWR) |
| XI.M26 | Fire Protection | Fire Protection |
| XI.M27 | Fire Water System | Fire Water System |
| XI.M28 | Buried Piping and Tanks Surveillance | Not Used. The aging effects associated with buried piping and tanks will be adequately managed by XI.M34, Buried Piping Inspection Program (B.2.1.22). |
| XI.M29 | Aboveground Steel Tanks | Above Ground Steel Tanks |

| NUREG-1801 Number | NUREG-1801 Program | Seabrook Station Program |
|---|---|---|
| XI.M30 | Fuel Oil Chemistry | Fuel Oil Chemistry |
| XI.M31 | Reactor Vessel Surveillance | Reactor Vessel Surveillance |
| XI.M32 | One-Time Inspection | One-Time Inspection |
| XI.M33 | Selective Leaching of Materials | Selective Leaching of Materials |
| XI.M34 | Buried Piping and Tanks Inspection | Buried Piping and Tanks Inspection |
| XI.M35 | One-Time Inspection of ASME Code Class 1 Small Bore-Piping | One-Time Inspection of ASME Code Class 1 Small Bore Piping |
| XI.M36 | External Surfaces Monitoring | External Surfaces Monitoring |
| XI.M37 | Flux Thimble Tube Inspection | Not Used. Not credited for aging management. |
| XI.M38 | Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components | Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components |
| XI.M39 | Lubricating Oil Analysis | Lubricating Oil Analysis |
| XI.S1 | ASME Section XI, Subsection IWE | ASME Section XI, Subsection IWE |
| XI.S2 | ASME Section XI, Subsection IWL | ASME Section XI, Subsection IWL |
| XI.S3 | ASME Section XI, Subsection IWF | ASME Section XI, Subsection IWF |
| XI.S4 | 10 CFR 50, Appendix J | 10 CFR Part 50 Appendix J |
| X1.S5 | Masonry Wall Program | Included In Structures Monitoring Program |
| XI.S6 | Structures Monitoring Program | Structures Monitoring Program |
| XI.S7 | RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants | Included In Structures Monitoring Program |
| XI.S8 | Protective Coating Monitoring and Maintenance Program | Not Used. Not credited for aging management. |
| XI.E1 | Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements |

| NUREG-1801 Number | NUREG-1801 Program | Seabrook Station Program |
|---|---|---|
| XI.E2 | Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits | Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Inst. Circuits |
| XI.E3 | Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements | Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Requirements |
| XI.E4 | Metal Enclosed Bus | Metal Enclosed Bus |
| XI.E5 | Fuse Holders | Fuse Holders |
| XI.E6 | Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements | Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements |
| X.M1 | Metal Fatigue Of Reactor Coolant Pressure Boundary | Metal Fatigue of Reactor Coolant Pressure Boundary Program |
| X.S1 | Concrete Containment Tendon Prestress | Not Used. Not credited for aging management. |
| X.E1 | Environmental Qualification (EQ) Of Electrical Components | Environmental Qualification (EQ) of Electric Components |
| N/A | Seabrook Station Plant Specific Program | 345 KV SF6 Bus |
| N/A | Seabrook Station Plant Specific Program | Boral Monitoring |
| N/A | Seabrook Station Plant Specific Program | Nickel-Alloy Nozzles and Penetrations |

## B.2.1 NUREG-1801 CHAPTER XI AGING MANAGEMENT PROGRAMS

### B.2.1.1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

**Program Description**

The Seabrook Station ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program is an existing program that manages the aging effects of cracking due to cyclic loading, primary water stress corrosion cracking, stress corrosion cracking, cracking due to thermal and mechanical loading, loss of fracture toughness due to thermal aging embrittlement, loss of material due to general, pitting and crevice corrosion, and loss of material due to wear in ASME Code Class 1, 2, and 3 piping and components within the scope of License Renewal. The program includes periodic visual, surface and/or volumetric examinations of all ASME Code Class 1, 2 and 3 pressure-retaining components, their supports and integral attachments (including welds, pump casings, valve bodies and pressure-retaining bolting) and leakage tests of pressure retaining components. These are identified in ASME Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components"*, or commitments requiring augmented Inservice Inspections, and are within the scope of license renewal.

The Code of Federal Regulations, 10 CFR 50.55a, *"Codes and Standards"*, requires that Inservice Inspection of ASME Code Class 1, 2, and 3 pressure retaining components, their integral attachments and supports be conducted in accordance with the latest edition of ASME Section XI approved by the NRC twelve months prior to the start of a ten-year interval. The Inservice Inspection Program for the second ($2^{nd}$) ten-year interval, which began on August 19, 2000 for Seabrook Station, implements the 1995 edition with the 1996 addenda, of ASME Section XI. The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives, and utilizes ASME Section XI, Subsections IWB, IWC, and IWD. The program provides for augmented inspections of components and their attachments as required or recommended by regulatory, or Nuclear Steam System Supplier (NSSS) technical publications.

The Seabrook Station ASME Section XI Program provides the requirements for Inservice Inspection, repair, and replacement of all ASME Code Class 1, 2, and 3 components within scope for license renewal. The program includes those components specified in subsections IWB-1100, IWC-1100, and IWD-1100 for Class 1, 2, and 3 components, respectively, and includes all pressure retaining components and their integral attachments. The components described in sub-articles IWB-1220, IWC-1220 and IWD-1220 (Components Exempt from Examination) are exempt from the examination and pressure test requirements of sub-articles IWB-2500, IWC-2500 and IWD-2500. The components in scope of the Seabrook Station ASME Section XI program are included in the Inservice Inspection program and repair and replacement activities, and are implemented in accordance with the requirements of ASME Section XI, subsection IWA-4000.

The Technical Specification commitments for the scope of this program, as described in the Seabrook Station In-service Inspection Reference Manual, are addressed in Technical Specification Sections 4.0.5, *"Surveillance Requirements for Inservice Inspection"*, and 6.7, *"Programs and Procedures"*.

The Seabrook Station Inservice Inspection Program consists of condition monitoring activities that detect degradation of components before loss of intended function. No preventive or mitigating attributes are associated with these activities.

The Seabrook Inservice Inspection program utilizes ASME Section XI Tables IWB-2500-1, IWC-2500-1 and IWD-2500-1, for Class 1, 2 and 3 components respectively, to determine the examination requirements, develop the examination procedures, and schedule the examinations required for each inspection interval and the examinations for each inspection period. The tables specify the extent and schedule of the inspection and the examination methods for the components of pressure-retaining boundaries.

The Seabrook Station ASME Section XI Inservice Inspection Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function.

The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of a component intended function. Inspection can reveal crack initiation and growth, loss of material due to corrosion, leakage, and indications of degradation caused by wear or stress relaxation, such as verification of clearances, settings, physical displacements, loose or missing parts, debris, wear, erosion, or loss of integrity at bolted or welded connections.

The program uses three types of examination; visual, surface, and volumetric in accordance with the general requirements of Article IWA-2000. VT-1 visual examination detects discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surface of components. VT-2 visual examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test. VT-3 visual examination (a) determines the general mechanical and structural condition of components by verifying parameters, such as clearances, settings, and physical displacements; and (b) detects discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

Surface examination uses magnetic particle, liquid penetrant, or eddy current examinations to indicate the presence of surface discontinuities and flaws.

Volumetric examination uses radiographic, ultrasonic, or eddy current (for steam generator tubing only) examinations to indicate the presence of discontinuities or flaws throughout the volume of material.. If used for examinations other than steam generator tube inspections, eddy current is considered a surface examination technique. The non-destructive examination (NDE) techniques used to inspect ASME Code Class 1, Class 2, and Class 3 components are consistent with the referenced ASME Section XI Code for those components.

Examination requirements for Class 1 and 2 pressure-retaining bolting are in accordance with ASME Section XI, Table IWB 2500-1 or IWC 2500-1. For Class 1 components, Table IWB 2500-1, examination category B-G-1, for bolting greater than 2 inches in diameter, specifies volumetric examination of studs and bolts and VT-1 visual examination of surfaces of nuts, washers, bushings, and flanges. Examination category B-G-2, for bolting 2 inches or smaller requires VT-1 visual examination of surfaces of bolts, studs, and nuts.

Examination Categories B-P and C-H, require VT-2 visual examination (IWA-5240) during system leakage testing of all pressure-retaining ASME Code Class 1 and 2 components, according to Tables IWB-2500-1 and IWC-2500-1 respectively. The extent and schedule of inspections, in accordance with Tables IWB-2500-1 and IWC-2500-1 ensure detection of aging degradation before the loss of the intended function. The Seabrook Station ASME Section XI Inservice Inspection Program performs the necessary inspections per the requirements of tables IWB-2500-1, IWC-2500-1, and IWD-2500-1. These inspections include the applicable portions of examination categories; B-A, B-B, B-D, B-G-1, B-G-2, B-K, B-L-2, B-M-2, B-N-1, B-N-2, B-N-3 B-O, B-P, C-A, C-B, C-C, C-F-1, C-F-2, C-H, D-A, and D-B. The applicable portions of categories B-F and B-J are currently included in the Risk Informed Inservice Inspection Program. Examination categories not listed above are not applicable to Seabrook Station.

The examinations listed above are based on specific ASME Code Editions and Addenda and will change throughout the extended period as required and/or allowed by 10 CFR 50.55a.

The examination schedules contained in the Seabrook Station Inservice Inspection Program meet the requirements of ASME Section XI, IWB-2412, IWC-2412, and IWD-2412, respectively (Inspection Program B). The Inservice Inspection Program also meets the requirements for the extent and frequency of examinations specified by the ASME Section XI, IWB-2500-1, IWC-2500-1, and IWD-2500-1 and in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives. If flaw indications or relevant conditions of degradation are found, additional examinations may be necessary.

In some cases, an evaluation in accordance with ASME Section XI, IWB-3100, IWC-3100, or IWD-3100 may be used to qualify a component with flaw indications as acceptable for continued service. In such cases, the areas containing such flaw indications and relevant conditions are reexamined during the next three inspection periods of IWB-2400 for Class 1 components and for the next inspection period of IWC-2400 and IWD-2400 for Class 2 and Class 3 components, respectively. Examinations that reveal indications that exceed the acceptance standards are extended to include additional examinations in accordance with ASME Section XI, IWB-2430, IWC-2430, or IWD-2430 for Class 1, 2, or 3 components, respectively. The Seabrook Station Inservice Inspection Program meets the ASME Section XI requirements with respect to inspection schedules, extent, method, and frequency of examination, flaw evaluations, and additional examinations.

Indications or relevant conditions are evaluated in accordance with IWB-3000, IWC-3000, or IWD-3000 for Class 1, 2, or 3 components, respectively. Examination results are evaluated in accordance with IWB-3100, IWC-3100, or IWD-3100 by comparing the results with the acceptance standards of IWB-3400 and IWB-3500 or IWC-3400 and IWC-3500 or IWD-3400 and IWD-3500 for Class 1 or Class 2 and 3 components, respectively. In rare cases, flaws exceeding the size of allowable flaws, as defined in IWB-3500 or IWC-3500 may be evaluated by using the analytical procedures of IWB-3600 or IWC-3600.

Repairs and replacements are performed in accordance with the Seabrook Station ASME Section XI Repair and Replacement Program.

**NUREG-1801 Consistency**

NUREG-1801, Rev 1, discusses the use of the 2001 edition including the 2002 and 2003 addenda of ASME Section XI code, but allows use of other editions of the ASME Code as long as there is justification. The Seabrook Station Inservice Inspection Program Plan for the second ten-year inspection interval effective from August 19, 2000 through August 18, 2010, approved per 10 CFR 50.55a, is based on the 1995 edition including the 1996 addenda. The next and subsequent 120-month inspection intervals for Seabrook Station will incorporate the requirements specified in the version of the ASME

Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

This program is consistent with NUREG-1801 XI.M1.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

The Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues and events. Those issues and events, whether external or plant specific, that are potentially significant to the ASME Section XI Inservice Inspection Program are evaluated. The ASME Section XI Inservice Inspection Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Site specific Operating Experience is included below.

1. In February 2001, a post-outage review of the work performed on two valves to replace the body to bonnet gaskets revealed that there had been an *opportunity* to perform a VT-3 visual examination of the internal surface of the valve body during that work. ASME Section XI, Table IWB-2500-1, Examination Category B-M-2, Item No. B12.50 requires an internal surface examination of at least one valve in each group of similar valves during the Ten-Year Inservice Inspection Interval. These valves are two of six in a particular group. Although there was no requirement to perform a VT-3 visual examination during this work, doing so might have precluded the need to open and inspect another valve in this group during the ten year interval.

   As a result of this discovery, the applicable maintenance valve procedures were revised to add a note to the prerequisites section. The note states that a VT-3 visual examination may be required when a Class 1 valve, greater than 4 inches is disassembled. A sign off step was added that states that engineering has evaluated the need for a VT-3 visual examination.

   This example demonstrates that deficiencies in procedures are identified and updated as necessary to ensure the program remains effective.

2. In March, 2008, while performing a VT-2 visual examination during an ASME Section XI Inservice Inspection of the Containment Building Spray system, the VT-2 visual examiner identified excessive dry boric acid accumulation on Containment Building Spray system valve (CBS-V-17) gland leak-off plug. Subsequently, the boric acid leakage was evaluated per the Seabrook Station Boric Acid Corrosion Program and the gland leak-off plug was tightened and the leakage was stopped.

   Similarly, on September 6, 2006, while performing a VT-2 visual examination during an ASME Section XI Inservice Inspection of the Main Steam supply to the Emergency Feedwater Pump turbine, the VT-2 inspector identified a packing leak on a Main Steam valve (MS-V-402) with the Emergency Feedwater steam supply header pressurized to Main Steam header pressure. Subsequently, the packing was adjusted to stop the packing leak.

   These examples demonstrate that the VT-2 inspections per the ASME Section XI Inservice Inspection Program have been effective in identifying degraded conditions and corrective actions have been taken prior to loss of intended function.

3. Prior to all refueling outages in which ASME Section XI Inservice Inspections are scheduled to be performed, a review is conducted of all of the previous Inservice Inspection data reports. During data review for the upcoming Refueling Outage 12 (Spring of 2008) Inservice Inspection activity, an observation was made related to the regenerative heat exchanger shell circumferential welds. The two welds at issue were last examined during Refueling Outage 9 (Fall of 2003) and were not part of the planned Refueling Outage 12 Inservice Inspection scope.

   The previous data indicated that the examination area only included the inner 1/3 thickness of the weld plus 1/2 inch of the adjacent base material and not the required full volume of the weld plus 1/2 inch of the adjacent base material required by ASME Section XI table IWC-2500-1. This meant that the examination was performed only on a portion of the examination volume specified in ASME Section XI and the Seabrook Station Inservice Inspection Reference Manual.

   As a result of this limited inspection, a procedure update was processed to prevent recurrence. The required examination volume was inspected during that upcoming outage.

   This example demonstrates that deficiencies in procedures are identified and updated as necessary to ensure that the program remains effective.

4. In July 2009, a Nuclear Oversight Audit was performed on the Seabrook Station ASME Section XI Inservice Inspection Program. The Inservice Inspection audit evaluated the Seabrook Station Inservice Inspection Program, processes, personnel qualifications, and documentation/records for compliance with Regulatory and Seabrook Station requirements. The audit was performed through document review, personnel interviews, and field observations.

   The audit team concluded that Seabrook Station's implementation of the Inservice Inspection Program and related activities, and related Inservice Inspection Technical Specification surveillances are being performed at the required frequency and are effectively implemented in accordance with Regulatory, ASME Code, Seabrook Station, and Industry requirements.

   This example demonstrates that periodic assessments are performed to evaluate the effectiveness of the program and to identify the areas that need improvement to maintain the effective performance of the program.

5. During Refueling Outage 13 (Fall of 2009), an ultrasonic examination was performed on Train "B" Residual Heat Removal (RH) Mixing Tee to detect thermal fatigue cracking as recommended by Electric Power Research Institute Material Reliability Program (EPRI MRP-192), *"Materials Reliability Program: "Assessment of RHR Mixing Tee Thermal Fatigue in PWR Plants"*. The area examined included the downstream tee-to-pipe weld and the adjacent base material for 0.25 inch past the counterbore. The ultrasonic examination identified a rejectable indication in the pipe wall downstream of the Mixing Tee.

   Evaluation of the condition concluded that the flaw was indicative of thermal fatigue cracking in the 'B' Train Residual Heat Removal piping downstream of the mixing tee. Thermal fatigue cracking had been identified at the location of hot and cold water mixing in Residual Heat Removal Systems in foreign nuclear power plants as discussed in EPRI MRP-192. Mixing of hot and cold fluids at the mixing tee downstream of the heat exchanger had caused thermal fatigue cracking of the piping as specifically described in the EPRI MRP.

   An Operability Determination was performed to demonstrate the continued operability of the system. As part of the extent of condition review, an ultrasonic examination of the Train "A" Residual Heat Removal Mixing Tee was performed and no indications of cracking were identified. The piping downstream of the Train "B" Residual Heat Removal Mixing Tee was replaced after full core offload. Residual Heat Removal system run time history was reviewed and documented for both trains from plant startup through and including Cycle 13 in support of developing re-inspection intervals in accordance with MRP-192. The affected areas were included in the ASME Section XI Inservice Inspection Program for inspection at a frequency commensurate with that defined in MRP-192.

This example demonstrates that augmented inspections are conducted and appropriate actions taken in response to industry events.

These examples provide objective evidence that the Seabrook Station ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program has been effective in identifying degraded conditions in advance of any loss of intended function. Appropriate guidance for evaluation, repair, or replacement is provided for locations where degradation is found. Assessments of the ASME Section XI Inservice Inspection program are performed to identify the areas that need improvement to maintain the effective performance of the program.

**Conclusion**

The Seabrook Station ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.2 WATER CHEMISTRY

**Program Description**

The Seabrook Station Water Chemistry Program is an existing program that manages the aging effects of cracking, loss of material, and reduction of heat transfer in the primary and secondary water systems. The primary scope of this program consists of the Reactor Coolant system and related auxiliary systems containing treated water, reactor coolant, treated borated water and steam. Chemistry programs are used to control water chemistry for impurities and rely upon periodic monitoring and control of detrimental contaminants below the levels known to cause cracking, loss of material or reduction of heat transfer. Seabrook Station is a pressurized water reactor and subscribes to those objectives and guidelines applicable to pressurized water reactors (PWRs).

NUREG-1801 Section XI.M2 for Water Chemistry state that the water chemistry program for PWRs relies on monitoring and control of reactor water chemistry based on industry guidelines such as the Electric Power Research Institute (EPRI) technical reports TR-105714, Rev. 3 and TR-102134, Rev. 3, or later revisions. The latest revisions to these industry guidance documents were published as 1014986, *"Pressurized Water Reactor Primary Water Chemistry Guidelines - Revision 6"*, and 1016555, *"Pressurized Water*

*Reactor Secondary Water Chemistry Guidelines - Revision 7"*. Seabrook Station uses these latest revisions as the bases for the plant water chemistry program, as allowed by NUREG-1801.

Seabrook Station uses a One-Time Inspection Program of selected components at susceptible (low-flow or stagnant) locations to verify the effectiveness of the chemistry control program.

The Water Chemistry Program mitigates the aging effects of loss of material due to general, pitting, and crevice corrosion; cracking due to Stress Corrosion Cracking (SCC); Steam Generator tube degradation caused by intergranular attack (IGA) and outer diameter stress corrosion cracking (ODSCC); and reduction of heat transfer caused by fouling for the internal surfaces of primary and secondary systems by controlling the chemical species that cause the underlying aging mechanisms that result in the aging effects. The chemistry parameters measured are defined and listed in the Seabrook Station Primary and Secondary Water Chemistry Monitoring Programs for all modes of operation. Because it is a mitigation program, the Water Chemistry Program does not detect aging effects directly. However, in selected areas it does monitor for the presence of iron and copper, which could indicate loss of material in some components.

The chemistry parameters, including chlorides, fluorides, dissolved oxygen and sulfate concentrations, are measured utilizing standard proven industry techniques. Water chemistry control is in accordance with EPRI 1014986, "*Pressurized Water Reactor Primary Water Chemistry Guidelines - Revision 6*", for primary water chemistry and EPRI 1016555, *"Pressurized Water Reactor Secondary Water Chemistry Guidelines - Revision 7"*, for secondary water chemistry.

The Seabrook Station Water Chemistry Program establishes the plant water chemistry specifications for chemical species, sampling and analysis frequencies, and corrective actions (e.g., actions levels and responses to out-of-specification water chemistry conditions). This program is administered in accordance with the Seabrook Station Chemistry Manual, including the Seabrook Station Primary Chemistry Control Program and Secondary Chemistry Control Program. These programs and procedures provide the necessary primary and secondary water chemistry controls to minimize contaminant concentrations and to mitigate loss of material due to general, crevice and pitting corrosion and cracking caused by SCC.

*PWR Primary Water Chemistry:*

The Seabrook Station Primary Chemistry Control Program presents a recommended sampling schedule, species to be analyzed, their limits, and short-term corrective actions for anomalous results for the following systems and components:

a. Reactor Coolant
b. Accumulators
c. Spent Fuel Pool
d. Refueling Canal
e. Demineralized-Water Header
f. Chemical and Volume Control
g. Boric Acid Storage Tanks
h. Refueling Water Storage Tank
i. Reactor Makeup Water Storage Tank
j. Spray Additive Tank

Additionally, the Seabrook Station Primary Chemistry Control Program provides parameter control limits for the respective sample points, as well as appropriate short-term responses to anomalous or out-of-control analysis results. The Primary Water Chemistry procedure follows EPRI water chemistry guidelines in monitoring the concentration of chlorides, fluorides, sulfates, lithium, and dissolved oxygen and hydrogen. One of the objectives of the Primary System Strategic Water Chemistry Plan is contaminant minimization. This objective is implemented through a routine sampling and analysis program, with the data reviewed by personnel at three different levels in the Seabrook organization.

*PWR Secondary Water Chemistry:*

The Seabrook Station Secondary Chemistry Control Program sets the normal sampling schedule, limits, and corrective actions when limits are exceeded for steam generator blow down, feed water and condensate under power operation, start-up, shutdown, and wet lay-up conditions. This procedure follows EPRI water chemistry guidelines in monitoring secondary plant parameters such as calculated concentration of hydrogen ions (pH) level, cation conductivity, sodium, chloride, sulfate, lead, dissolved oxygen, iron, copper, and hydrazine. The program also includes sampling and control of chemistry parameters for the auxiliary systems such as the Auxiliary Boiler, Demineralized Water and Condensate Storage Tank.

The Seabrook Station Water Chemistry Program specifies the frequency of sampling. Routine primary and secondary system sampling frequencies are specified in station procedures in accordance with EPRI water chemistry guidelines. The Seabrook Station Water Chemistry Program contains guidance on increasing sampling frequency to address an abnormal chemistry condition.

The Seabrook Station Water Chemistry Program contains the acceptance criteria for various contaminants including limits specified in the EPRI water chemistry guidelines. The program also contains the actions to be taken for different levels of the contaminants. Actions to be taken upon reaching each chemistry action level are described in Seabrook Station procedures as required by the EPRI water chemistry guidelines. Evidence of aging effects or unacceptable water chemistry results are evaluated using these procedures and addressed by the Seabrook Station Corrective Action Program.

The Seabrook Station Water Chemistry Program follows the EPRI water chemistry guidelines in identifying actions to be taken and time periods to be imposed when returning parameters found to be outside the specified limits back within the acceptable range. Unexpected and unacceptable chemistry conditions are documented and evaluated in accordance with the Seabrook Station Corrective Action Program.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M2.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

The Water Chemistry Program is a mitigation program that assures contaminants are maintained below applicable limits to mitigate the aging of plant piping and components. Demonstrations that the aging effects are effectively managed are achieved through objective evidence that shows that cracking, loss of material and reduction of heat transfer are being adequately managed. The following examples of operating experience provide objective evidence that the Seabrook Water Chemistry aging management program will be effective in ensuring that intended functions will be maintained consistent with the current licensing bases for the period of extended operation.

1. In December 2001, Condensate Storage Tank oxygen levels trended upwards, increasing above 75 ppb. Review of the data indicated a potential source of air in-leakage into the Condensate Storage Tank. The Demineralized Water Storage Tanks were verified to have low oxygen levels indicating the condition did not involve Demineralized Water production or storage in the Water Treatment and Demineralized Water systems.

The oxygen levels did not decrease as expected when the Steam Generator Blowdown system was aligned to the ocean and the Condensate Storage Tank was partially refilled several times from the Demineralized Water Storage Tank. As part of the corrective actions; a) the affected portions of the Condensate and Demineralized Water systems were walked down, b) oxygen samples were taken from alternate sample points to validate the samples taken from the Condensate Storage Tank, c) a work order was initiated to inspect the floating seal on the Condensate Storage Tank, d) a work order was initiated to rework the mechanical seal on 1-CO-P-233 (Condensate transfer pump), and e) the affected Condensate system piping was inspected with 1-CO-P-233 shutdown for potential leakage from the mechanical joints. Based on troubleshooting results and samples taken by the Chemistry Department, the Condensate transfer pump1-CO-P-233 was identified as the most likely source for the air in-leakage. Following repair of the mechanical seal on 1-CO-P-233, the Condensate Storage Tank chemistry returned to within Chemistry Program's specifications. This example demonstrates that the Seabrook Station program is able to detect adverse changes in chemistry parameters, identify the source of the condition by the current monitoring and troubleshooting techniques, and correct the condition to return the chemistry parameters to acceptable levels.

2. During Refueling Outage 9 (Fall of 2003) Steam Generator Sludge Analysis results indicated that low or less than detectable concentration of contaminants (e.g. chloride, sulfate, and fluoride) and sulfur were detected by bulk deposit analysis. This detection confirmed that chemistry control was within EPRI Secondary Chemistry Control Guidelines. This example demonstrates the effectiveness of the Secondary Chemistry Control program.

3. In May 2005, during the planned down power for the "A" Main Feedwater Pump lubricating oil strainer work, the control room was notified (at 85% power) that the sodium levels in the Steam Generators were elevated. The Chemistry Department was asked to perform a backup analysis of the Steam Generators and the down power was stopped at the request of the Chemistry Department Supervisor.

   During the down power, the "C" Steam Generator exceeded action level 2 for sodium at approximately 61 ppb. The down power was stopped to determine the reason for the elevated sodium levels and develop a course of action. The reason for the sodium increase was determined to be due to hideout return of the sodium deposited in the Steam Generator crevices from the Moisture Separator Reheater replacement that occurred during Refueling Outage 10 (Spring of 2005). The down power was continued at a rate of 5% per hour after clearing action level 2 to allow continued cleanup of the Steam Generators. Upon completion of the Main Feedwater Pump filter swap, the plant was held at approximately 55% power until all four Steam Generators had cleared action level 1 for sodium. The plant then was returned to full power. The Chemistry Department's discovery of the hideout return was determined to be beneficial in that it allowed cleanup of sodium from the crevices of the Steam Generators. This example demonstrates that the Seabrook Station Water Chemistry program is able to detect and identify the source of the small excursions using the current monitoring and troubleshooting techniques.

4. In May 2005, when placing the Condensate Polishing System demineralizer CPS-DM-35C into service, the Condensate system oxygen level spiked up to 226 ppb by local analyzer indication. Condensate system oxygen level exceeded the action level 2 value of 30 ppb for approximately 38 minutes.

   Chemistry trouble shooting activities identified that the cause of the oxygen spike was inadequate venting of the cation bed after the resin transfer and prior to placing the standby cation bed on recycle. Due to the steps required to ensure complete resin transfer from the cation storage tank (CPS-TK-278) to any of the service vessels (CPS-DM-35 A/B/C), the water in the standby vessel is saturated with air. It was determined that when the Condensate Polisher is first started to support plant startup after an outage, oxygen is not an issue when the condenser vacuum is broken. However, once vacuum is established and the plant is in Mode 1, oxygen intrusion from the Condensate Polishing System directly impacts Condensate system oxygen levels.

   As part of the evaluation of the condition, it was determined that oxygen will be an issue after a resin charge is transferred into its service vessel. The transfer process will introduce air into a service vessel, especially the air mix step to ensure that mixed bed resin is properly mixed in the service vessel. Subsequently, certain steps in the procedures were identified as points in the resin transfer process where oxygen monitoring can occur. Accordingly, procedure changes were implemented into Chemistry and Operations procedures to support oxygen monitoring and to prevent recurrence of this issue.

   Although the Condensate Polishing system is not within the scope of License Renewal, this example demonstrates that the Seabrook Station Chemistry program is able to detect and identify the source of the small excursions and take corrective actions to prevent recurrence using the current monitoring and troubleshooting techniques.

5. In November 2009, a Nuclear Oversight Audit was performed on the Seabrook Station's Chemistry Control Program. The audit included a review

of the Primary and Secondary Chemistry Control Programs. The audit results were as follows:

Primary Water Chemistry:

The audit reviewed Chemistry Department sample results for the following five primary systems:

- Reactor Coolant System
- Reactor Makeup Water Storage Tank
- Boric Acid Storage Tanks
- Safety Injection Accumulators
- Pressurizer Liquid

The audit found that the sample data met the sampling frequency and requirements of the Seabrook Station Chemistry Manual and EPRI Guidelines.

The audit also found that the Chemistry Department, with coordination and support from the Operations Department, continued to maintain control of Reactor Coolant system lithium within tight limits, which results in a stable pH and low Reactor Coolant system corrosion rates.

The audit also reviewed the Chemistry Department efforts to reduce the transport of silica from the Spent Fuel Pool, Reactor Coolant system, and the Refueling Water Storage Tank. The audit determined that Seabrook Station is effectively addressing the Boraflex deterioration in the Spent Fuel Pool, which results in the release of silica to the Reactor Coolant System and Refueling Water Storage Tank during refueling outages. The addition and use of the Silica Removal Skid (CBS-SKD-161) installed during cycle 13, which employs the use of reverse osmosis to remove silica from the Refueling Water Storage Tank, was determined by the audit to have been effectively implemented.

The audit concluded that Primary Chemistry Control met the requirements of the Seabrook Station Chemistry Manual and EPRI Primary Chemistry Control Guidelines.

Secondary Water Chemistry:

The audit reviewed the Chemistry Department sample results for the following six secondary systems:

- Steam Generator Blowdown
- Feedwater System
- Condensate System
- Demineralized Water Storage Tank

- Auxiliary Demineralized Water Storage Tank
- Condensate Storage Tank

The audit found that the sampling frequency and analytical results met the requirements of the Seabrook Station Chemistry Manual and EPRI Secondary Chemistry Control Guidelines.

The audit concluded that the Seabrook Station Chemistry Program minimized corrosion product transport and scale formation in Steam Generators to prevent fouling and additional risk of stress corrosion cracking by controlling secondary system pH. The audit also concluded that volatile amines including Hydrazine ($N_2H_4$), Methoxypropylamine (MPA) and Ethanolamine (ETA) were added as necessary to control system pH.

The audit concluded that the Secondary Plant Chemistry control was satisfactory.

The operating experiences discussed above include examples of abnormal transients that were identified by routine monitoring activities and corrective actions that were put in place to correct or prevent reoccurrence of such transients in the future. Assessments of the Water Chemistry program are performed to identify the areas that need improvement to maintain the effective performance of the program.

**Conclusion**

The Seabrook Station Water Chemistry Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.3 REACTOR HEAD CLOSURE STUDS

**Program Description**

The Seabrook Station Reactor Head Closure Studs Program is an existing program that manages the aging effects of cracking and loss of material in the Reactor Vessel flange stud hole threads, reactor head closure studs, nuts, and washers per the requirements of ASME Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components"*. The Seabrook Station program implements the requirements of ASME Section XI code as described in the Seabrook Station ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. Seabrook Station implements the guidance outlined in Regulatory Guidance (RG) 1.65, *"Material and Inspection for*

*Reactor Vessel Closure Studs"*, for preventive measures. These preventive measures include material selection, appropriate coatings, and lubricants.

Potential cracking and loss of material in Reactor Vessel flange stud hole threads, reactor head closure studs, nuts, and washers are detected through visual or volumetric examinations in accordance with ASME Section XI requirements. These inspections are conducted during refueling outages while the vessel studs are removed. Studs, nuts, and washers are stored in protective racks after removal. Reactor Vessel flange holes are plugged with water tight plugs during cavity flooding. These methods assure the holes, studs, nuts, and washers are protected from borated water during cavity flooding.

The reactor flange and head are sealed by two metallic O-rings. Leak-off connections are provided between the O-rings and beyond the outer O-ring. Reactor Vessel flange leakage is monitored prior to reactor startup during Reactor Coolant System pressure testing each refueling outage. Following reactor startup, any flange leakage is piped to the Reactor Coolant Drain Tank. A high temperature indication in the piping indicates Reactor Coolant leakage.

Seabrook Station follows ASME Subsection IWB, Examination Category B-G-1 for pressure retaining bolting greater than 2 inches in diameter. The appropriate examinations, as specified in ASME Section XI, Table IWB-2500-1 and Code Case N307-3, *"Ultrasonic Examination of Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1 Section XI, Division 1,"* are used to manage the aging effects of loss of material due to corrosion or wear, crack initiation and crack growth due to stress corrosion and intergranular stress corrosion cracking on the reactor head closure components.

Seabrook Station uses ASME Section XI Table IWB 2500, Examination Category B-P visual (VT-2) inspections during pressure testing to detect leakage from the Reactor Vessel head to vessel interface.

Seabrook Station implements the guidance outlined in RG 1.65 for preventive measures. These preventive measures include material selection and use of appropriate coatings and lubricants. Seabrook Station has 54 reactor head closure studs and 54 spare studs. All are manufactured from SA-540, Class 3, Grade B24 material UFSAR Table 5.2-2). The maximum tensile strength is less than 170 ksi (UFSAR Section 1.8). The reactor head closure studs are coated with an anti-galling compound (PlasmaBond) and a station approved lubricant is utilized during installation/removal of the studs that do not contain molybdenum disulfide ($MoS_2$).

The Seabrook Reactor Head Closure Studs Program schedules and performs inspections to insure that degradation of vessel flange stud hole threads, closure studs, nuts, and washers is discovered before loss of intended function. The program utilizes visual and volumetric examinations in accordance with the general requirements of Subsection IWA-2000 of ASME Section XI to detect the presence of surface discontinuities, flaws, cracking and loss of material by corrosion or wear. The frequency of the inspections is in accordance with the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-G-1. Seabrook Station does not have bushings on the reactor head studs.

Reactor closure head studs are removed from the Reactor Vessel each refueling outage. ASME Section XI Inservice Inspection examinations are performed with the studs removed and consist of a volumetric examination in accordance with Code Case N-307-3. The Code Case states that when conducting ultrasonic examinations from the end of the stud or from the center-drilled hole to satisfy the requirements of Table IWB-2500-1, the surface examination requirement of the table (Item No. B6.30) may be eliminated. The Reactor Vessel threads in the flange (Table IWB-2500-1 Item B6.40) are inspected by volumetric examination. The closure head nuts and washers are inspected by visual, VT-1, examination.

The Seabrook Station inspection schedule provides for timely detection of cracks, loss of material, and leakage. The program complies with the schedule requirements of ASME Section XI, IWB-2400, Inspection Program B, and the extent and frequency of Table IWB-2500-1.

The Seabrook Station program requires that any indication or relevant condition of degradation in the Reactor Vessel flange stud holes or the closure stud bolting is evaluated in accordance with IWB-3100 by comparing the inspection results with the acceptance standards of IWB-3400 and IWB-3500.

Repairs and replacements are performed in accordance with the requirements of ASME Section XI as identified in Seabrook Station ASME Section XI Repair and Replacement Program and the material and inspection guidance of RG 1.65. NUREG-1801 specifies that repair and replacement should be performed in accordance with the requirements of IWB-4000 and IWB-7000, respectively. Sections IWB-4000 and IWB-7000 no longer exist in ASME Section XI. The applicable repair and replacement guidance in the 1995 edition of ASME Section XI is contained in Section IWA-4000. The Seabrook Station does not consider this to be an exception to NUREG-1801.

**NUREG-1801 Consistency**

NUREG-1801, Rev 1, discusses the use of the 2001 edition including the 2002 and 2003 addenda of ASME Section XI code, but allows use of other editions of the ASME Code as long as there is justification. The Seabrook Station Inservice Inspection Program Plan for the second ten-year inspection interval effective from August 19, 2000 through August 18, 2010, approved per 10 CFR 50.55a, is based on the 1995 edition including the 1996 addenda. The next and subsequent 120-month inspection intervals for Seabrook Station will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

This program, with the exception noted below, is consistent with NUREG-1801 XI.M3.

**Exceptions to NUREG-1801**

1. NUREG-1801 XI.M3 states *"Components are examined and tested as specified in Table IWB-2500-1. Examination category B-G-1 for pressure-retaining bolting greater than 2 in. diameter in reactor vessels specifies volumetric examination of studs in place, from the top of the nut to the bottom of the flange hole, and surface and volumetric examination of studs when removed":*

   At Seabrook Station, the reactor closure head studs are removed from the Reactor Vessel during each refueling outage. ASME Section XI Inservice Inspections are performed with the studs removed and consist of a volumetric examination only as allowed by Code Case N-307-3, *"Ultrasonic Examination of Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1 Section XI, Division 1"*, and current version of the ASME Section XI Code.

   <u>Justification for the Exception</u>

   The Generic Aging Lessons Learned (GALL) aging management program inspection requirements for Reactor Head Closure Studs are based the ASME Section XI Code requirements. NUREG 1801, Revision 0 requirements were based on Table IWB 2500-1 of the 1995 Edition through the 1996 addenda of the ASME Code, Section XI. This code edition included requirements for surface and volumetric examination of reactor head closure studs when removed. The later version of the code now endorsed by NUREG 1801, Revision 1 program (2001 edition including the 2002 and 2003 addenda) has been updated to include the Code Case N-307-3 allowance not to require surface examination when studs are removed. However, the detection of aging effects section of

NUREG 1801 Revision 1 for this program still makes reference to the "*surface*" examination as a requirement from the 1995 edition with 1996 addenda of the ASME Section XI Code.

The Seabrook Station Inservice Inspection Program Plan for the second ten-year inspection interval effective from August 19, 2000 through August 18, 2010, approved per 10 CFR 50.55a, is based on the 1995 edition including the 1996 addenda of the ASME Section XI Code including the provisions of Code Case N-307-3. As allowed under the Code Case, Seabrook Station no longer performs surface examinations and only performs volumetric examination of the reactor head closure studs when removed.

*Program Elements Affected: Element 4 (Detection of Aging Effects).*

**Enhancements**

None

**Operating Experience**

Review of plant-specific operating experience has not revealed any cases of cracking or wear with the Seabrook Station Reactor Vessel studs, nuts, flange stud holes, or washers.

The Inservice Inspection Program at Seabrook Station is updated to account for industry operating experience. ASME Section XI is also revised every three years and addenda issued in the interim, which allows the code to be updated to reflect operating experience. The requirement to update the Inservice Inspection Program to reference more recent editions of ASME Section XI at the end of each inspection interval ensures the Inservice Inspection Program reflects enhancements due to operating experience that have been incorporated into ASME Section XI.

1. During Refueling Outage 5 (Spring of 1997), Seabrook Station experienced one stuck reactor head closure stud. This stud was cut out and appropriate ASME Section XI repairs/retests completed. The condition was attributed to galling, which prompted an investigation into suitable anti-galling compounds. In 2000, a spare set of reactor head closure studs were coated with a nickel-silver palladium anti-galling compound. This coating process was initially developed by Westinghouse and Texas Utilities under the name of PlasmaBond (formerly known as Maglon). The coating process has been successfully used at other nuclear power plants. The PlasmaBond process was qualified for use at Seabrook Station by an Engineering Change document. A pre-service inspection of these studs (ultrasonic and magnetic particle testing) was performed in accordance with

ASME Section XI prior to coating. During Refueling Outage 7 (Fall of 2000), the Seabrook Station reactor head closure studs were replaced with the PlasmaBond coated studs. The studs removed were also coated using the PlasmaBond process and stored as spares. These spare reactor head closure studs were installed during Refueling Outage 13 (Fall of 2009) as part of a periodic replacement of the PlasmaBond coated reactor head studs. This example demonstrates that the Plasmabond coating has been successful as an anti-galling treatment and has not adversely affected the stud function.

2. During Refueling Outage 8 (Spring of 2002), during Reactor Vessel disassembly, a condition report was generated when the workers reported difficulty in removing the reactor head closure studs compared to previous outages. Discoloration was also reported on some of the studs. Subsequent inspections did not indicate any thread damage on any of the studs. The discoloration on the PlasmaBond coating was determined to be the lubricant used for stud removal and was considered to be normal and not an indication of a degraded condition. Further evaluation and resolution of this issue resulted in: a) purchase of a new stud removal tool, b) revision to the Maintenance procedures to provide better direction to ensure that the studs are protected from damage during installation and removal, and c) development of a new maintenance procedure for inspection of the PlasmaBond coating on the reactor closure head studs.

3. During Refueling Outage 10 (Spring of 2005), while performing verification of the final elongation values of the reactor head studs, a condition report was initiated, which identified that Stud No. 30 was out of specified elongation range by 0.002 inches. The acceptance values for final elongation of the studs post tensioning were from 0.049 inches to 0.056 inches. Stud No. 30 was found to be tensioned to an elongation of 0.047 inches. Subsequently, an engineering evaluation was performed. This evaluation concluded that the preload induced by the post tensioned final elongation value of 0.047 inches for Stud No. 30 was more than adequate to carry the Reactor Vessel pressure design loads. This condition did not alter the original design intent or the function of the stud to maintain Reactor Coolant system pressure boundary and structural integrity of the reactor head and flange connection for all service conditions.

4. The Seabrook Station Reactor Vessel studs, nuts and washers are 100% ultrasonically inspected once every Inservice Inspection ten year interval. The ultrasonic inspections of the 54 studs were divided up between the scheduled refueling outages during the ten year interval. During Refueling Outage 13 (Fall of 2009), which was the last scheduled outage for the second ten year interval, the final 9 of the 54 studs were ultrasonically

inspected with acceptable results. No unacceptable results have been identified through these inspections.

The operating experience of the Reactor Head Closure Studs Program shows that there are no signs of age related degradation. The above examples provide objective evidence that any anomalies or deficiencies are entered into the corrective action process, the conditions are evaluated, and corrective actions are taken when necessary to prevent recurrence.

**Conclusion**

The Seabrook Station Reactor Head Closure Studs Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### B.2.1.4 BORIC ACID CORROSION

**Program Description**

The Seabrook Station Boric Acid Corrosion Program is an existing program that manages the aging effects of loss of material in mechanical, electrical, and structural components due to leakage from systems containing borated water. The Seabrook Station Boric Acid Corrosion Program implements the recommendations of NRC Generic Letter (GL) 88-05, *"Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"*. Degradation of the component due to boric acid corrosion can not occur without leakage of borated water. Therefore, the program requires periodic visual inspection of all systems within the scope of license renewal that contain borated water for evidence of leakage, accumulations of dried boric acid, or boric acid wastage. The scope of this program includes sources of borated water leakage that are outside the scope of GL 88-05 but are in proximity to structures and components that are subject to aging management review. The program provides for visual inspections and early discovery of borated water leaks such that mechanical, electrical, and structural components that may be contacted by leaking borated water will not be adversely affected or their intended functions impaired.

The Boric Acid Corrosion Program includes provisions for identification of components exhibiting boric acid accumulations or leakage, evaluation of the acceptability for continued service of components exhibiting boric acid accumulations or leakage, trending and tracking of previously identified leaks or boric acid accumulations, and taking appropriate corrective actions.

As part of the Boric Acid Corrosion Program, Seabrook Station monitors operating experience related to boric acid leaks and takes appropriate corrective actions. By conducting visual inspections, locating the source of the leaks when they are discovered, performing engineering evaluations, and reviewing internal and external operating experience, the program ensures that Structures, Systems and Components (SSCs) within the scope of license renewal will continue to perform their intended functions.

Loss of material by boric acid wastage is also discussed for the Reactor Vessel (RV) head by the Nickel-Alloy Penetration Nozzles Welded to the Upper RV Closure Heads of PWRs Program, B.2.1.5, and for the Reactor Head closure studs by the Reactor Head Closure Studs Program, B.2.1.3.

The Seabrook Station Boric Acid Corrosion program includes requirements for:

a. monitoring of borated water systems for boric acid leakage,

b. visual inspections of mechanical, electrical, and structural component surfaces that are potentially exposed to borated water leakage,

c. inspections for boric acid leakage during pressure testing,

d. timely discovery of the principal location of the leak, the leakage pathway, and extent of condition,

e. removal of boric acid residue,

f. assessment of the corrosion and evaluation of the effects of leakage and corrosion on components in a timely manner to maintain component integrity, and

g. follow-up inspection for adequacy of corrective actions.

Preventive actions include improving maintenance practices such as revising the valve packing program to improve packing design and techniques, performance of periodic walk downs and leakage surveillances to identify components that may require corrective maintenance, and monitoring locations where potential leakage could occur. Timely repair of detected leakage prevents or mitigates boric acid corrosion, and is accomplished in accordance with the program's corrective action process.

The program relies on visual inspections conducted during normal plant operation and when the plant is shutdown for refueling. Visual inspections include both focused inspections and observations by plant personnel during normal operational activities. Personnel in the plant look for boric acid residue as a white crystal-like substance or any discoloration or moisture. The program follows the guidelines in NRC GL 88-05 and provides for timely detection of leakage during pressure testing or by observance of boric acid crystal deposits during plant walkdowns and maintenance.

The Seabrook Station program includes trending of boric acid leaks and status for adverse conditions.

The Seabrook Station program acceptance criterion requires corrective actions and/or further evaluation if any leakage or residue of boric acid is observed. The Seabrook Station maintenance procedure for cleaning and inspection of components subjected to boric acid leakage provides direction for performing initial screening, inspection, cleaning and follow-up inspection of components identified as boric acid leaking components. Engineering evaluation of boric acid leakage effects on structures and components are performed per a Seabrook Station procedure to ensure that the intended functions of the affected structures and components remains consistent with the design basis.

When boric acid leaks are discovered, they are entered into the Corrective Action Program, evaluated, and are corrected through the corrective maintenance process. This approach ensures that identified problems are corrected and that component aging related to boric acid corrosion is effectively managed.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M10.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Industry operating experience indicates that boric acid leaks can cause significant corrosion damage to susceptible plant structures and components. Program effectiveness reviews and self assessments of the boric acid corrosion program identify the areas that need improvement to maintain the quality of the program. Performance indicators for the Boric Acid Corrosion Program show that the program is compliant with existing regulations and will be able to manage boric acid corrosion during the period of extended operation.

1. In August 2005, Institute of Nuclear Power Operations (INPO) conducted a Primary Systems Review at Seabrook Station. This review noted a large boric acid leakage backlog of work orders related to boric acid leakage. Subsequent INPO mid-term and scheduled evaluation and assessment

visits noted a continuing growth in the backlog. Subsequently, Seabrook Station conducted benchmarking visits of INPO-recommended programs. These visits identified common methods that had proved effective in addressing boric acid leaks. Many of these methods were incorporated into the Seabrook Station Boric Acid Corrosion Program. These changes reduced the significant work order backlog from several hundred to less than twenty within one operating cycle.

2. Several Seabrook Station condition reports document the effectiveness of the Boric Acid Corrosion Program. Through field observations, the condition reports document that when a boric acid deposit is observed, it is promptly reported, evaluated, and repaired. One example is the discovery of boric acid residue on a Chemical and Volume Control system valve (CS-V-158) during a plant walkdown in 2008. The valve was categorized as an active leak per the Boric Acid Corrosion Program. The boric acid leakage was evaluated and the packing leak was repaired to eliminate the leakage. Another example is the discovery of medium boric acid buildup on a Residual Heat Removal system valve (RH-V-15), in 2005, during ASME Section XI Inservice Inspection of the Residual Heat Removal system. The dry boric acid build up was at the body to bonnet joint on the valve. The boric acid leakage was evaluated per the Seabrook Station Boric Acid Corrosion Program and the body to bonnet gasket was replaced to eliminate the leakage.

3. In November 2008, a self-assessment of the Boric Acid Corrosion Program was performed. This self-assessment compared the Seabrook Station Boric Acid Corrosion Program against the current industry guidance document, which is WCAP-15988-NP, Revision 1, *"Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors"*. This WCAP was issued in February 2005 and identifies potential enhancements to the Boric Acid Corrosion Programs described in the utility responses to the GL 88-05.

   Using the eleven Westinghouse Commercial Atomic Power (WCAP) objectives, the Seabrook Station Boric Acid Corrosion Program was assessed. This self-assessment identified the process as operationally sound and in a mode of continuous improvement. Enhancements were identified that would more closely align the Seabrook Station Boric Acid Corrosion Program to the guidance document. The self-assessment identified that the Seabrook Boric Acid Corrosion Program could benefit by being more prescriptive in addressing some of the WCAP objectives. Four condition reports were generated during the course of this Self Assessment. None of these condition reports identified programmatic failures and all were directed towards future boric acid program enhancements.

These examples provide objective evidence that appropriate guidance exists for identification, evaluation, and repair/replacement of locations where boric acid deposits are observed. Assessments of the Boric Acid Corrosion Program are performed to identify the areas that need improvement to maintain the effective performance of the program.

**Conclusion**

The Seabrook Station Boric Acid Corrosion Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis during the period of extended operation.

### B.2.1.5 NICKEL-ALLOY PENETRATION NOZZLES WELDED TO THE UPPER REACTOR VESSEL CLOSURE HEADS OF PRESSURIZED WATER REACTORS

**Program Description**

The Seabrook Station Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program is an existing program that is part of the Inservice Inspection program and manages the aging effects of cracking due to primary water stress corrosion cracking in the reactor coolant environment. This program was established to ensure that augmented In-Service Inspections of all nickel-alloy penetration nozzles welded to the upper Reactor Vessel head will continue to be performed as mandated by the latest requirements. The original program requirements were contained in NRC Order EA-03-009, *"Issuance of Order Establishing Interim Inspection Requirements for Reactor Vessel Heads at Pressurized Water Reactors"*, as amended by the First Revision of the Order. The GALL program incorporates any subsequent NRC requirements that may be established to supersede the requirements of NRC Order EA-03-009. On September 10, 2008, the NRC revoked Order EA-3-009 and replaced it with ASME Code Case N-729-1, "*Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1, Supp 4,*" as modified in 10 CFR 50.55a (g)(6)(ii)(D). Therefore, the NRC requirements contained in 10 CFR 50.55a (g)(6)(ii)(D) are the currently applicable NUREG-1801 aging management program for nickel-alloy penetration nozzles welded to the upper RV closure heads of PWRs.

The program is focused on managing the effects of cracking due to primary water stress corrosion cracking of the nickel-alloy used in the fabrication of the upper vessel head penetration nozzles.

The program monitors for cracking due to primary water stress corrosion cracking and loss of material due to boric acid corrosion in the upper vessel head penetration nozzles to ensure that flaw indications are detected prior to loss of their intended safety function and prior to any challenge to the structural integrity of the nozzles. The program also monitors for evidence of Reactor Coolant system leakage as a result of through-wall cracks that may exist in the upper vessel head penetration nozzles or their associated partial penetration J-groove welds. The Seabrook Station inspections include bare metal visual inspections of 100% of the Reactor Vessel head surface and ultrasonic testing of each Reactor Vessel head penetration nozzle on a frequency prescribed by ASME Code Case N-729-1, as modified in 10 CFR 50.55a.

Inspections in accordance with EA-03-009 or the First Revision to the Order have been completed during Refueling Outage 8 (Spring of 2002), Refueling Outage 9 (Fall of 2003), and Refueling Outage 11 (Fall of 2006). No degradation of Reactor Vessel head penetration nozzles has been discovered during these inspections. Based on degradation-free inspection results, ASME Code Case ranking and operating schedule, the next inspection is scheduled for Refueling Outage 14 (Spring of 2011), and will be performed in accordance with ASME Code Case N-729-1, as modified in 10 CFR 50.55a (g)(6)(ii)(D).

The Seabrook Station program meets the requirements of the ASME Code Case N-729-1, as modified in 10 CFR 50.55a, and is included as part of the Seabrook Station Reference Manual - RCS Materials Degradation Management Reference. The Seabrook Station Reactor Vessel head penetrations are Alloy 600 and include one (1) penetration for the top head vent and seventy-eight (78) penetrations for control rod drives and instrumentation. Initial evaluations performed in accordance with NRC Order EA-03-009 determined that Seabrook Station fell into the low susceptibility category for primary water stress corrosion cracking. ASME Code Case N-729-1, as modified in 10 CFR 50.55a, requires calculations for Effective Degradation Years (EDY) and Re-Inspection Years (RIY). These numbers are used as a basis for determining inspection frequency. Seabrook Station has a current value for EDY of 3.03 and the RIY value is 0.95 at the end of cycle 13 which ended in the fall of 2009. In accordance with Table 1, including note (4) of ASME Code Case N-729-1, as modified in 10 CFR 50.55a, these values result in an re-inspection frequency for visual exams of every third refueling outage or 5 calendar years, and for volumetric examinations of every 8 calendar years.

The Seabrook Station Water Chemistry Program (B.2.1.2) is credited as a preventive measure to mitigate primary water stress corrosion cracking.

Repair, replacement, and mitigation activities are conducted in accordance with the Seabrook Station ASME Section XI Repair/Replacement program.

If flaw indications attributed to primary water stress corrosion cracking are identified, whether acceptable or not for continued service under paragraphs 3130 or 3140 of ASME Code Case N-729-1, as modified in 10 CFR 50.55a, the re-inspection frequency must be increased to each refueling outage instead of the re-inspection intervals in accordance with Table 1, note (8) of ASME Code Case N-729-1, as modified in 10 CFR 50.55a.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M11A.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Seabrook Station has not detected primary water stress corrosion cracking in the Reactor Vessel head penetration nozzles and J-groove welds of the Reactor Vessel closure head penetrations. Several inspections have been performed and no evidence of degradation of the vessel head penetration nozzles and J-groove welds was found. A summary of these inspection results from the past refueling outages is listed below.

1. During Refueling Outage 8 (Spring of 2002), a bare metal visual inspection of the Reactor Vessel top head was performed. The robotic inspection provided a 360° view of each penetration and adjacent surfaces. No evidence of penetration leakage was observed.

2. During Refueling Outage 9 (Fall of 2003), a control rod drive mechanism canopy seal weld leak was discovered by evidence of boric acid on the Reactor Vessel head flange during Reactor Vessel disassembly. An under insulation inspection was conducted and another canopy seal weld leak was discovered. Subsequently, the boric acid deposits were removed and the Reactor Vessel head was inspected. No underlying corrosion was found. The remaining control rod drive mechanism canopy seal welds were also inspected and no additional leaks were discovered and no evidence of penetration leakage was observed. The evaluation of the condition determined that the canopy seal weld failure mechanism was transgranular stress corrosion cracking due to the presence of halogen species, most likely chloride, in combination with oxygen. These types of canopy seal weld leaks had been previously observed in other PWRs. Approximately forty-six canopy seal weld leaks had been reported in the industry as of

October of 2003. The canopy seal weld is a non-structural weld that provides a seal to prevent Reactor Coolant leakage onto the Reactor Vessel head. During Refueling Outage 9, these two leaks were repaired by the installation of canopy seal clamp assemblies. The canopy seal clamp assembly is a mechanical device that seals the leaking weld by introducing a compressive load into the weld to close the crack and precludes further flaw propagation. This repair developed by Westinghouse is considered a permanent fix.

Follow up inspections were performed during Refueling Outage 10 (Spring of 2005) and again during Refueling Outage 11 (Fall of 2006), and no additional canopy seal weld leaks were identified.

3. During Refueling Outage 11 (Fall of 2006), the Reactor Vessel head inspections were conducted as required by First Revised NRC Order EA-03-009. The inspections included a robotic bare metal visual inspection of each penetration and head surface from the top. A robotic inspection of J-Groove welds and penetration tubes from the underside of the head was performed using ultrasonic and surface examination techniques. No unacceptable indications were discovered.

   The next scheduled bare metal visual inspection of the Reactor Vessel top head penetrations is scheduled for Refueling Outage 14 (Spring of 2011).

The review of operating experience provides objective evidence that the closure head components are in good condition and that the Nickel-Alloy Penetration Nozzles Welded to the Upper RV Closure Heads of PWRs Program is effective in detecting any flaw indications prior to the loss of intended function. Appropriate guidance for evaluation, repair, or replacement is provided for locations where degradation is found.

**Conclusion**

The Seabrook Station Nickel-Alloy Penetration Nozzles Welded to the Upper RV Closure Heads of PWRs Program provides reasonable assurance that the aging effects will be will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### B.2.1.6 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS)

This program is not used at Seabrook Station. The Seabrook Station Reactor Coolant system contains statically cast fittings constructed of SA-351 Grade CF8A material in a service condition greater than 482°F. However, the aging effect in NUREG-1801 for this material and environment combination is not applicable because the molybdenum and ferrite contents for these components are below the industry accepted threshold (<0.5% molybdenum and <20% ferrite). Therefore, loss of fracture toughness due to thermal aging embrittlement is not applicable.

## B.2.1.7 PWR VESSEL INTERNALS

### Program Description

The Seabrook Station PWR Vessel Internals Program is a new program that will manage the aging effects of crack initiation and growth due to irradiation-assisted stress corrosion cracking, primary water stress corrosion cracking and stress corrosion cracking; reduction of fracture toughness due to radiation and thermal embrittlement and void swelling; changes in dimensions due to void swelling; and loss of preload due to stress relaxation, in Reactor Vessel Internals components.

The Seabrook Station PWR Vessel Internals Program will be based on inspection guidance provided in MRP-227 Rev. 0, *"Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"* and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. MRP-227 is based upon industry operating experience, research data, and vendor evaluations. The examination methods, coverage, and schedule prescribed in MRP-227 account for aging experience in both domestic and international Reactor Vessel Internals.

Industry operating experience related to the aging degradation of PWR Vessel Internals is described in the Operating Experience program element. The Seabrook Station PWR Vessel Internals Program will continue to evolve as additional inspection experience is gained through FPL/NextEra participation in EPRI material reliability program activities.

The Seabrook Station PWR Vessel Internals aging management will consist of four major elements: (1) component categorization and aging management strategy development; (2) selection of aging management methodologies for vessel internals that are both appropriate and based on an adequate level of applicable experience; (3) qualification of the recommended methodologies that is based on adequate technical justification; and (4) implementation of the recommendations based on the Industry Initiative for the Management of Materials Issues, NEI Guideline, 03-08.

<u>Program Elements</u>

The following provides the results of the evaluation of each program element against the 10 elements described in Appendix A of NUREG-1800 Rev. 1, *"Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants".*

Element 1 - Scope of Program

The Seabrook Station PWR Vessel Internals Program addresses the management of aging effects of the Seabrook Station Reactor Vessel internals components, both non-bolted and bolted. The Seabrook Station Reactor Vessel internals consist of two basic assemblies, the upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and the lower internals assembly that can be removed, if desired, following a complete core off-load.

The scope does not include fuel assemblies, control rod drive assemblies, nuclear instrumentation, and welded attachments. Fuel assemblies are periodically replaced (i.e., short lived), and therefore, are not subject to aging management review. Control rod drive assemblies are active components and therefore, are not subject to aging management review. Nuclear Instrumentation (i.e., incore neutron flux detectors) are active electrical components, and therefore, are not subject to aging management review. The scope also does not include welded attachments to the Reactor Vessel. Welded attachments to the Reactor Vessel interior are subject to examination in accordance with the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, B.2.1.1. The IWB, IWC, and IWD Program conducts visual inspection of the accessible interior attachment welds per ASME Section XI, Table IWB-2500-1, examination category B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels.

The Seabrook Station PWR Vessel Internals Program will be focused on managing crack initiation and growth due to irradiation-assisted stress corrosion cracking, primary water stress corrosion cracking and stress corrosion cracking; reduction of fracture toughness due to radiation and thermal embrittlement and void swelling; changes in dimensions due to void swelling; and loss of preload due to stress relaxation, in Reactor Vessel internals components. Loss of fracture toughness due to radiation and thermal embrittlement is of consequence only if cracks exist and the local applied stress intensity exceeds the reduced fracture toughness. Cracking, if it occurs, is expected to initiate at the surface and is detectable by the augmented inspections performed under this program.

The Seabrook Station PWR Vessels Internals Program complies with NRC guidance and includes the following commitments:

1. A PWR Vessel Internals Program will be implemented, prior to period of extended operation, as described in this section.

2. An inspection plan for Reactor Vessel Internals will be submitted for NRC review and approval at least twenty-four months prior to entering the period of extended operation.

This submittal will include any necessary revisions to the Seabrook Station PWR Vessel Internals Program, as well as any related changes to the Seabrook Station scoping, screening, and aging management review results for PWR Vessel Internals, to conform to the NRC approved inspection and evaluation guidelines.

MRP-227 inspection and evaluation guidelines were organized around a framework and strategy for managing the effects of aging in PWR internals together with a substantial database of material data and supporting results. This process permitted further categorization of PWR vessel internals into the functional groups that follow:

a. *Primary:* Those PWR vessel internals that are highly susceptible to the effects of at least one of the aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in MRP-227.

b. *Expansion:* Those PWR vessel internals that are highly or moderately susceptible to the effects of at least one of the aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group.

c. *Existing Programs:* Those PWR vessel internals that are susceptible to the effects of at least one of the aging mechanisms, and for which generic and plant-specific existing aging management program requirements are capable of managing those aging effects, were placed in the Existing Programs group.

d. *No Additional Measures:* Those PWR vessel internals for which the effects of all aging mechanisms are below the screening criteria were placed in the No Additional Measures group. No further action is required by MRP-227 for managing the aging of the No Additional Measures components.

This categorization process does not supersede the ASME Section XI Inservice Inspection requirements.

Element 2 - Preventive Actions

The Seabrook Station PWR Vessel Internals Program will be a condition monitoring program and does not include any preventive or mitigative actions. Preventive and mitigative actions for the Reactor Vessel Internals components are established and implemented in accordance with the

Seabrook Station Water Chemistry Program described in B.2.1.2. The Water Chemistry Program manages aging effects by controlling concentrations of known detrimental chemical species, such as chlorides, fluorides, sulfates and dissolved oxygen, below the levels known to cause degradation. The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of water chemistry.

Element 3 - Parameters Monitored/Inspected

The Seabrook Station PWR Vessel Internals Program will monitor the effects of aging related degradation mechanisms on the intended function of Reactor Vessel Internals components through one-time, periodic, and conditional examinations, and other aging management program methodologies, as needed, in accordance with the MRP-227 and ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. This program will be credited for managing cracking due to irradiation-assisted stress corrosion cracking, primary water stress corrosion cracking and stress corrosion cracking; reduction of fracture toughness due to radiation and thermal embrittlement and void swelling; changes in dimensions due to void swelling and loss of preload due to stress relaxation in PWR Vessel Internals components.

The Seabrook Station PWR Vessel Internals Program aging management methodologies will include visual examinations, surface examinations, volumetric examinations, and physical measurements. VT-3 visual examinations detect the general degradation conditions, whereas VT-1 visual and enhanced EVT-1 visual examinations will be conducted to detect discontinuities and imperfections on the surface of components. Surface examinations will characterize discontinuities on the surface of components, and the volumetric inspections will indicate the presence of discontinuities or flaws throughout the volume of material. Aging effects may involve changes in clearances, settings, and physical displacements that will be monitored by visual means or physical measurements.

Once per Inservice Inspection interval, the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program requires a VT-3 visual examination of the Reactor Vessel removable core support structures in accordance with Table IWB-2500-1, Examination Category B-N-3.

The inspection and test techniques prescribed by the ASME Code, Section XI and augmented by MRP-227 are designed to maintain structural integrity by ensuring the aging effects will be detected and corrective actions taken, before the loss of intended function of the Seabrook Station Reactor Vessel Internals.

Element 4 - Detection of Aging Effects

Inspection and evaluation to manage aging of Reactor Vessel Internals will consist of the following:

a. selection of items for aging management

b. selection of the type of examination or other methodologies appropriate for each applicable degradation mechanism

c. specification of the required level of examination qualification

d. schedule of first examination and frequency of any subsequent examinations

e. sampling and coverage

f. expansion of scope if sufficient evidence of degradation is observed;

g. examination acceptance criteria

h. methods for evaluating examination results not meeting the examination acceptance criteria

i. updating the program based on industry-wide results, and

j. contingency measures to repair, replace, or mitigate

The Seabrook Station PWR Vessel Internals Program will use visual inspection, surface examination and volumetric inspection techniques to manage the applicable effects of aging. Aging management methodologies described in MRP-227 are based on either existing inservice examinations required by the ASME Code, Section XI or on well-documented and well-demonstrated examination methods with which the industry has considerable experience. The inspection techniques are described below:

a. VT-3 visual examinations will be conducted to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, or corrosion; and by identifying conditions that could affect operational or functional adequacy of components. VT-3 visual examinations of internals will be conducted using remote examination techniques, due to personnel radiation exposure issues.

b. The VT-1 visual examination and the enhanced EVT-1 visual examination will be utilized where a greater degree of detection capability than VT-3 visual examination is needed to manage the aging effect. Unlike the detection of general degradation conditions by VT-3 visual examination, VT-1 visual and enhanced EVT-1 visual examinations will be conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, corrosion, or erosion.

c. Surface examination may be used to supplement either VT-3 visual or VT-1/EVT-1 visual examinations, in order to further characterize discontinuities on the surface of components. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface discontinuities. Surface examinations may be conducted using the eddy current (ET) inspection method.

d. An ultrasonic examination will be utilized where visual or surface examination is unable to detect the effect of the age-related degradation for some PWR vessel internals. The ultrasonic examination detects the presence of discontinuities or flaws throughout the volume of material.

e. Physical measurements, in some cases, can manage the loss of preload or clamping force caused by thermal and irradiation-enhanced stress relaxation, and excessive distortion or deflection caused by void swelling. These aging effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements that characterize the magnitude of the effects. This methodology may be used in conjunction with VT-3 visual examination, which includes verifying parameters, such as clearances, settings, and physical displacements.

Components designated as having no significant aging effects will require no additional measures for future inspections other than the ASME Code inspections per Section XI, Examination Category B-N-3, for removable internal structures. Once per Inservice Inspection interval the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program conducts a VT-3 visual examination of the Reactor Vessel removable core support structures under Table IWB-2500-1, Examination Category B-N-3.

Documentation of inspection results associated with the PWR Vessel Internals Program will be in accordance with approved inspection procedures. All necessary program implementing documents, including inspection procedures, will be developed in accordance with the FPL/NextEra Energy Quality Assurance Program which implements the requirement of 10 CFR Part 50, Appendix B. The inspections will be performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50, Appendix B.

Examination methods, coverage, and schedule of the inspections prescribed by ASME Section XI and MRP-227 are intended to maintain structural integrity by ensuring the aging effects will be detected and corrective actions taken, before the loss of intended function of the Seabrook Station Reactor Vessel Internals.

FPL/NextEra Energy participates in the industry programs for investigating and managing aging effects on reactor internals. The program will implement applicable results of the industry programs.

Element 5 - Monitoring and Trending

Implementation of one-time, periodic, and conditional examinations and other aging management methodologies, scheduled in accordance with the ASME Section XI and MRP-227 will provide timely detection of aging effects. In addition to the primary components, program expansion components have been defined should the scope of examination and re-examination need to be expanded beyond the primary group due to detection of significant aging effects. Any flaw indications detected during the required examinations will be dispositioned in accordance with the acceptance criteria and corrective actions program elements that follow.

Element 6 - Acceptance Criteria

Seabrook Station PWR Vessel Internals Program inspections, indications and relevant conditions detected during examination will be evaluated in accordance with ASME Section XI, Article IWB-3500. MRP-227 provides additional information for the examination acceptance criteria for the primary and expansion components. The criteria for expanding the examinations beyond the Primary Components will include the Expansion Components. The examination acceptance criteria will include: (i) specific, descriptive relevant conditions for the VT-3 visual examinations; (ii) requirements for recording and dispositioning surface breaking indications that are detected and sized for length by the VT-1/EVT-1 visual examinations; and (iii) requirements for system-level assessment of bolted or pinned assemblies with volumetric (UT) examination indications that exceed specified limits.

Detected conditions that do not satisfy these examination acceptance criteria will be dispositioned using the Seabrook Station Corrective Action Program. The acceptance criteria, for any indications, will document that the component intended functions will be maintained during the period of extended operation.

Element 7 - Corrective Actions

All indications will be evaluated per the acceptance criteria. Unacceptable indications will be corrected through implementation of appropriate repair or replacement activities.

Indications noted will be entered into the Seabrook Station Corrective Action Program for appropriate disposition. A repair, replacement, or evaluation will be performed for all flaws that exceed the acceptance standards. Additional guidance for disposition of unacceptable conditions for reactor vessel internals will be found in the ASME Code, Section XI; in MRP-227 Guidelines; and in reports referenced therein or demonstrated through an appropriate technical justification. MRP-227 provides information on methodology that will be used for the evaluation of detected conditions that exceed the examination acceptance criteria. The flaw evaluation methodology accounts for the accumulated neutron exposure and the resulting loss of fracture toughness due to radiation embrittlement in assessing the suitability of the component for continued service. Justification for flaw evaluation fracture toughness limits is provided in Section 6 of MRP-227.

Repair or replacement activities comply with ASME Section XI as invoked by 10 CFR 50.55a or approved ASME Code Cases as referenced in the latest version of NRC Regulatory Guide 1.147, "*Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1.*" Proposed alternative repair/replacement activities, if any, will be submitted to the NRC for review and approval in accordance with 10 CFR 50.55a(a)(3)(i) or 10 CFR 50.55a(a)(3)(ii).

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 7 Corrective Actions.

Element 8 - Confirmation Process

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 8 Confirmation Process.

Element 9 - Administrative Controls

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 9 Administrative Controls.

Element 10 - Operating Experience

As discussed in MRP-227, the PWR Vessel Internals aging degradation has been observed in foreign PWRs, with emphasis on cracking of baffle-former bolting. Because of the foreign operating experience, the U.S. PWR owners and operators began a program a decade ago to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. To support future inspections and evaluations the industry also began substantial laboratory testing projects in order to gather the materials data necessary.

The Seabrook Station PWR Vessel Internals Program is a new program to be implemented prior to the period of extended operation. A review of plant specific operating experience related to the Seabrook Station Reactor Vessel Internals identified no aging management issues. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, Table IWB-2500-1, Examination Category B-N-3 inspections of the Seabrook Station Reactor Vessel Internals, have not identified any unacceptable indications.

The EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, which forms the basis of the Seabrook Station PWR Vessel Internals Program, is based upon industry operating experience, research data, and vendor evaluations. Development of the program relies on the consensus review and inputs of the MRP Reactor Internals Core and Focus Groups, which include representatives from utilities, research scientists, and vendors. This program will continue to evolve as additional experience is gained. Reactor Vessel internals failures, both domestic and foreign, have been considered in the development of MRP-227.

FPL participates in the industry programs for investigating and managing aging effects on PWR Vessel Internals. Through its participation in EPRI MRP activities, FPL and Seabrook Station will continue to benefit from the reporting of PWR Vessel Internals inspection information, and will share its own internals inspection results with the industry, as appropriate. The Seabrook Station PWR Vessel Internals Program will implement applicable results of the industry programs.

**Exceptions to NUREG-1800**

None

**Enhancements**

None

**Conclusion**

The Seabrook Station PWR Vessel Internals Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.8 FLOW-ACCELERATED CORROSION

**Program Description**

The Seabrook Station Flow-Accelerated Corrosion (FAC) Program is an existing program that manages the aging effects of loss of material due to wall thinning on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, tees, expanders, and valve bodies which contain high energy fluids (both single phase and two phase flow). The program is based on the EPRI guideline NSAC-202L-R2, *"Recommendations for an Effective Flow Accelerated Corrosion Program"* and uses Chexal Horowitz Engineering/ Corrosion Workstation (CHECWORKS) as a predictive tool. Included in the program are: (a) an analysis to determine FAC susceptible lines, (b) performance of baseline inspections, (c) follow-up inspections to confirm the predictions, and (d) repairing or replacing components, as necessary.

The Seabrook Station FAC Program includes component susceptibility determination, inspection requirements, acceptance criteria, repair and replacement criteria, expansion criteria, and reporting requirements.

The Seabrook Station FAC Program is an analysis, inspection and verification program; thus, there is no preventive action.

The Seabrook Station Water Chemistry program, B.2.1.2, is utilized to control pH and dissolved oxygen. When the FAC Program identifies components to be repaired or replaced, the use of FAC resistant materials and or configuration changes are considered.

This aging management program monitors the aging effects of flow-accelerated corrosion on the intended function of piping and components by measuring wall thickness using non-destructive examination and performing analytical evaluations.

Components are inspected for wall thinning due to flow-accelerated corrosion using ultrasonic or radiography examinations. Ultrasonic examination provides more complete data for measuring the remaining wall thickness. As described in the EPRI Recommendations for an Effective Flow Accelerated Corrosion Program, radiography is commonly used on small-bore piping because it can be performed without removing pipe insulation and during plant operation with components in service. Evaluation of the results is performed by FAC engineers and is not used to identify other mechanisms (i.e., cracking or weld indications). Valves, orifices, equipment nozzles, and other like components that cannot be inspected completely with ultrasonic examinations due to their shape and thickness are evaluated based on the wear of piping located immediately downstream. Analytical models developed with computer programs, including CHECWORKS, are used to predict locations that are susceptible to flow-accelerated corrosion in piping systems based on specific plant data including material, configuration, hydrodynamic conditions, and operating conditions.

The FAC inspection schedule is developed based on the CHECWORKS predictive code, re-examinations from previous outages, and plant specific and industry operating experience. The predicted extent of wall thinning for susceptible components is updated after each refueling outage. The component examination data is compiled and maintained with identification of the specific outage or time of inspection. The next scheduled inspection is based on the remaining service life recalculated after each inspection. The FAC Program inspection results are used to calculate the number of operating cycles remaining before the component reaches minimum allowable wall thickness. If calculations indicate that an area will reach the minimum wall thickness before the next inspection interval, the component must be repaired, replaced, or re-evaluated.

The CHECWORKS model was revised in January 2005 to reflect plant conditions following the Seabrook Station power uprate. Operating conditions affecting flow accelerated corrosion were updated in CHECWORKS and the predictive model rerun to identify additional areas of susceptibility and any re-ranking of highly-susceptible locations. The current scope and frequency of FAC related inspections are based on those conditions.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M17.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Wall thinning problems in single-phase systems have occurred throughout the industry in Feedwater and Condensate systems, and in two-phase piping in Extraction Steam lines and Moisture Separator Reheater and Feedwater heater drain lines. The FAC Program was originally outlined in NUREG-1344, *"Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants"* (April 1989) and implemented through GL 89-08, *"Erosion/Corrosion-Induced Pipe Wall Thinning"* (May 1989). The Seabrook Station program has evolved through industry experience and is now implemented using the guidelines of EPRI NSAC-202L-R2 and CHECWORKS as a predictive tool. Selection of monitoring locations and inspection methods have improved over time based on industry and plant operating experience.

Seabrook Station has not experienced any failures of piping covered by the FAC program with the exception of a minor leak in a non-safety related small-bore socket welded fitting not modeled in CHECWORKS. Wall loss by flow-accelerated corrosion within a socket welded fitting cannot be found using ultrasonic inspection techniques. Seabrook Station has implemented a radiography inspection program to screen susceptible small-bore piping for potential FAC related degradation. This inspection is focused on areas where personnel hazards may be created in the event of a through wall leak.

Further review of operating experience at Seabrook Station, as discussed below, demonstrates that the FAC Program is effective at detection, correction, and resolution of conditions leading to and resulting from flow accelerated corrosion.

1. Following the initial CHECWORKS implementation in 1991, the Seabrook Station high pressure Extraction Steam piping was determined to be highly susceptible to FAC and several areas were inspected as part of the first FAC inspections. As a result, a significant portion of this piping was found to be degraded and was replaced with a more FAC-resistant material (Chrome-Moly). No failure of this piping occurred prior to replacement and no degradation has been noted in the replacement piping.

2. In September 2006, a self-assessment of the FAC Program was performed. No program weaknesses were identified during the self-assessment. The following were identified as strengths: a) Detailed FAC Susceptibility Analysis, b) Incorporation of power uprate condition into the CHECKWORKS model, c) Conversion of CHECKWORKS to the latest EPRI Steam/Feedwater Application version, and d) Use of operating experience for evaluating applicability to Seabrook Station. Areas for continued program improvement were identified for future inclusion to strengthen the overall program.

   Additionally, the FPL/NextEra Energy Corporate input to the 2006 Seabrook Station FAC Self-Assessment cited the following as examples of appropriate assessment and response to industry operating experience:

   a. In August 2004, a 22 inch pipe downstream of a condensate system flow orifice ruptured killing five workers in the Mihama [Japan] nuclear power plant. The cause of the accident was attributed to pipe wall thinning due to flow accelerated corrosion immediately downstream of a flow orifice. Piping and component configuration play an integral part in the FAC process. The Mihama event was assessed for applicability to Seabrook Station by the FAC Engineer. It was determined that condensate and Feedwater system flow measuring devices at Seabrook Station use venturis and not orifice plates as was the case at Mihama. Piping downstream of five of the six venturis had previously been inspected with no unacceptable wear noted. The sixth downstream location was included in the scope for the subsequent outage and found to have no unacceptable degradation.

b. Subsequent to the discovery of degradation/failure of Feedwater heaters in the industry, Seabrook Station developed a program to assess the condition of Feedwater heaters. Based on the guidance of the CHECKWORKS Users Group recommendations, Seabrook Station developed and implemented an inspection plan to systematically evaluate the conditions of these heaters.

   As of Refueling Outage 13 (Fall of 2009), Seabrook Station has completed inspection of all Feedwater heaters for degradation. In Refueling Outage 08 (Spring of 2002), only one heater, CO-E-25A, was found to have wall thinning. The thinning was in an area below the Extraction Steam inlet nozzle. The area was repaired using weld overlay and is being monitored each refueling outage. No additional wall thinning has been noted since this repair.

3. During Refueling Outage 13 (Fall of 2009), a section of 18 inch Feedwater pipe immediately downstream of a main Feedwater control valve was replaced after being monitored for several cycles and showing signs of continued wear. An engineering change document was developed for the replacement of this pipe section. The use of a FAC-resistant material (chrome-moly) was considered, however, the pipe section was replaced with the original material (carbon steel) due to concerns identified by EPRI that the creation of a chrome-moly to carbon steel interface could create a downstream location that is more susceptible to FAC. The replaced piping will remain in the FAC program and be monitored for wall thinning.

The operating experience of the Seabrook Station Flow-Accelerated Corrosion Program provides objective evidence that the program effectively monitors and trends the aging effects of FAC on piping and components and takes appropriate corrective action prior to the loss of an intended function. Assessments of the Flow-Accelerated Corrosion Program are performed to identify the areas that need improvement to maintain the effective performance of the program.

**Conclusion**

The Seabrook Station Flow-Accelerated Corrosion Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.9 BOLTING INTEGRITY

### Program Description

The Seabrook Station Bolting Integrity Program is an existing program that manages the aging effects of cracking due to stress corrosion cracking, loss of material due to general, crevice, pitting, and galvanic corrosion, microbiologically influenced corrosion, fouling and wear, and loss of preload due to thermal effects, gasket creep, and self loosening associated with bolting. The program manages these aging effects through the performance of periodic inspections. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication and assembly of bolted joints. The program follows the guidelines and recommendations delineated in NUREG-1339, *"Resolution of Generic Safety Issue 29; Bolting Degradation or Failure of Bolting in Nuclear Power Plants"*, EPRI NP-5769, *"Degradation and Failure of Bolting in Nuclear Power Plants"* (with the exceptions noted in NUREG-1339), and EPRI TR-104213, *"Bolted Joint Maintenance and Application Guide"* for comprehensive bolting maintenance. The Seabrook Station Bolting Integrity Program credits other aging management programs for the inspection of bolting. These programs and their scopes are:

a. The Seabrook Station ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program (B.2.1.1) provides the requirements for Inservice Inspection of ASME Class 1, 2, and 3 piping, which includes pressure retaining bolting.

b. The Seabrook Station ASME Section XI, Subsection IWE Program (B.2.1.27) includes steel containment shells and their integral attachments.

c. The Seabrook Station ASME Section XI, Subsection IWF Program (B.2.1.29) provides the requirements for Inservice Inspection of ASME Class 1, 2, and 3 component supports.

d. The Seabrook Station Buried Piping and Tanks Program (B.2.1.22) provide the requirements for the periodic visual inspections of corrosion on buried piping and tanks, including bolting.

e. The Seabrook Station External Surfaces Monitoring Program (B.2.1.24) provides the requirements for the inspection of bolting for steel components such as piping.

f. The Seabrook Station Structures Monitoring Program (B.2.1.31) provides the requirements for the inspection of structural support bolting.

The program includes periodic inspection of closure bolting assemblies to detect signs of leakage that may be indicative of loss of preload, loss of material, or crack initiation. Periodic inspection of bolted closures in conjunction with the Seabrook Station Inservice Inspection Program and Seabrook Station External Surfaces Monitoring Program will detect the aging effects and joint leakage. Operator rounds and system walkdowns will also identify joint leakage.

This program covers bolting within the scope of license renewal, including: 1) safety-related bolting, 2) bolting for nuclear steam supply system component supports, 3) bolting for other pressure retaining components, including non-safety related bolting, and 4) structural bolting. The aging management of reactor head closure studs is addressed by Seabrook Station Reactor Head Closure Studs Program (B.2.1.3) and is not included in this program.

The Seabrook Station Bolting Integrity Program manages the aging effects associated with bolting through material selection and testing, bolting assembly and pre-load control, operation, maintenance, and the performance of periodic inspections. The program also includes repair and replacement controls and requirements on the selection of thread lubricants, consideration of lubricant use on torque determination, and assembly requirements (bolting/torque patterns).

The selection of bolting materials and the use of lubricants at Seabrook Station are based on design specifications, vendor and industry recommendations, and station specifications for torque and bolting material substitution, and follow the guidance of EPRI NP-5769 and NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting. Bolting replacement activities include the application of appropriate gasket alignment, torque, and preload, based on EPRI documents.

ASME Class 1, 2, and 3 pressure boundary closures are inspected for leakage, loss of material, cracking, and loss of preload by Seabrook Station ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program (B.2.1.1). ASME Class 1, 2, and 3 component support bolting is managed by Seabrook Station ASME Section XI, Subsection IWF Program (B.2.1.29). High strength bolts (≥ 150KSI) are used in ASME Section XI component support applications. These bolts use slotted holes and double nuts in lieu of prestressing, thereby avoiding concerns for stress corrosion cracking associated with high pre-stress loads. Additionally, these bolts are not subjected to molybdenum disulfide lubricants – a potential driver of stress corrosion cracking.

ASME Class 1, 2, and 3 pressure boundary closure inspection requirements are in accordance with the requirements of ASME Section XI, Tables IWB

2500-1, IWC 2500-1, and IWD-2500-1, editions endorsed in 10 CFR 50.55a(b)(2) and the recommendations of EPRI-5769.

The ASME Section XI programs use three types of examination: surface, volumetric, and visual. Surface examinations indicate the presence of surface discontinuities and may be conducted by magnetic particle, liquid penetrate, or an eddy current test method. Volumetric examination indicates the presence of discontinuities throughout the volume of material and may be conducted from either the inside or outside of a component. Structural bolting and fasteners are inspected by visual examinations. Components found defective are further inspected to assess the extent of degradation.

Visual examinations cover a number of observation techniques. VT-1 visual examinations detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, or corrosion. VT-2 visual examinations detect evidence of leakage from pressure retaining components. VT-3 visual examinations determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements and by detecting discontinuities and imperfections, such as loss of integrity at bolted connections, loose or missing parts, debris, corrosion, wear, or erosion. These inspection techniques will identify incipient degradation such as crack initiation or loss of material due to corrosion which could result in closure bolting leakage.

Examination schedules meet the requirements of ASME Section XI, Subsections IWB, IWC, and IWD for Class 1, 2, and 3 pressure retaining bolting. Examination schedules also meet the requirements of ASME Section XI, Subsection IWF for Class 1, 2, and 3 component supports bolting.

For bolting covered by other inspection programs, identified leakage is evaluated through the site Corrective Action Program and appropriate actions such as immediate repair, increased monitoring, etc. are implemented based on the significance and impact of the leak. This may include daily checks, such as those performed during operations walk downs, or other actions depending on the significance, trend, and As Low As Reasonably Achievable (ALARA) considerations.

Indications of aging identified in ASME pressure retaining bolting during Inservice Inspection are evaluated per ASME Section XI, Subsections 3600. Indications of aging identified in other pressure retaining bolting, nuclear steam supply system component supports, or structural bolting are evaluated through the Corrective Action Program.

Upon detection of degraded conditions, follow-up inspections, repairs, replacements, or application of additional testing methods are performed as required by the site Corrective Action Program and applicable acceptance criteria of ASME Section XI. Follow-up actions could include torque checks, bolt removal, ASME Section XI type examinations, or use of other diagnostic techniques. For ASME pressure retaining bolting, repairs and replacements are performed in accordance with the Seabrook Station ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program (B.2.1.1).

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M18.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Both the industry and NRC have revealed a number of concerns involving bolting ranging from material control and certification to bolting practices and the impact of aging mechanisms. The Seabrook Station Bolting Integrity Program is effective at identifying, correcting, or improving issues associated with these concerns. Examples of supporting Operating Experience include the following.

1. In August 2000, during a System Engineer's periodic system walkdown, several deficiencies were noted in the Residual Heat Removal Equipment Vault, including corroded pipe support bolting. Engineering evaluated the area and found most of the support bolting in acceptable condition. The actual rusted bolts were not in severe condition and a work order was issued to correct the condition. The support was painted in June 2001 to preventing further degradation.

2. In June 2001, the inspection of bolting on a Train "A" Primary Component Cooling Water valve revealed surface corrosion. Subsequently, the subject bolting was inspected by engineering and it was determined that the structural integrity of the flanged bolting was not an immediate concern. However, it was determined that the bolting needed to be replaced as soon as possible. A design change was issued to allow the use of coated bolts as replacements for the originals. Accordingly, the bolting on Train "A" valve was replaced in September 2001. As part of the extent of condition review, the bolting on Train "B" Primary Component Cooling Water valve was also replaced with coated bolts.

3. In July 2001, during a system engineer's periodic system containment walk-down, the Primary Component Cooling Water system inlet and outlet valves associated with the Containment Air Handling coolers were identified as having corrosion on the valve bonnets and body to bonnet bolting due to condensation. Corrective action included painting the valve bonnets and associated bolting as previous painting of the affected valve bodies had been shown to be effective at preventing further degradation of the carbon steel components caused by condensation.
4. During Refueling Outage 8 (Spring of 2002), bolting replacement on the Primary Component Cooling Water system piping flanges associated with one of the Containment Air Handling Coolers showed signs of galvanic corrosion. The corrosion was attributed to condensation and the combination of carbon steel bolting and copper alloy flanges. Corrective actions included issuing a design change to authorize the substitution of studs and nuts made of stainless steel material in place of the carbon steel studs and nuts originally specified. Subsequently, between July 2003 and April 2004, all of the studs and nuts associated with all six Containment Air Handling Coolers were replaced.
5. In February 2005, during the performance of an ASME Section XI 18 month leakage reduction walkdown, a small boric acid leak was noted on Containment Building Spray heat exchanger channel head bolted connection. An engineering review concluded that there was no degradation of materials that would affect system operability at that time. The subject gasket was scheduled for replacement at the next outage and the subject joint was identified for increased monitoring during system walkdowns. Subsequently, the gasket was replaced during Refueling Outage 11 (Fall of 2006). The same bolted connection was reported to be leaking on February 8, 2007. On March 16, 2007, the torque value was increased to stop the leakage.

These examples demonstrate that the deficiencies associated with bolted joints are effectively identified and corrected.

**Conclusion**

The Seabrook Station Bolting Integrity Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

## B.2.1.10 STEAM GENERATOR TUBE INTEGRITY

### Program Description

The Seabrook Station Steam Generator Tube Integrity Program is an existing program that manages the aging effects of cracking due to intergranular attack, outer diameter stress corrosion cracking, primary water stress corrosion cracking, and stress corrosion cracking; loss of material due to general, crevice and pitting corrosion, erosion, fretting and wear; reduction of heat transfer due to fouling, and wall thinning from flow accelerated corrosion of the Steam Generator components. The Seabrook Station program is applicable to managing the aging of Steam Generator tubes, tube plugs, and tube supports. Seabrook Station has not used tube sleeving repair.

Industry experience has shown that mill annealed alloy 600 Steam Generator tubes have experienced tube degradation due to corrosion, primary water stress corrosion cracking, outside diameter stress corrosion cracking, intergranular attack, pitting, and wastage, along with other mechanically induced degradation, such as denting, wear, impingement damage, and fatigue. The Seabrook Station Steam Generator tubes are Alloy 600 thermally treated tubes. The dominant degradation mode at this time for thermally treated alloy 600 tubes is wear.

The Seabrook Station Steam Generator Tube Integrity Program is based on NEI 97-06 Rev. 2, *"Steam Generator Program Guidelines"*, the response and commitment to Generic Letter 97-06, *"Degradation of Steam Generator Internals"*, and Seabrook Station Technical Specification 3/4.4.5, "*Steam Generators*," which ensure that the performance criteria for structural integrity, accident-induced leakage, and operational leakage are not exceeded. Seabrook Station has implemented the operational leakage limits found in NUREG-1431, *"Standard Technical Specifications for Westinghouse Pressurized Water Reactors"*.

This program identifies and maintains Steam Generator design and licensing basis, and establishes a framework for prevention, inspection, evaluation, repair and leakage monitoring measures to ensure that program requirements are met. Operational leakage limits are included to ensure that, should substantial tube leakage develop, prompt action is taken. These limits are described in Seabrook Station Technical Specifications.

Seabrook Station Technical Specification 6.7.6.k, *"Steam Generator (SG) Program"*, specifies Steam Generator inspection scope, frequency, and acceptance criteria for the plugging and repair of flawed tubes. NRC Regulatory Guide (RG) 1.121, *"Bases for Plugging Degraded Steam Generator Tubes"*, provides guidelines for determining the tube repair criteria and operational leakage limits.

The program includes preventive measures to mitigate degradation related to corrosion phenomena through water chemistry control and the secondary side cleaning and inspection. The Water Chemistry Program, B.2.1.2, mitigates the potentially corrosive effects of the primary and secondary water on the interior and exterior surfaces of the Steam Generator tubes and other Steam Generator internals.

The Steam Generator Tube Integrity Program includes foreign material exclusion guidance, consistent with NEI 97-06. The program includes prevention and detection of foreign objects in the secondary side of the Steam Generators as a means to inhibit wear degradation by performing foreign object search and retrieval at each inspection outage when the hand-hole covers are removed for Steam Generator cleaning.

The objectives of the secondary side inspection plan are to inspect the Steam Generators for foreign objects, perform visual assessments of sludge and visual inspections for secondary side integrity and corrosion. The secondary side inspections have been expanded to examine additional areas of the upper tube bundle and inner tube bundle. The upper internal regions of the tube bundle are inspected for sludge accumulation on the tube support plates. Secondary side inspections are performed every third refueling outage.

The program provides criteria for the qualification of personnel, specific inspection techniques, and associated acquisition and analysis of data, including procedures, probe selection, analysis protocols, and reporting criteria. The performance criteria pertain to structural integrity, accident-induced leakage, and operational leakage.

Nondestructive examination techniques are used to inspect all tubing materials to identify tubes with degradation that may need to be removed from service or repaired. Tubes containing flaws that do not meet the acceptance criteria are plugged. Assessment of tube integrity and plugging or repair criteria of flawed tubes is in accordance with Seabrook Station Technical Specifications.

Degraded plugs and tube supports are evaluated in accordance with the Seabrook Station Corrective Action Program. The program includes requirements for assessment of degradation mechanisms that consider operating experience from similar steam generators and, for each mechanism, defines the inspection techniques as well as the sampling strategy. Compliance with NRC Regulatory Guide 1.121 for plugging or repairing steam generator tubes is achieved through implementation of the NEI 97-06 criteria as incorporated into the program and Seabrook Station Technical Specifications.

Tube inspection scope and frequency, plugging or repair, and leakage monitoring are in accordance with the Seabrook Station Technical Specifications and the Seabrook Station Steam Generator Tube Integrity Program implemented in accordance with NEI 97-06.

Plug inspection scope and frequency, plugging or repair, and leakage monitoring are in accordance with the Seabrook Station Steam Generator Tube Integrity Program implemented in accordance with NEI 97-06.

Tube support plate inspection scope and frequency are in accordance with the Seabrook Station Steam Generator Tube Integrity Program implemented in accordance with NEI 97-06 as well as the program enhancements committed to in Seabrook Station's response to GL 97-06.

Tube integrity is demonstrated by satisfying the structural integrity and leakage performance criteria in conjunction with the performance acceptance standards. Condition monitoring and assessments are performed after inspections to verify that structural and leakage integrity will be maintained for the operating interval between inspections. Comparison of the results of the condition monitoring assessment with the predictions of the previous operational assessment provides feedback for evaluation of the adequacy of the operational assessment and additional insights that can be incorporated into the next operational assessment.

**NUREG-1801 Consistency**

This program, with the exception noted below, is consistent with NUREG-1801-XI.M19.

**Exceptions to NUREG-1801**

1. NUREG-1801 XI.M19 states "*... the licensee's commitment to implement the SG degradation management program described in NEI 97-06, are adequate to manage the effects of aging on the SG tubes, plugs, sleeves, and tube supports.*" The References section for NUREG-1801 XI.M19 identifies NEI 97-06, *"Steam Generator Program Guidelines"* as Revision 1, dated January 2001.

   The Seabrook Station Steam Generator Tube Integrity Program is based on NEI 97-06, *"Steam Generator Program Guidelines"*, Revision 2, dated May 2005.

   Justification for Exception

   Revision 2 of NEI 97-06 did not reduce the functional requirements of Revision 1. In NEI correspondence with the NRC (Alex Marion to Dr. Brian

W Sheron) dated September 9, 2005, *"Steam Generator Program Guidelines, Revision 2"*, NEI states that Revision 2 of NEI 97-06 is consistent with Technical Specification Task Force Traveler TSTF-449 Revision 4, *"Steam Generator Tube Integrity."* The NRC staff review and approval of TSTF-449, Revision 4, was documented in Generic Letter 2006-01, *"Steam Generator Tube Integrity and Associated Technical Specifications"*. Seabrook Station implemented TSTF-449 with License Amendment 115 to Technical Specifications in June of 2007. The approval of TSTF-449 Revision 4 justifies the use of Revision 2 of NEI 97-06.

*Program Elements Affected: Element 1 (Scope of Program).*

**Enhancements**

None

**Operating Experience**

1. Seabrook Station is a four-loop plant with Westinghouse Model F Steam Generators. There are 5626 tubes in each of the four Steam Generators. The design of these Steam Generators includes Alloy 600 thermally treated tubing, full-depth hydraulically expanded tubesheet joints, with urethane (hydrostatic) tack expansions at the tube ends. The tube support plates are broach-holed quatrefoil plates fabricated of Type 405 stainless steel. The U-bends of the first ten rows of tubing were stress relieved after bending. Seabrook Station has completed 13 cycles of plant operations. To date, Seabrook Station has identified the following tube degradation mechanisms:
    a. anti-vibration bar wear due to flow induced vibration in the U-bends of larger radius tubes
    b. minor wear at flow distribution baffles associated with pressure pulse cleaning
    c. possible wear due to foreign objects
    d. outside diameter stress corrosion cracking in a small subset of tubes with elevated residual stress in Steam Generator "D"
    e. top of tubesheet outside diameter stress corrosion cracking in one tube in Steam Generator "C"

2. During Refueling Outage 8 (Spring of 2002), axial indications were identified on several Steam Generator tubes at the quatrefoil tube support plates in Steam Generator "D" with eddy current testing and confirmed with ultrasonic examination. A root cause evaluation was performed and concluded that outside diameter stress corrosion cracking was the degradation mechanism for the Seabrook Station's Steam Generators. The

root cause of the cracking has been determined to be high residual stress due to a manufacturing anomaly in a defined subset of Seabrook Station Steam Generator tubes. A total of 21 tubes were identified in the subset, 15 of which had cracks and were plugged. The remaining 6 tubes were inspected and plugged in Refueling Outage 9 (Fall of 2003). Subsequent inspections have shown that outside diameter stress corrosion cracking was limited to the defined subset of the tube population and no longer exists in the Seabrook Station Steam Generators for tubes with high residual stress.

3. During Refueling Outage 12 (Spring of 2008), foreign objects were discovered in Steam Generator "B" during the inspection of the steam drum area. The root cause evaluation was performed, which concluded that the cause of the foreign objects being in the Steam Generator was inadequate foreign material exclusion controls of material used in Steam Generator "B" steam drum inspection. The root cause evaluation's recommended corrective action to prevent re-occurrence was to revise the job plan for Steam Generator inspection to include a pre-use inspection of all materials brought into the Steam Generators for concealed/loose foreign material. This corrective action has been implemented.

4. The Steam Generator degradation assessment for Refueling Outage 13 (Fall of 2009) identified operating experience at Vogtle Unit 1 where axial and circumferential outside diameter stress corrosion cracking was reported at the top of the hot leg tubesheet. Vogtle has Westinghouse Model F Steam Generators with Alloy 600 thermally treated tubing similar to Seabrook Station. Vogtle Unit 1 was the first U.S. plant to report axial outside diameter stress corrosion cracking at the top of the tube sheet in a Model F Steam Generator. Accordingly, this operating experience was incorporated into the implementation plan for Refueling Outage 13 (Fall of 2009) as part of the Steam Generator inspections.

   Subsequently, during Seabrook station's Refueling Outage 13, top of tube sheet inspections were completed. An axial outside diameter stress corrosion cracking indication was found on one tube in Steam Generator "C" hot leg. The indication was approximately 0.2 inches below the top of the tube sheet and was 0.10 inches long. The tube was plugged on both the hot leg and cold leg sides.

   The Steam Generator degradation assessment for Refueling Outage 13 also discusses the status of anti-vibration bar wear and the minor wear at flow distribution baffles associated with pressure pulse cleaning.

   The flow distribution baffle wear was discovered in Steam Generators "A" and "D". A single wear indication was reported during Refueling Outage 9

(Fall of 2003) at the flow distribution baffle. These indications were retested at Refueling Outage 11 (Fall 2006) to determine if there was any progression of the wear. These indications are attributed to a prior pressure pulse cleaning of the steam generators, based on the location of the indications relative to the pressure pulse locations. Similar indications have been observed in other Model F steam generators at other plants that have applied the pressure pulse cleaning process. The re-examination of these indications at Refueling Outage 11 (Fall of 2006) resulted in no degradation found at the location in Steam Generator "A" and no progression of the wear of the indication in Steam Generator "D".

The anti-vibration bar wear is flow induced vibration at the intersections of the tubes with the anti-vibration bars and is an existing indication in all four Steam Generators. Analysis has determined that for Model F Steam Generators, the number of tubes considered susceptible to anti-vibration bar wear is typically less than 3% of the total number of tubes, with only a fraction of the susceptible tubes expected to require plugging. Tubes that were plugged for anti-vibration bar wear continue to wear after plugging and have been observed to wear through wall after a period of time. Analysis for Seabrook Station concluded that the originally worn plugged tubes would not achieve a condition that could present risk of tube separation before contacting the adjacent tubes. It was further determined that, if an active tube is adjacent to a worn, plugged tube, the progression of contact wear on the active tube would be very slow and sufficient operating time is available under the current Seabrook Station inspection plan (4 Steam Generators per outage, every other outage) that wear would be identified during the planned inspections. Continuing wear on a plugged tube is benign with respect to tube separation since no risk of tube separation was identified for any axially oriented degradation mechanisms.

5. Through Refueling Outage 13 (Fall of 2009), Seabrook Station has plugged a total of 173 tubes in the Steam Generators (A-34, B-25, C-50, and D-64).

The operating experience of the Seabrook Station Steam Generator Tube Integrity Program provides objective evidence that the program effectively identifies degradation prior to loss of intended function. Appropriate guidance for evaluation, repair, or replacement is provided for locations where degradation is found. External operating experience is effectively reviewed and incorporated into the Seabrook Station Steam Generator Tube Integrity Program.

### Conclusion

The Seabrook Station Steam Generator Tube Integrity Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.11 OPEN-CYCLE COOLING WATER SYSTEM

### Program Description

The Seabrook Station Open-Cycle Cooling Water System Program is an existing program that manages the aging effects of:

a. hardening and loss of strength due to elastomer degradation,

b. loss of material due to erosion, due to general, pitting, crevice and galvanic corrosion, due to microbiologically influenced corrosion and fouling and due to liner/coating degradation, and

c. reduction of heat transfer by fouling of specific components.

The program manages aging effects in both safety related and non-safety related components in portions of the Circulating Water, Primary Component Cooling Water, Service Water and Diesel Generator systems. At Seabrook Station, the ultimate heat sink complex consists of the Atlantic Ocean and the draft evaporative cooling tower.

This program relies on the implementation of the recommendations of NRC Generic Letter (GL) 89-13, "*Service Water System Problems Affecting Safety-Related Equipment*" to ensure that the aging effects on the open-cycle cooling water systems will be adequately managed for the period of extended operation. The program, as mandated by GL 89-13, includes (a) surveillance and control of corrosion, erosion, protective coating failure, bio-fouling, silting, and heat transfer degradation, (b) tests to verify heat transfer, (c) routine inspection and maintenance of plant components, (d) system walk downs to ensure compliance with the stations licensing basis and (e) a review of maintenance, operating and training practices and procedures to ensure the effectiveness of established programs.

The Seabrook Station Chlorine Management Program controls micro-biological and macro-biological fouling. Periodic visual inspection of piping and periodic inspection and cleaning of Service Water supply structures ensure adequate monitoring and removal of biofouling agents, corrosion products and silt. Heat transfer capabilities are ensured by periodic performance verification of the heat exchangers within the scope of this program.

Subsequent to the receipt of NRC GL 89-13, Seabrook Station completed the recommended system walk down and review of maintenance, operating and training practices and procedures to ensure the effectiveness of established programs. The Seabrook Station Open-Cycle Cooling Water System Program complies with NRC GL 89-13 recommendations.

The Open-Cycle Cooling Water System piping and components within the scope of this program are constructed of various materials with specific consideration given to their suitability for service in this environment. Materials of construction include carbon steel (lined and unlined), stainless steels, copper and nickel alloys, titanium and elastomers. Where appropriate, piping and components are lined with cement, polyurethane, rubber, or other material to provide protection to the underlying metal surfaces from aggressive cooling water environments. New and replacement materials (such as AL6XN stainless steel) which are highly resistant to corrosion in a salt water environment are included in engineering design change considerations.

The Seabrook Station Open-Cycle Cooling Water System Program includes a variety of inspection and testing methods such as visual, eddy current and ultrasonic test (UT) inspections on plant heat exchangers and piping These activities are designed to detect degradation due to corrosion, microbiologically influenced corrosion, biofouling, silt, debris, and scaling prior to loss of intended function. Visual inspections of elastomers (e.g., rubber expansion joints) are performed to detect erosion and elastomer degradation. Rubber expansion joints are also examined when removed for maintenance activities to provide a more detailed evaluation of condition.

Loss of material due to microbiologically influenced corrosion, fouling, erosion, and corrosion (general, pitting, crevice and galvanic) is managed by Seabrook Station's Chlorine Management Program through aggressive control of the attachment and growth of marine organisms within these systems. Where the piping is lined, the condition of the liner provides sufficient evidence to verify that water has not penetrated the liner material where it could contribute to loss of material by corrosion. If there is indication of possible liner degradation, the portion of the liner is removed and the affected area inspected by UT measurement for wall loss. Following any necessary repairs, the liner is repaired with a suitable material.

Commitments to NRC GL 89-13 established routine inspections and methods of testing for Service Water system piping and components such that corrosion, erosion, silting, and biofouling do not degrade the performance of the Open-Cycle Cooling Water System components.

System Engineers routinely perform system walk downs in accordance with Plant Engineering Guidelines to assess system health and material condition. Service Water flow is routinely monitored by System Engineers as part of normal system performance monitoring to ensure design flow requirements are available for the intended functions.

Performance of Primary Component Cooling Water system and Diesel Generator system safety related heat exchangers is monitored in accordance with Seabrook Station's Service Water Heat Exchanger Program to ensure that heat transfer capabilities comply with the design bases requirements of the systems. These are the only heat exchangers within the scope of License Renewal that are supplied by Service Water. The key elements of this monitoring program are:

a. Temperature Ratio Monitoring of Primary Component Cooling Water heat exchangers is performed at least monthly, and increased to weekly if the condenser pressure deviates by a specified amount from the clean condenser baseline curve. Due to similarities in materials (titanium tubes) and flow media (low temperature sea water), sensitive and readily indicated parameters in the condensers are used as leading indicators of conditions in the Primary Component Cooling Water heat exchangers.

b. Fouling Factor Determination of Diesel Generator Jacket Water Cooling heat exchangers is performed at least once per year. These heat exchangers are normally idle, but are placed in service at least weekly to eliminate stagnant flow conditions.

c. Micro-biological fouling monitoring is performed at least weekly on the Primary Component Cooling Water system and Diesel Generator system heat exchangers while the plant is between 90% and 100% power.

d. The System Engineer documents and trends results and, as necessary, coordinates disposition of unusual or unexpected result with Operations, Chemistry or Engineering personnel. Any unusual or unexpected results are documented and tracked by the Seabrook Station Corrective Action Program.

Selected portions of the Open-Cycle Cooling Water System are subjected to periodic functional and pressure tests as required for safety-related components by ASME Section XI Inservice Inspection.

The Seabrook Station Open-Cycle Cooling Water System Program ensures removal of biofouling agents, corrosion products and silt by periodic visual inspection of piping and by periodic inspection and cleaning of Service Water supply structures. Each Service Water train contains an in-line strainer that is located downstream of both the ocean and the cooling tower pumps and upstream of all other major components in this system. Accumulation of biofouling agents, corrosion products and silt in the in-line Service Water strainers would be indicated by increasing differential pressure across an affected strainer. This differential pressure is monitored each shift by Operations personnel and high differential pressure alarms in the control room.

Open-Cycle Cooling Water System water supply structures, including the sea water inlet and outlet transition structures, the Service Water and Circulating Water pumphouse forebays and the Service Water Cooling Tower basin, are periodically inspected and cleaned to minimize accumulation of biofouling agents, corrosion products, biological material, and silt. The pumphouse forebays are inspected and cleaned each refueling outage and the cooling tower basin is inspected and cleaned, if required, every third refueling outage. Aging effects for these structures will be adequately managed by the Structural Monitoring Program (B.2.1.31).

Inspection and cleaning of the on-shore and off-shore Circulating Water Intake and Discharge Structures is performed periodically to minimize attraction of marine animals and to ensure that there is no impediment to cooling water flow. The on-shore intake and discharge structures are inspected and cleaned at least once every fourth fuel cycle. The off-shore intake and discharge structures are inspected and cleaned at least once every other fuel cycle.

The Seabrook Station Chlorine Management Program provides proceduralized preventive measures to monitor and inject chemicals (chlorine) to mitigate microbiologically influenced corrosion and buildup of macroscopic biological fouling species, such as blue mussels, oysters or clams, and to inhibit scale formation on heat transfer surfaces in the Open-Cycle Cooling Water systems.

Seabrook Station does not have a history of significant biofouling or microbiologically influenced corrosion in the Open-Cycle Cooling Water systems. This is attributable to an aggressive chlorination program, and substantiated by the recent lack of indication of any notable form of bio-fouling documented during the internal surfaces inspections.

In addition to inspection for evidence of fouling, internal piping inspections are used to monitor for potential loss of material. Each refueling outage, a portion of the Service Water system is chosen for inspection of internal surfaces. The primary objective of these inspections is detection of liner degradation, conditions that could lead to liner degradation, and signs of behind-liner corrosion. A video record of each inspection is provided to the System Engineer for review. Indications of liner degradation are evaluated during the outage in which they are identified and any necessary repairs or extent of condition evaluations performed prior to plant startup.

Indication of pipe liner degradation is evaluated for potential loss of material due to general, pitting, crevice and galvanic corrosion in the piping material beneath. Where necessary, the suspect pipe liner is removed and the pipe wall internal surface inspected for corrosion and loss of material. The External Surfaces Monitoring Program (B.2.1.24), is credited for monitoring and trending of above ground Open-Cycle Cooling Water System piping and components for loss of material where that loss of material mechanism may originate from the internal surface (pitting, general corrosion, etc) but be undetected by indications in the liner material. These indications would become evident as through-wall leaks. The Buried Piping and Tank Inspection Program (B.2.1.22), is credited for monitoring and trending of buried Open-Cycle Cooling Water System piping and components for loss of material that may originate from the internal surface but be undetected by indications in the liner material. These programs are also credited for monitoring and trending Open-Cycle Cooling Water System piping and components for external surface aging effects.

The acceptance criteria are specified in the procedures that control the inspections of components. Pipe wall thickness is measured and compared to the minimum wall thickness for fabrication specified in the applicable pipe specification. Any area found to be below this minimum wall thickness is evaluated by engineering for adherence to design minimum wall thickness for the specific application.

Biofouling is chemically controlled or removed as part of the activities performed under this program. Seabrook Station Open-Cycle Cooling Water System heat exchangers are monitored, cleaned, inspected and tested as necessary to ensure the components are maintained and not degraded such that the component and system heat transfer function is maintained and reliable. Acceptance criteria are based on maintaining the system free of significant sediment and biofouling, and surveillances to ensure the Open-Cycle Cooling Water System is able to perform its intended functions.

Unusual or unexpected conditions noted during Open-Cycle Cooling Water System inspections are documented in accordance with the Seabrook Station Corrective Action Program. Such conditions are evaluated by engineering personnel to ensure the continued reliability of the Open-Cycle Cooling Water System.

**NUREG-1801 Consistency**

This program, with the exception noted below, is consistent with NUREG-1801 XI.M20.

**Exceptions to NUREG-1801**

1. NUREG-1801 XI.M20 states *"The system components are constructed of appropriate materials and lined or coated to protect the underlying metal surfaces from being exposed to aggressive cooling water environments"*. The Seabrook Station Open Cycle Cooling Water System includes both unlined and lined piping as part of its design. Since NUREG 1801 program only addresses lined piping and the fact that the Seabrook Station system design includes unlined piping is considered an exception to the NUREG-1801 program.

   <u>Justification for the Exception</u>

   The selection of materials for use in this cooling water environment (salt water) included the installation of unlined piping and components of materials such as Inconel, copper-nickel, aluminum-bronze and stainless steels (including newer stainless steels such as AL6XN) specifically for their resistance to the effects of salt water. These materials are subjected to the same aging mechanisms as if they were lined or coated, and are subject to the same aging management activities described the NUREG-1801 XI.M020 Open-Cycle Cooling Water System aging management program. Therefore, this program will manage unlined piping as well as it will manage lined piping.

   *Program Elements Affected: Element 2 (Preventive Actions).*

**Enhancements**

None

**Operating Experience**

1. During initial installation of Service Water cement lined piping, X-Pando joint compound was used to seal the gaps in the cement lining at the field welded joints. Improper application or long term degradation of this joint compound resulted in localized corrosion of the carbon steel pipe due to sea water intrusion through the gap at the joints. The following improvements have been made to the Service Water piping in order to prevent degradation of the Service Water piping at the field-welded joints.

   a. The cement lined underground piping has been refurbished by installing AMEX-10/WEKO seals at the field welded joints. The AMEX-10/WEKO seal is an elastomer boot seal installed from inside the pipe to prevent sea water from reaching the welded joint. Follow up inspections have been performed to confirm that the seals have been effective in preventing corrosion of the field welded joints. Selected seals will continue to be inspected during refueling outages to ensure that seals are not degraded and continue to be effective in sealing the field welded joints from sea water intrusion.

b. The cement lined above ground piping associated with the Diesel Generator heat exchangers has been replaced with flanged Plastisol PVC lined carbon steel spool pieces. The size and accessibility of this piping did not permit the use of AMEX-10/WEKO seals. Follow up inspections of weld areas by ultrasonic testing and internal visual examinations during refueling outages have confirmed that the engineering design change has been effective in preventing loss of material.

c. Above ground piping located in the turbine building has been refurbished by installing AMEX-10/WEKO seals at field welded joints similar to the methodology utilized for refurbishing the underground piping discussed above.

d. Ultrasonic examinations have been performed on above ground field welded joints where AMEX-10/WEKO seals had not been installed (or could not be installed due to piping configuration) to assess the condition of the piping at the field-welded joints. Joints which revealed wall thinning have been repaired.

2. The original Primary Component Cooling Water system heat exchangers were fabricated with 90-10 CuNi tubing. Both heat exchangers were retubed with 90-10 CuNi tubes due to evidence of tube leakage. As a result of the issues with CuNi tubing, both Primary Component Cooling Water system heat exchangers were subsequently replaced with titanium tubed heat exchangers during Refueling Outage 5 (Spring of 1997). The follow up eddy current inspections in Refueling Outages 10 (Spring of 2005) and 13 (Fall of 2009) have shown that the performance of the titanium tubes has been excellent.

3. The Service Water pump house forebays are inspected and cleaned during each refueling outage. During the cleaning of the Service Water pump house forebay during Refueling Outage 7 (Spring of 2001), an increase in the amount of silt and debris was observed in the area of the Service Water pumps. The normal levels of silt in previous outages were up to 3 feet starting at the traveling screens tapering down to zero around the pump suction bell. Refueling Outage 7 inspections revealed levels of up to 6 feet at the traveling screens tapering down to zero at the pump suction bell. It is approximately 32 feet from the screens to the pump suction. The material was sand, silt, and mussel shells. Evaluation of the condition indicated that the additional debris accumulation was caused by a combination of events as described below.

Seabrook Station experienced an unusually high mussel shell impingement during the summer of 2000. This was attributed to an extended cessation of intake tunnel chlorination from late October 1999 to early May 2000. For a six month period, mussel larvae entrained by the tunnel had an enhanced opportunity to settle and grow on the intake tunnel wall surface, in a chlorine-free environment. Although the Service Water pumphouse forebays were continuously chlorinated during that period, on resumption of intake tunnel chlorination, a large number of mussels were released in the intake tunnel and carried to the forebay. This increase in shells in the forebay led to shell pieces being carried over the traveling screens when screen washing was initiated.

Service Water system strainer valve isolation issues during this same period precluded the ability to isolate the Service Water strainers for cleaning. To minimize the likelihood of carryover of mussel shells to the strainers and possibly creating a high differential pressure condition, screen wash activities were minimized until Refueling Outage 7 at which time the strainer isolation valves could be repaired or replaced. This allowed sand and silt to build up at a higher rate than normal due to the fact we were no longer upsetting the areas around the traveling screens.

Sand and silt is small enough to pass through the system with no interruption of flow. Large amounts of sand and silt may affect the bearing surfaces on the Service Water pump shafts, but the pumps will continue to provide sufficient Service Water flow. Increased wear on the bearings from excessive silt would be first noticed with an increase in vibration levels. Pump vibration levels, flows, and differential pressures are tested quarterly. This condition (silt accumulation in the forebay) did not present a plant operability issue.

As a result of this condition, the Seabrook Station Chemistry Department has re-evaluated the period during which the ocean water intake tunnels should be continuously chlorinated. This expanded chlorination period, in addition to batch chlorination of the Service Water forebay as necessary during the rest of the year has resulted in no further carryover of debris at the traveling screens. Inspections of the Service Water forebay during subsequent inspections have showed no excessive accumulations. The Service Water strainer isolation valves have been restored to service and are available if needed. Traveling screen washing activities were returned to original schedules.

Although the events contributing to this condition are not necessarily aging related, this example does demonstrate that monitoring and control of silt and fouling mechanisms is effective in preventing conditions that could adversely affect the intended function of Service Water components. Such conditions are evaluated and corrected in accordance with GL 89-13 requirements for surveillance and control of biofouling to ensure that silting and biofouling cannot degrade the performance of safety related systems serviced by Open Cycle Cooling Water.

4. As discussed above in example 1, during initial installation of Service Water cement lined piping, X-Pando joint compound was used to seal the gaps in the cement lining at the field welded joints. Improper application or long term degradation of this joint compound results in localized corrosion. An upgrade project was initiated due to the localized corrosion of the carbon steel pipe due to sea water intrusion through the gap at field welded joints. Since the completion of the Service Water Upgrade Project and installation of drop-out spools and AMEX-10/WEKO seals in the mid 1990's, ten through wall leaks have developed. In January 2006, the Service Water System Engineer initiated a Service Water Improvement Plan to evaluate and resolve issues pertaining to these leaks in the Service Water system.

   The Service Water Improvement Plan evaluated current practices for AMEX-10/WEKO seal installation, inspection and testing, field weld ultrasonic examinations, and the history of pipe leaks following the upgrade project. Of particular note was that following the upgrade project, there had been no leaks at Service Water welded joints except for a single occurrence of weepage from a 2 inch Weldolet to pipe weld. The results indicated that the Upgrade Project had adequately resolved the issue of through wall leaks at field welded joints.

   Based on the history of leaks and the areas where those leaks were being found, the System Engineer concluded that the surface ultrasonic examinations performed at welded joints was ineffective because the examination area was not where the leaks were developing and the technique did not work well to detect minor, pin-hole leaks as they were developing. Visual examination of the pipe liner had proved to be a more efficient way to identify areas of pipe wall degradation. Review of video tapes of prior AMEX-10/WEKO seal inspections showed indications on the concrete liner where through wall leaks later developed. Recommendations from this Plan included preparation of a long term plan for visual inspections of above ground pipe through either direct or remote visual methods, enhancement of the current plan for visual inspections of buried pipe to include a specific long range scope and schedule, removal and analysis of a section of installed cement lined Service Water pipe to evaluate the condition of the liner material after more than 25 years of service, and enhancement of the AMEX-10/WEKO seal testing and inspection activity to specifically include monitoring for surface conditions that may indicate beneath-liner corrosion.

Implementation of the Service Water Improvement Plan recommendations was completed by January 2007, with the exception of the laboratory analysis of the cement lined pipe section. The laboratory analysis of the cement lined pipe section was completed in 2008 and concluded that the cement liner material was still in excellent condition and would continue to provide adequate protection of the pipe.

While individual problems, such as, those discussed above, have been identified, the conditions identified did not cause impact on the safe operation of the plant, and adequate corrective actions were taken to prevent recurrence. Appropriate guidance for evaluation, repair, or replacement is provided for locations where degradation is found. Assessments of the Open-Cycle Cooling Water System Program are performed to identify the areas that need improvement to maintain the effective performance of the program. The previous examples of operating experience provide objective evidence that the Seabrook Station Open-Cycle Cooling Water System Program will be effective in ensuring that intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

### Conclusion

The Seabrook Station Open-Cycle Cooling Water Systems Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.12 CLOSED – CYCLE COOLING WATER SYSTEM

### Program Description

The Seabrook Station Closed-Cycle Cooling Water System Program is an existing program that manages the aging effects of (a) cracking due to stress corrosion cracking, (b) loss of material due to general, crevice, pitting and galvanic corrosion, and (c) reduction of heat transfer due to fouling. These aging effects are managed in closed cycle cooling water systems that are not subject to significant sources of contamination, in which water chemistry is controlled and in which heat is not directly rejected to the ultimate heat sink. The program scope includes activities to manage aging in the Primary Component Cooling Water system and Emergency Diesel Generator Jacket Water cooling systems. The program also includes the fire pump diesel engine glycol coolant system, the Control Building Air Handling glycol coolant system (safety-related), and the Thermal Barrier Cooling Water system.

The chemical treatment used for the Primary Component Cooling Water and the Thermal Barrier Cooling Water is hydrazine. The chemical treatment used for the Diesel Generator Jacket Water, the fire pump diesel coolant, and the Control Building Air Handling coolant is glycol.

The Seabrook Station design control program ensures that appropriate materials are used within the system to minimize corrosion and stress corrosion cracking. The systems included in the Seabrook Station Closed-Cycle Cooling Water System Program contain no lined pipe and rely upon appropriate system material selection to minimize corrosion.

The program includes maintenance of system corrosion inhibitor concentrations to minimize degradation, and inspections of opportunity to assess management of component aging.

Seabrook Station implements the industry standards established by EPRI Technical Report 1007820, *"Closed Cooling Water Chemistry Guideline, Revision 1"*, which was issued in April 2004. The program includes monitoring and control of cooling water chemistry to minimize exposure to aggressive environments and use of corrosion inhibitors in the closed cycle cooling water systems to mitigate general, crevice, and pitting corrosion as well as stress corrosion cracking. The program provides the action levels associated with criteria being outside normal operating levels. EPRI 1007820, issued in April 2004, is Revision 1 to EPRI TR-107396 which was issued in October 1997. Since NUREG-1801 refers to EPRI TR-107396 *"Closed Cooling Water Chemistry Guideline"*, Seabrook Station has identified this as an exception.

The Seabrook Station Closed-Cycle Cooling Water System Program identifies the water chemistry parameters to be monitored to ensure that corrosion inhibitor concentrations are maintained within the specified EPRI 1007820 guidelines for the closed-cycle cooling water systems. The program includes guidance to control and monitor the inhibitor concentrations, and the action levels associated with criteria being outside normal operating levels. The program also identifies the normal range, as well as administrative limits (where applicable), sampling frequency, and corrective actions. Sampling frequencies are based on the EPRI 1007820 guidelines.

The system corrosion inhibitor concentrations are maintained within the limits specified in the EPRI 1007820 *"Closed Cooling Water Chemistry Guideline, Revision 1"* except as noted below.

a. Seabrook Station's normal operating range for hydrazine in the Thermal Barrier Cooling Water (a separate closed-cycle cooling loop within the Primary Component Cooling Water system) is specified as 5-300 parts per million (ppm). The EPRI Guidelines consider a normal operating range of 5-50 ppm (5-200 ppm for all-ferrous metallurgy). The higher limit to the operating range was established as a method to minimize radiation exposure required for hydrazine makeup during power operations. Action

Levels 1 and 2 (<5 ppm and <1 ppm, respectively) remain consistent with those specified by the EPRI 1007820 guideline.

b. The Seabrook Station normal range for sulfates in Thermal Barrier Cooling Water is specified as 100-500 parts per billion (ppb) with Action Level 1 and 2 limits of >500 ppb and >1000 ppb, respectively. The EPRI Guidelines consider a normal operating range of ≤150 ppb with Action Level 1 and 2 limits of >150 ppb and >1000 ppb, respectively. The higher operating range and the higher Action Level 1 limit were assigned prior to issuance of EPRI 1007820 *"Closed Cooling Water Chemistry Guideline, Revision 1."* The previous revision did not specify an operating range. The current values were evaluated in a plant Chemistry Study / Technical Information Document. The Action Level 2 limit is consistent with that specified in EPRI 1007820 guideline.

c. The frequency of sampling Thermal Barrier Cooling Water hydrazine and pH remains at a monthly interval based on Seabrook Station operating experience instead of weekly as shown in the EPRI 1007820 guideline for Tier 1 Systems. Data trends support the fact that hydrazine concentration and system pH remain stable between the monthly entries into Containment to obtain samples.

The fire pump diesel engine glycol coolant system, the Control Building Air Handling glycol coolant system, and the Diesel Generator Jacket Water glycol coolant system are periodically monitored. Frequencies of testing and control parameters are consistent with the EPRI guidelines for blended glycol formulations.

Corrosion test coupons are placed in the Primary Component Cooling Water and Thermal Barrier Cooling Water systems to check the effectiveness of the inhibitor and/or the corrosion rates. The criteria for and frequency of coupon inspection is specified in the Seabrook Station Chemistry Procedure *"Corrosion Determination by Coupons"*.

The Seabrook Station Closed-Cycle Cooling Water System Program recognizes that component inspections are an important part of an overall chemistry program to assess corrosion control and chemistry control effectiveness. Seabrook Station Chemistry Procedure, *"Visual Inspection Format for Plant Components"* provides instructions for visual inspection of individual components (e.g., valves, pumps, piping segments, heat exchangers) looking for pitting, general corrosion film presence, biological activity, deposits, etc. when a system or component is open for maintenance. Historical records of these inspections, including photographs, are maintained for comparative purposes.

System walkdowns, System Engineer monitoring and trending, and preparation of periodic system health reports help to ensure the systems' performance meets established design basis requirements.

**NUREG-1801 Consistency**

This program, with the exceptions noted below, is consistent with NUREG-1801 XI.M21.

**Exceptions to NUREG-1801**

1. NUREG-1801 XI.M21 states "*The program relies on the maintenance of system corrosion inhibitor concentrations within the specified limits of Electric Power Research Institute (EPRI) TR-107396 to minimize corrosion and SCC.*"

   EPRI TR-107396, *"Closed Cooling Water Chemistry Guideline"*, was issued in October 1997. Seabrook Station implements the guidance provided in EPRI 1007820, *"Closed Cooling Water Chemistry Guideline, Revision 1"*, issued in 2004.

   Justification for the Exception

   Seabrook Station has reviewed EPRI 1007820 and determined that the most significant difference is that the revision provides more prescriptive guidance. EPRI 1007820 meets the same requirements of EPRI TR-107396 for maintaining conditions to minimize corrosion and microbiological growth in closed cooling water systems for effectively mitigating aging effects. The NUREG-1801 aging management program requirements are unchanged by the transition from EPRI TR-107396 to EPRI 1007820 guideline.

   *Program Elements Affected: Element 2 (Preventive Actions), Element 5 (Monitoring and Trending), and Element 6 (Acceptance Criteria).*

2. EPRI 1007820, *"Closed Cooling Water Chemistry Guideline, Revision 1"*, in Section 5.6 *"Hydrazine-Based Programs"* Table 5-5, *"Operating Ranges and Monitoring Frequencies for Hydrogen-Based Programs"*, provides the following normal operating range for hydrazine:

| Parameter | Normal Operating Range |
|---|---|
| Hydrazine, mixed metallurgy | 5–50 ppm as $N_2H_4$ |
| Hydrazine, all-ferrous | 5–200 ppm as |

| | |
|---|---|
| metallurgy | $N_2H_4$ |

Seabrook Station's Thermal Barrier Cooling Water system is a separate closed-cycle cooling water system that is age managed under the Seabrook Station Closed-Cycle Cooling Water System Program. Seabrook Station has specified the normal operating range for hydrazine in this system at between 5-300 ppm.

<u>Justification for the Exception</u>

The Thermal Barrier Cooling System is located entirely inside the containment. The higher limit to the hydrazine operating range was established to minimize radiation exposure required for hydrazine makeup during power operations. Action Levels 1 and 2 (<5 ppm and <1 ppm, respectively) remain consistent with those specified by the latest revision of the EPRI guidelines. Seabrook Station Chemistry evaluated the significance of operation of the system within this range for hydrazine and determined it to be acceptable based on a review of the potential effects on components and materials and a plan to slowly increase the concentration while monitoring system parameters. In September of 2002, it was decided to raise the upper band for hydrazine in the system from 150 ppm to 300 ppm. The increase was performed in 50 ppm increments every six months until 300 ppm was reached or until there was evidence of increased copper corrosion. The upper limit of 300 ppm was reached without indication of increased copper corrosion.

Corrosion strip test coupons are placed in the Thermal Barrier Cooling System to check the effectiveness of the inhibitor and its effect on corrosion rates.

*Program Elements Affected: Element 2 (Preventive Actions), Element 5 (Monitoring and Trending), and Element 6 (Acceptance Criteria)*

3. EPRI 1007820, *"Closed Cooling Water Chemistry Guideline, Revision 1"*, in Section 5.6 *"Hydrazine-Based Programs"* Table 5-5, "*Operating Ranges and Monitoring Frequencies for Hydrogen-Based Programs*" provides the following normal operating range for sulfates.

| Parameter | Normal Operating Range | Action Levels (AL 1 – 90 days AL 2 – 30 days) |
|---|---|---|
| Chloride, Fluoride, Sulfate | ≤150 ppb as ion (each) | AL 1: >150 ppb<br>AL 2: >1000 ppb |

The Seabrook Station Thermal Barrier Cooling Water System is a separate closed-cycle cooling water system that is age managed under the Closed Cycle Cooling Water Program. Seabrook Station has specified a normal operating range for Sulfates in this system between 100-500 ppb with Action Level 1 and 2 limits of >500 ppb and >1000 ppb, respectively.

Justification for the Exception

The higher allowable sulfate operating range (100 to 500 ppb) and the higher Action Level 1 limit (> 500 ppb) for the Thermal Barrier Cooling Water were assigned prior to issue of EPRI 1007820. The previous revision did not specify an operating range. The Action Level 2 limit is consistent with that specified in the latest revision of the EPRI guidelines. Seabrook Station Chemistry evaluated the significance of continued operation of the system within this range for sulfates. The evaluation was to assess the current corrosion environment and estimate the future effect in the Thermal Barrier System based on the current system chemistry parameters, particularly, sulfate. The conclusion of this evaluation was that the low oxygen levels, alkaline pH and absence of sulfides exhibited in the Seabrook Station Thermal Barrier Cooling Water would mitigate the concern regarding the levels of sulfate above 150 ppb.

*Program Elements Affected: Element 2 (Preventive Actions) and Element 5 (Monitoring and Trending).*

4. EPRI 1007820, *"Closed Cooling Water Chemistry Guideline, Revision 1"*, Section 5.6 *"Hydrazine-Based Programs"* Table 5-5, *"Operating Ranges and Monitoring Frequencies for Hydrogen-Based Programs"* provides the following monitoring frequency for hydrazine and pH for Tier 1 systems.

| Parameter | Monitoring Frequency |
|---|---|
| | Tier 1 Systems |
| Hydrazine, mixed metallurgy | Weekly |
| Hydrazine, all-ferrous metallurgy | Weekly |
| pH, mixed metallurgy | Weekly |
| pH, all-ferrous metallurgy | Weekly |

Seabrook Station has kept the Thermal Barrier Cooling Water system monitoring frequency for hydrazine and pH at monthly instead of weekly as shown in the latest revision of the EPRI guidelines for Tier 1 Systems.

Justification for the Exception

The Thermal Barrier Cooling Water system is located entirely inside the Containment. Keeping the monitoring frequency at monthly was desired to minimize radiation exposure. System data trends show that hydrazine concentration and system pH remain stable between the monthly entries into Containment to obtain samples, which demonstrates that a monthly monitoring frequency is sufficient.

*Program Elements Affected: Element 3 (Parameters Monitored/Inspected) and Element 5 (Monitoring and Trending).*

5. NUREG-1801 XI.M21 states; "*The aging management program monitors the effects of corrosion and stress corrosion cracking by testing and inspection in accordance with guidance in EPRI TR-107396 to evaluate system and component condition. For pumps, the parameters monitored include flow, discharge pressures, and suction pressures. For heat exchangers the parameters include flow, inlet and outlet temperatures, and differential pressure.*"

   The Seabrook Station Closed-Cycle Cooling Water System Program does not rely on performance and functional testing to verify the effectiveness of chemistry controls and management of aging effects.

   Justification for the Exception

   EPRI 1007820 notes that performance testing is typically part of an engineering program that verifies the component active functions. These activities would fall under the Maintenance Rule (10 CFR 50.65). This being the case, performance and functional testing is not included as a part of the Seabrook Station Closed-Cycle Cooling Water System Program. Seabrook Station uses corrosion monitoring and internal inspections of opportunity to monitor program effectiveness at managing component degradation that could impact a passive function. Corrosion monitoring is accomplished through trending of the normal plant periodic sampling, and monitoring of corrosion coupons. The periodic sampling tests for corrosion products which when trended will give an indication of the rate of corrosion ongoing in a system. Seabrook Station also places test coupons in the Primary Component Cooling Water system and Thermal Barrier Cooling Water System to check the effectiveness of the corrosion inhibitor by quantifying the corrosion rates of the coupons. Seabrook Station Chemistry procedure, *"Visual Inspection Format for Plant Components"* provides instructions for visual inspection of individual components (e.g., valves, pumps, piping segments, heat exchangers) looking for pitting, general corrosion film presence, biological activity, deposits, etc. This procedure is used to monitor for corrosion in the component cooling water systems when the system or components are opened for maintenance.

*Program Elements Affected: Element 3 (Parameters Monitored/Inspected), Element 4 (Detection of Aging Effects), Element 5 (Monitoring and Trending), and Element 6 (Acceptance Criteria)*

**Enhancements**

The following enhancement will be made prior to entering the period of extended operation.

1. The Seabrook Station Closed-Cycle Cooling Water System Program will be enhanced to include visual inspection for cracking, loss of material and fouling in the Primary Component Cooling Water system, Thermal Barrier Cooling Water system, Diesel Generator Jacket Water cooling system, Fire Pump Diesel Engine coolant system, and the Control Building Air Handling coolant system when these systems are opened for maintenance.

   *Program Elements Affected: Element 4 (Detection of Aging Effects).*

**Operating Experience**

Seabrook Station has a comprehensive operating experience program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the closed-cycle cooling water systems at Seabrook Station are evaluated. The Seabrook Station Closed-Cycle Cooling Water System Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

1. During Operating Cycle 9 (May 2002 - October 2003), Chemistry Department personnel reported that the hydrazine consumption in the Thermal Barrier Cooling Water system had gone up significantly since Refueling Outage 7 (Fall of 2000). Chemistry personnel reported that only two chemical additions were required during each of Cycles 5, 6 and 7. However, the chemical additions had increased to four during Cycle 8 and to six by middle of Cycle 9. Investigation of the condition identified potential air entrapment in the Thermal Barrier Cooling Water system. As a result, during Refueling Outage 9 (Fall of 2003), the Thermal Barrier Cooling Water system was statically and dynamically vented, which resolved the issue.

2. A 2003 Industry Operating Experience notice was reviewed by Seabrook Station Chemistry Department for applicability. The issue was cracking of brass bolting in the Diesel Generator Jacket Water cooling system. Evaluation showed that Seabrook Station utilizes a glycol mixture which is similar in constituents, but much higher in concentration than that used at the facility issuing the operating experience. For example, tolytriazole in the reporting facility's system is maintained between 5-20 ppm while the minimum specification for tolytriazole at Seabrook Station is 300 ppm. Seabrook Station's limits are based on recommendations from the vendor, a contract laboratory, and EPRI guidelines for closed-cycle cooling water systems. The tolytriazole is specifically added to minimize copper (brass) corrosion. The EPRI guidelines on Closed Cooling Water Chemistry states that tolytriazole reacts with the copper ion to form a thin film that reinforces the oxide film on the copper layer. This film helps to protect the copper from degrading. The tolytriazole levels are checked on a semi-annual basis which has been proven to be sufficient through trending. The evaluation concluded that Seabrook Station's tolytriazole dosage levels are sufficient to minimize brass degradation. The evaluation shows Seabrook Station's use of industry experience to challenge existing program practices, and validate existing program procedures through experiences or standards applied at other nuclear power plants.

3. In November 2009, a Nuclear Oversight audit of the Seabrook Station's Chemistry Control Program was conducted. This audit included a review the chemistry control of the Closed Cooling Water systems at Seabrook Station. The audit results were as follows:

   The audit found that the *"Seabrook Strategic Water Chemistry Optimization Plan – Closed Cooling Water Systems"* is used to establish site-specific Water Chemistry Program that represents the best approach for minimizing corrosion damage and performance losses in the various closed cooling water systems. Seabrook Station *"Miscellaneous Systems / Closed Cooling Water Systems Chemistry Control Program"* provided the details for sampling and analysis of Closed Cooling Water systems.

   The audit concluded that the Seabrook Station Closed Cycle Cooling Water System Program contained all the required attributes found in the EPRI Closed Cooling Water Chemistry Guidelines, Revision 01, Section One *"Instructions"*, and Section 10 *"Methodology for Plant Specific Treatment Optimization."*

   The audit reviewed Chemistry Department logs and Nuclear Oversight Daily Quality Summary Reports associated with closed cycle cooling water system chemistry for compliance with the schedule and analyses specified in the Seabrook Station *"Miscellaneous Systems / Closed Cooling Water*

*Systems Chemistry Control Program"* and EPRI *"Closed Cooling Water Chemistry Guideline, Revision 1."*

During the audit, all sample data for the Closed Cooling Water systems listed below were reviewed for sampling frequency and analytical results:

a. Primary Component Cooling Water
b. Thermal Barrier Cooling Water
c. Secondary Component Cooling Water
d. Diesel Generator Cooling Water
e. Diesel Fire Pump Cooling Water
f. Safety Related Control Building Air Handling Train "A"
g. Safety Related Control Building Air Handling Train "B"
h. Control Building Air Handling (Non-Safety)

The audit found no instances where required samples were not obtained or programmatic requirements were not met for Closed Cooling Water. A review of sample results indicated that samples are obtained and evaluated as detailed in the Seabrook Station *"Miscellaneous Systems / Closed Cooling Water Systems Chemistry Control Program"* and the Chemistry Department uses the Corrective Action Program to document the occurrence of out-of-specification Closed Cooling Water system results.

The audit concluded that the Closed Cooling Water Systems Chemistry Control Program met the requirements of Station Chemistry Manual and the EPRI Closed Cooling Water Chemistry Control Guidelines and that the Chemistry control of the Closed Cooling Water systems was satisfactory.

These operating experience examples provide objective evidence that the Seabrook Station Closed-Cycle Cooling Water Program is effective in monitoring closed-cycle cooling water chemistry parameters, identifying anomalies, initiating corrective actions, and effectively evaluating external operating experience. Assessments of the Closed-Cycle Cooling Water System Program are performed to identify the areas that need improvement to maintain the effective performance of the program.

**Conclusion**

The Seabrook Station Closed-Cycle Cooling Water System Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.13 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

### Program Description

The Seabrook Station Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program is an existing program that manages the aging effects of loss of material due to general corrosion and due to wear of structural components of lifting systems and the effects of loss of material due to wear on the rails in the rail system, for lifting systems within the scope of license renewal. Included in scope are those cranes encompassed by the Seabrook Station commitments to NUREG-0612, "*Control of Heavy Loads at Nuclear Power Plants*," plus two cranes related to fuel handling.

The program manages loss of material due to general corrosion on structural steel members and rails of the cranes within the scope of license renewal. Included are the structural steel members of the bridges, trolleys and monorails. The program also manages loss of material due to wear on rails. Only the structural portions of the in-scope cranes and monorails are in the scope of this program. The individual components of these overhead handling systems that are subject to periodic replacement, or those which perform their intended function through moving parts or a change in configuration, are not in the scope of this program.

The program employs the use of visual inspections to identify aging effects prior to loss of function. Preventive actions are not associated with these activities.

The design of the cranes within the scope of this program did not impose a limit to the number of overcapacity lifts that they would withstand. Additionally, these systems have their loads limited to those within their rated capacity. Because of these procedural controls and the design basis, deterioration of the structural members due to operational fatigue is not expected, and usage of these systems is not recorded.

Structural inspections are conducted under the Seabrook Station lifting systems manual. Periodic inspections are conducted at the frequencies, and include the applicable items, delineated in ANSI B30.2, "*Overhead and Gantry Cranes,*" ANSI B30.11, "*Monorails and Underhung Cranes,*" ANSI B30.16, " *Overhead Hoists (Underhung),*" and ANSI B30.17, "*Overhead and Gantry Cranes (Top Running Bridge, Single Girder, Underhung Hoist)*" for a periodic inspection and in accordance with appropriate manufacturer's recommendations. Inspections are conducted yearly. All periodic inspections are documented on work orders.

Because the program is an inspection program, there are no monitoring and trending activities required.

Degradation of the crane structure due to loss of material through corrosion or wear is evaluated according to vendor recommendations and applicable industry standards as specified in the respective crane inspection procedures. If the crane was designed to a specific Crane Manufacturers Association of America Service Class, the specification that was applicable at the time the crane was manufactured is used.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M23.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancements will be made prior to entering the period of extended operation.

1. The Seabrook Station Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program Lifting System Manual will be enhanced to include monitoring of general corrosion on the crane and trolley structural components and the effects of wear on the rails in the rail system.

   *Program Elements Affected: Element 1 (Scope of Program) and Element 3 (Parameters Monitored/Inspected)*

2. The Seabrook Station Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program Lifting Systems Manual will be enhanced to list additional cranes related to the refueling handling system.

   *Program Elements Affected: Element 1 (Scope of Program) and Element 4 (Detection of Aging Effects)*

**Operating Experience**

There has been no history of corrosion-related degradation that has impaired cranes. Likewise, because cranes have not been operated beyond their design lifetime, there have been no significant fatigue-related structural failures noted by industry experience. Seabrook Station has a comprehensive Operating Experience Program that monitors and assesses industry issues/events for applicability. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events. Preventive Maintenance Work Orders are used for tracking, identifying, and maintaining crane structural components of lifting systems and crane rail systems.

1. A review of a sample of Preventive Maintenance Work Orders, for the cranes within the scope of license renewal reveals no history of wear or structural degradation. Examples include periodic inspections of the Spent Fuel Pool Crane and the Cask Handling Crane, and refueling outage inspection of the Containment Polar Gantry Crane.

2. A review of the Seabrook Station Corrective Action Program by tag number shows that there has been no history of rail or structural wear or corrosion related degradation due to aging related mechanisms that has impaired prevented these cranes from performing their intended function.

3. A condition report generated in 1997 identified a potential for an over load lift by overhead cranes in the Diesel Generator rooms. Evaluation of the condition report determined that no over load lifts had occurred, and provided modifications to the two cranes to preclude any future overload.

The above operating experience review provides evidence that the Seabrook Station Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program is effective in maintaining equipment condition, identifying potential deficiencies, and taking effective corrective actions to resolve issues that may challenge the long-term operability and reliability of the associated systems, structures and components.

**Conclusion**

The Seabrook Station Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.14 COMPRESSED AIR MONITORING

### Program Description

The Seabrook Station Compressed Air Monitoring Program is an existing program that manages the aging effects of (a) hardening and loss of strength due to elastomer degradation, (b) loss of material due to crevice, general, galvanic, and pitting corrosion, and (c) reduction of heat transfer due to fouling of the plant compressed air systems components. The Seabrook Compressed Air Program manages aging in three systems: the plant compressed air system, the containment compressed air system, and the Diesel Generator compressed air subsystem. The program ensures an oil free dry air environment in the compressed air systems. The systems are comprised of components made of stainless steel, carbon steel and copper alloys.

The Seabrook Station Compressed Air Monitoring Program was developed using:

a. Generic Letter (GL) 88-14, *"Instrument Air Supply Problems Affecting Safety related Equipment"*,
b. INPO SOER 88-01, "*Instrument Air System Failures",*
c. ANSI.ISA-7.0.01, *"Quality Standard for Instrument Air - formerly ANSI/ISA S7.0.01-1996",*
d. ISA-S7.3, *"Quality Standard for Instrument Air";*
e. NRC information Notice (IN) 81-38, *"Potentially Significant Equipment Failures Resulting From Contamination of Air-Operated Systems"*,
f. IN 87-28, *"Air System Problems at U.S. Light Water Reactors*",
g. EPRI guidance document NP-7097, *"Instrument Air Systems, A Guide for Power Plant Maintenance Personnel",* issued in 1990,
h. Technical Report TR-108147, *"Compressor and Instrument Air System Maintenance Guide",* and
i. The Standard and Guide for Operation and Maintenance of Nuclear Power Plants, ASME OM-S/G-1998, Part 17, *"Performance Testing of Instrument Air Systems in Light Water Reactor Power Plants"*.

In its response to Generic Letter 88-14, Seabrook Station committed to maintain instrument air quality in accordance with the Quality Standard for Instrument Air, ISA-S7.3, *"Quality Standard for Instrument Air"*. Compliance with ISA-S7.3 is verified by continuous monitoring and periodic testing. In-line dew point monitors are used to verify that the dew point of instrument air at the outlet of the instrument air system dryers is at or below a calculated limit. In-line filters are installed which limit air system maximum entrained particle size. These in-line filters meet or exceed the requirements of the quality standard. Periodic replacement of filters is part of the preventative maintenance program for instrument air systems. Air samples are obtained at least annually and tested to ensure compliance with air quality standards. Additionally, a preventative maintenance program encompassing air system component inspection and repair has been in place since the system was initially placed in service.

The Seabrook Station Compressed Air Monitoring Program includes preventive maintenance activities to check compressed air quality at several locations in the system. The testing includes samples from the compressed air systems on an annual frequency. The program includes leak testing (monitoring) of system components.

Air samples are taken at several locations in the plant compressed air and containment compressed air systems at least annually and tested to ensure compliance with air quality standards. The testing includes samples from the compressed air systems. Dew point is not tested during this annual sampling for the plant compressed air system and the containment compressed air system. Dew point is measured continuously at the discharge of the air dryers in these systems so the sample frequency for dew point is continuous instead of being annual. This is not considered an exception from NUREG 1801.

In-line dew point monitors in the plant compressed air system and the containment compressed air system are used to continuously verify that the dew point of instrument air at the outlet of the air dryers is at or below a calculated limit. The dew point at the outlet of the plant compressed air system dryers is maintained at or below -40 °F. Containment compressed air dew point is maintained at least less than 18°F below containment ambient temperature. The Diesel Generator compressed air sub system dew point is maintained at least 18°F below the lowest expected ambient temperature and is not higher than 39°F.

In-line filters are installed which limit air system maximum entrained particle size. Particles are removed by duplex filters in each instrument air header and filter/regulators supplied with each end user. These in-line filters meet or exceed the 40 micron particle size in the industry standard (ISA-S7.3). All duplex filters are inspected and cleaned or replaced every 5 years.

The instrument air headers are blown down at various low points on an approximately 18 month frequency. As part of this process, the air quality is observed by looking for moisture and loose particulate. The blow down observations are documented to allow for trending of instrument air header quality. The compressed air system air receiver tanks are blown down on a weekly basis.

The plant compressed air system, containment compressed air system and Diesel Generator compressed air sub system air receiver tanks are subject to a New Hampshire State inspection, which is a visual inspection of the (1) vessel internal for structural integrity, (2) the code stamp and (3) relief valve. The inspection removes the tank access covers for internal tank inspection.

The system operating parameters trended by the System Engineer include the instrument air and containment compressed air discharge header pressures, containment header and instrument air header dew points, air compressor operating parameters, and air dryer operating parameters.

The annual testing results are analyzed to verify that the performance of the system is in accordance with its intended function.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M24.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancement will be made prior to entering the period of extended operation.

1. An annual air quality test requirement will be added to ensure compliance with air quality standards for the Diesel Generator compressed air sub system.

   *Program Elements Affected: Element 1 (Scope of Program), Element 2 (Preventive Actions), and Element 3 (Parameters Monitored/Inspected), Element 4 Detection of Aging Effects), and Element 6 (*Acceptance Criteria)

**Operating Experience**

The following examples of operating experience provide objective evidence that the Seabrook Station Compressed Air Monitoring Program is effective in ensuring that intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

1. Investigation of the failure of a containment air compressor in Feb. 2001 identified rust on the compressor cylinder in the area of the piston and rider rings. The rust was caused by condensation that formed in the compressor cylinder from the Primary Component Cooling Water system piping when the compressor was shutdown. In addition to rebuilding the compressor, operations procedures were revised to isolate Primary Component Cooling

Water to the containment air compressors when that compressor is shutdown.

2. In August 2006, a minor secondary plant transient occurred at Seabrook Station due to an air leak in the tubing to a Heater Drain System level control valve. The leak developed due to fretting of the instrument tubing in a stainless steel support clamp. As part of the corrective actions, guidance and operating experience was added to system walkdown guidelines to look for fretting of tubing in supports. A material change from copper to stainless steel for smaller instrument lines was also added to maintenance procedures. Walkdowns of critical air-operated valves in high risk systems were completed for extent of condition. All of the required actions to prevent recurrence have been completed.

3. Seabrook Station reviewed industry operating experience presented in the 2007 INPO Digest *"Recent Experiences Involving Instrument Air Line Failures"* for applicability. The operating experience discussed the failure of an Instrument Air line due to poor workmanship in soldering the copper pipe in the Instrument Air header during initial plant construction. Seabrook Station's piping specification does not allow for soldered joints in Instrument Air piping. Connections must be welded or threaded. The Seabrook Station piping specification requires the 2-inch and 3-inch air headers to be fabricated from carbon steel. Smaller piping downstream of the duplex filters is red brass. Instrument tubing, 1/2-inch or smaller, is either copper or stainless steel. The Instrument air piping and tubing is inspected during system walkdowns. An ultrasonic leak detector has been used to inspect for air leakage. Regularly performed system walkdowns have been performed to identify any system air leakage and corrective actions were initiated to correct any identified leakage.

4. Industry operating experience documented a failure of instrument air un-annealed red brass piping at a nuclear facility in March 2008. The cause of the failure was determined to be stress corrosion cracking due to exposure to ammonia or sulfur during fabrication or construction process.

   The instrument air system piping at Seabrook Station is constructed of either A-106 Grade B carbon steel, or ASTM B-43 red brass, per the piping specification. Typically, carbon steel pipe was used for the larger air headers upstream of the duplex filters, and red brass pipe was used for the 1-inch lines downstream of the duplex filters, up to the point where the line size is reduced to tubing. Per ANSI B31.1, the B-43 red brass material is annealed.

   Failures of the red brass piping (longitudinal cracks) were experienced at Seabrook Station in the early 1990's. The failed piping was analyzed and found to not have been properly annealed. The cause of the failure was attributed to stress corrosion cracking, which may occur when the improperly annealed piping is exposed to ammonia or sulfur-based compounds during fabrication or construction process. All cracks identified were from the outside diameter of the pipe.

As a corrective action, all red brass piping that was in stock (not installed) at the time was disposed of, and proper annealing requirements were specified for all replacement piping. Engineering review concluded that wholesale replacement of all the installed red brass piping was not warranted, as none of the piping failures were catastrophic and air flow to end users was not interrupted. In the case that any piping failures should result in an increased air load exceeding the capacity of the operating air compressor, the Instrument Air system is designed with multiple redundant compressors that will auto-start on a reduction in air pressure.

5. A self assessment of the compressed air monitoring system was completed by Seabrook Station in June 2008 to ensure completeness and proper documentation. The self assessment determined that previously performed response actions for key recommendations were determined to be adequate and that the required maintenance and air quality testing were being performed as specified. This self assessment and corrective action to ensure that commitments are protected served to document that the commitments made in the response to SOER 88-01 are in place and have been effective.

As demonstrated in the operating experience examples above, the Seabrook Station Compressed Air Monitoring Program is effectively managing the aging effects. Conditions identified would not have caused significant impact to the safe operation of the plant, and adequate corrective actions were taken to prevent recurrence. There is sufficient confidence that the implementation of the Compressed Air Monitoring Program will effectively identify degradation prior to failure. Appropriate guidance for evaluation, repair, or replacement is provided for locations where degradation is found. The previous examples of operating experience provide objective evidence that the Seabrook Station Compressed Air Monitoring Program is effective in ensuring that intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

**Conclusion**

The Seabrook Station Compressed Air Monitoring Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.15 FIRE PROTECTION

### Program Description

The Seabrook Station Fire Protection Program is an existing program that manages the effects of aging on fire protection and suppression components through detailed inspections in accordance with the Seabrook Station Surveillance Test Procedures. Specifically, the program manages the following aging effects: (a) cracking due to restraint, shrinkage, creep, and aggressive environment, (b) cracking, loss of bond, loss of material (spalling, scaling) due to corrosion of embedded steel, (c) expansion and cracking due to reaction with aggregates, (d) increased hardness, shrinkage, and loss of strength due to weathering, (e) increase in porosity and permeability, cracking, loss of material (spalling, scaling) due to aggressive chemical attack, and (f) loss of material due to, general, pitting, crevice, galvanic, microbiologically influenced corrosion, abrasion, flaking, fouling, and wear of fire protection and suppression components. Age-related degradation of the diesel-driven fire pump's fuel oil supply line is managed through regularly scheduled fire pump performance tests.

Seabrook Station does not use a $CO_2$ Fire Suppression System. The Halon Fire Suppression System is used in a non safety-related Computer Room in the Control Building and therefore is not in-scope for License Renewal.

The Seabrook Station Fire Protection Program is an existing program that is included in the Seabrook Station Fire Protection Manual. The program provides for managing aging of the penetration seals, fire barrier walls, ceilings and floors and all fire rated doors (automatic or manual) that perform a fire barrier function. The program also manages the aging effects on the intended function of the fuel supply line to the Diesel Fire pumps.

The Fire Protection Evaluation and Comparison to BTP APCSB 9.5-1, Appendix A and Safe Shutdown Capability Report Appendix R quantifies the combustible loading and assesses the fire severity for all plant fire areas.

The Seabrook Station Administrative Procedure for Fire Protection Maintenance and Surveillance Testing incorporates activities that serve to prevent, detect or manage aging of the Fire Protection System. These include regular inspections of fire barriers, penetration seals, and fire rated doors. Performance tests and flushes are performed on fire pumps. These inspections and tests ensure that aging related degradation will be detected in its early stages in order to prevent loss of intended function.

Procedures are established to test and inspect penetration seals, fire barriers, fire doors, and diesel-driven fire pumps for indications of degradation. The Seabrook Station Fire Protection Program activities monitor a variety of parameters to prevent loss of intended function due to age-related degradation prior to loss of intended function. These parameters include:

<u>Penetration Seals</u>

Fire rated assembly penetration seals are inspected every 18 months. Seabrook Station procedural guidelines specify visual inspection of 10% of the Technical Requirement fire rated penetration seals within each category of seals for signs of deficiencies and degradation such as cracking, seal separation from walls and components, separation of layers of material, and rupture and puncture of seals, which are directly caused by increased hardness and shrinkage.

Such Within each seal category, if non-functional penetration seals resulting from deterioration are found, an additional 10% of that category are sampled and inspected.

<u>Fire Barriers walls, floors and ceilings</u>

Fire rated assembly exposed surfaces (barrier walls, floors, and ceilings) are inspected every 18 months. Seabrook Station procedural guidelines specify visual inspection of fire barriers by a fire protection qualified inspector for signs of degradation such as cracking, spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates.

<u>Fire Doors</u>

The Seabrook Station procedure requires surveillance and post maintenance inspection of technical requirements fire rated doors every 6 months and ensures that fire rated doors will be inspected for clearance (gaps) and wear, missing parts on automatic closer mechanisms and latches by a fire protection qualified inspector. The inspections will detect degradation before there is a loss of intended function.

<u>Diesel-Driven Fire Pump</u>

The Seabrook Station procedure requires performance of a flow capacity check on both the "A" and "B" Diesel Fire Pumps every 18 months. The performance of the fire pumps is monitored during testing in order to detect any degradation on the fuel supply line. The Diesel Fire Pumps operation procedure documents test data. If the test data exceeds the acceptance criteria, corrective action will be taken and trended as necessary. The tests will detect degradation before there is a loss of intended function.

Acceptance criteria are defined in the Seabrook Station procedures used to perform tests and inspections of the Fire Protection System. Fire penetration seal inspection results are acceptable if there are no visual indications (outside those allowed by approved penetration seal configurations) of cracking, separation of seals from building structures and components, and no rupture or puncture of seals. Fire barrier inspection results are acceptable if there are no visual indications of cracking, spalling and loss of material caused by freeze-thaw, chemical attack and reaction with aggregates. Fire door inspection results are acceptable if there are no visual indications of wear, holes, damaged or missing parts, and clearances are within limits. Diesel-driven fire pump inspections are acceptable if there is no evidence of loss of material or leaks on the fuel oil supply line. Acceptance criteria for diesel-driven fire pump capacity are contained within the test procedure.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M26.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancements will be made prior to entering the period of extended operation.

1. The Seabrook Station Fire Protection Program implementing documents will be enhanced to perform visual inspection of penetration seals by a fire protection qualified inspector.

   *Program Elements Affected: Element 4 (Detection of Aging Effects)*

2. The Seabrook Station Fire Protection Program implementing documents will be enhanced to include specific age related degradation and inspection qualification as follows:

   a. Enhance existing inspection requirements to list additional age related degradation such as spalling and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates.

   b. Enhanced to perform visual inspection of fire-rated exposed surfaces (Barrier walls, floors and ceilings) by a fire protection qualified inspector.

   *Program Elements Affected: Element 3 (Parameters Monitored/Inspected) and Element 4 (Detection of Aging Effects)*

3. The Seabrook Station Fire Protection Program implementing documents will be enhanced to perform visual inspection of fire-rated doors by a fire protection qualified inspector.

   *Program Elements Affected: Element 4 (Detection of Aging Effects)*

**Operating Experience**

Seabrook Station has a comprehensive Operating Experience Program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events. Preventive Maintenance Work Orders are used for tracking, identifying, and repairing any minor repairs needed to Fire Protection systems, structures and components identified during surveillances. Those issues and events, whether external or plant specific, that are potentially significant to fire protection at Seabrook Station are evaluated.

The Fire Protection Program has been rigorously inspected through various audits such as NRC Triennial Inspections, Nuclear Oversight audits and Quarterly Health Performance Reports listed below are some results of these audits.

1. In September 2002, during surveillance testing of the diesel driven fire pump, a leak was identified on the pump casing vent. The pump was stopped and a work order was issued to repair the leak. This example provides objective evidence that the Fire Protection program satisfactorily identifies degraded conditions in fire suppression systems and that deficient conditions are entered into the Corrective Action Program and corrected.

2. In October 2002, during a surveillance activity, two degraded Appendix R fire barriers were identified. Work orders were issued to repair the two barriers. This example provides objective evidence that during surveillance activities, deficient conditions are identified, entered into the Corrective Action Program, and corrected.

3. In April 2003, a broken door handle was identified on an Appendix R fire door. A work order was issued to repair and retest the door. This example provides objective evidence that the Fire Protection program satisfactory identifies deficient fire door conditions; the deficient condition was entered into the Corrective Action Program and was corrected.

4. In February 2008, an Appendix A fire door was identified as not latching completely and as difficult to open. A work order was issued to repair and retest the door. This example provides objective evidence that the Fire Protection program satisfactory identifies deficient fire door conditions; the deficient condition was entered into the Corrective Action Program and was corrected.

5. The NRC Triennial Fire Inspection Report dated 07/31/2008 identified no findings of significance.

6. A Nuclear Oversight audit report dated 12/7/09 concluded that on an overall basis, the Fire Protection Program was being effectively implemented. Findings were identified related to Combustible Material Controls, Area Pre Fire Strategies, documentation of Action Requests,, and Remote Safe Shutdown Panel equipment tagging. None of these findings were related to managing the effects of aging in the Fire Protection system.

7. Review of the Fire Protection Quarterly System Health Reports for 2009 indicated that the overall system performance was acceptable, with low significance findings related to manpower, design, and fire alarms. None of these findings were related to managing the effects of aging in the Fire Protection system.

In summary, Seabrook Station routinely evaluates NRC and industry communications on fire protection issues for applicability. The Station also initiates and evaluates condition reports during Seabrook Station Fire Protection Program Surveillances and through the Seabrook Station Corrective Action Program

### Conclusion

The Seabrook Station Fire Protection Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.16 FIRE WATER SYSTEM

### Program Description

The Seabrook Station Fire Water System Program is an existing program that manages the effects of aging on Fire Water System components through detailed inspections in accordance with the Seabrook Station Surveillance Test Procedures. Specifically, the program manages the following aging effects: (a) loss of material due to general, crevice, pitting, galvanic, and microbiologically influenced corrosion, (b) fouling, and (c) reduction of heat transfer due to fouling of the Fire Water System components..

Fire Water System components are tested in accordance with the applicable National Fire Protection Association codes and standards. The water supply for the plant Fire Protection system is obtained from two 500,000-gallon heated water storage tanks, of which 300,000 gallons in each tank is reserved for fire protection. Domestic water from the town of Seabrook is used to fill the fire water tanks. A metering pump automatically injects sodium hypochlorite into the fire water tank fill line as required. Water for fire protection is supplied to the system by one motor-driven fire pump and one diesel-driven pump. A second diesel-driven fire pump is provided as a spare. Each pump is capable of taking suction from either tank.

Two motor-driven jockey pumps maintain the fire system pressure, and prevent unnecessary starting of the main fire pumps. System operating pressure is monitored continuously and low pressure is alarmed in the Control Room.

Systems, structures and components included within the scope of the Seabrook Station Fire Water System Program include both fire suppression and fire mitigation components. The program focuses on managing loss of material due to corrosion, microbiologically influenced corrosion, or biofouling of copper alloy, copper alloy > 15% Zn galvanized steel, stainless steel, steel and gray cast-iron components exposed to water, and age-related degradation of components.

The Seabrook Station Fire Water System Program manages aging of the following system components: sprinklers, nozzles, fittings, filters, valves, hydrants, hose stations, flow gages and flow elements, pumps, standpipes, aboveground and underground Piping and Components, water storage tanks and heat exchangers.

The Seabrook Station Fire Protection Manual incorporates many activities that serve to prevent or manage aging of the Fire Water System. These include regular inspections of the Fire Water components, periodic flushing, system performance testing and inspections are conducted to ensure no significant corrosion, microbiologically influenced corrosion or biofouling has occurred in the Fire Water System.

The Seabrook Station Chemistry Manual provides method and directions for adding chemicals to plant systems in order to prevent microbiological growth, inhibit scale formation, dispense solids contained in water, improve chlorination efficiency, and maintain pH level to prevent corrosion of piping and components, including the Fire Water Tanks.

Seabrook Station procedures require the performance of visual inspection of all spray or sprinkler headers for damage and obstruction, visual inspection of dry pipe spray and sprinkler systems for integrity and obstruction, and auto initiation of deluge and preaction sprinkler valves.

Procedures are established to test and inspect fire protection piping and components for indications of degradation. The Fire Water System Program will be enhanced to perform periodic flow testing of the fire water system in accordance with National Fire Protection Association (NFPA) 25 guidelines.

Seabrook Station procedures provide guidance to flush external ring header of the fire suppression water system and to simultaneously conduct the hydraulic performance test. Flushing the header reduces the possibility of corrosion, microbiologically influenced corrosion, and biofouling, which prevents pipe wall thickness reduction.

Seabrook Station procedures require the performance of periodic flow tests to verify required operating pressure and visual inspection for corrosion, deterioration and or damage for all Fire Water Sprinkler System piping and components. Fire Protection System buried pipes are either polyvinylchloride or carbon steel pipe with an internal cement liner and a coal tar epoxy coating on the exterior. Seabrook Station procedures require the performance of a thorough inspection of all internal parts including corrosion and replace any worn or damaged parts.

The Seabrook Station Aboveground Steel Tanks Program, B.2.1.17, includes required inspections of the fire water tanks and fire protection fuel oil tanks.

Seabrook Station procedures require the performance of a visual inspection of fire hose houses, an inspection to ensure required equipment is present at each hose house, a hydrant inspection and operability test, a fire hydrant hose hydrostatic tests, and a hose replacement and gasket inspection and replacement monthly, semi-annually and annually.

Seabrook Station procedures require the performance of cleaning and a tube inspection of the Fire Pump House Heat Exchangers.

Acceptance criteria are defined in the Seabrook Station procedures used for performing tests and inspections of the Fire Water System Program. Sprinkler inspections are acceptable if there is no indication of biofouling in the sprinkler system. Piping inspections and tests are acceptable if there are no indications of unacceptable signs of degradation such as corrosion, microbiologically influenced corrosion or biofouling and that the fire protection system is able to maintain required pressure. Hydrant inspections are acceptable if there is no indication of degradation, such as corrosion.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M27.

**Exceptions to NUREG-1801**

None

## Enhancements

The following enhancements will be made prior to entering the period of extended operation.

1. The Seabrook Station Fire Water System Program will be enhanced to include NFPA 25 criteria for "*where sprinklers have been in place for 50 years, they will be replaced or representative samples from one or more sample areas will be submitted to a recognized testing laboratory for field service testing*". This sampling will be performed every 10 years after the initial field service testing to ensure that signs of degradation, such as corrosion, are detected in a timely manner.

   *Program Elements Affected: Element 4 (Detection of Aging Effects)*

2. The Seabrook Station Fire Water System Program will be enhanced to include the performance of periodic flow testing of the fire water system in accordance with NFPA 25 guidelines.

   *Program Elements Affected: Element 3 (Parameters Monitored/Inspected)*

3 The Seabrook Station Fire Water System Program will be enhanced to include the performance of periodic visual inspection or volumetric inspection, as required, of the internal surface of the fire protection system upon each entry to the system for routine or corrective maintenance to evaluate wall thickness and inner diameter of the fire protection piping. This inspection will be performed no earlier than 10 years before the period of extended operation.

   *Program Elements Affected: Element 4 (Detection of Aging Effects)*

## Operating Experience

Both the industry and NRC have revealed a number of instances of potential problems with sprinklers and hydrants. Seabrook Station has a comprehensive Operating Experience Program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events.

The Fire Water System is audited along with the Fire Protection Program. The program has been rigorously inspected through various audits such as NRC Triennial Inspections, Quality Assurance Department audits and Quarterly System Health Reports. Review of recent operating experience revealed the following:

1. In July 2003, a surveillance activity identified a failure to develop required discharge pressure for the Fire Protection Booster Pump (1-FP-P-374).

Engineering evaluation concluded the pump sizing requirement assumptions were incorrect and re-calculated pump design requirements. An engineering change document was developed to replace the pump impeller to increase total design head. In February 2004 the pump impeller was replaced and the surveillance was re-run with acceptable results. This operating experience provides objective evidence that routine system surveillances discover deficient conditions that are evaluated and corrected per the Corrective Action Program.

2. In July 2004, while performing maintenance on a leaking valve, a maintenance mechanic observed pipe corrosion in the fire pump recirculation header. The mechanic wrote a work order and the pipe was replaced in November of 2005. This example provides objective evidence that through routine system maintenance and inspections, conditions on the Fire Water System piping are monitored, degraded conditions are discovered, and identified deficient conditions are properly repaired and entered into the Corrective Action Program.

3. In September 2006, a section of 4 inch Fire Protection piping that was removed to facilitate installation of a Feedwater spool was found to have significant pitting inside the pipe. Subsequently, engineering performed a visual inspection of the removed piping section (approximately 21 feet) and found the pitted area was approximately 1 inch x 3/8 inch at one end of the removed pipe. It was recommended that the entire piping section be replaced and reinstalled. The pitting appeared to be an isolated case. Based on the service condition and the water being used, no other corrective actions were required. This example provides objective evidence that through routine system maintenance and inspections, conditions on the Fire Water System piping are momitored, degraded conditions are discovered, and identified deficient conditions are properly repaired and entered into the Corrective Action Program.

4. The NRC Triennial Fire Inspection Report dated 07/31/2008 identified no findings of significance.

5. A Nuclear Oversight audit report dated 12/7/09 concluded that on an overall basis, the Fire Protection Program was being effectively implemented. Findings were identified related to Combustible Material Controls, Area Pre Fire Strategies, documentation of Action Requests, and Remote Safe Shutdown Panel equipment tagging. None of these findings were related to managing the effects of aging in the Fire Water system

6. Review of the Fire Protection Quarterly System Health Reports for 2009 indicated that the overall system performance was acceptable, with low significance findings related to manpower, design, and fire alarms. These

findings have no impact on managing the effects of aging in the Fire Water System.

### Conclusion

The Seabrook Station Fire Water System Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.17 ABOVEGROUND STEEL TANKS

### Program Description

The Seabrook Station Aboveground Steel Tanks Program is an existing program that manages the aging effects of loss of material due to general, pitting, and crevice corrosion for aboveground steel tanks within the scope of License Renewal. The Program includes preventive measures to mitigate corrosion and periodic inspections to validate the effectiveness of the preventive actions.

The Seabrook Station Program utilizes the application of protective coatings on the exterior surfaces of the in-scope steel tanks to protect from environmental factors. To ensure that the exterior surfaces of the tanks remain protected, the protective coatings are visually inspected in accordance with the Seabrook Station Structures Monitoring Program.

Inaccessible locations, such as a tank bottom, are surveyed by ultrasonic thickness measurements from inside the tank to detect any exterior material degradation. The ultrasonic thickness measurements of fuel oil tanks within the scope of this program will be performed in accordance with the Seabrook Station Fuel Oil Chemistry Program, Section B.2.1.18. In addition, the structural integrity of the tank foundations and anchor bolts is managed by the Seabrook Station Structures Monitoring Program, Section B.2.1.31.

Caulking and flashing are applied along the tank and ground interface of the Auxiliary Boiler fuel oil storage tank and the two Fire Protection water storage tanks. The tanks are on concrete foundations with a compacted oiled sand foundation. The two diesel fire pump fuel oil tanks are raised on steel supports, clear of their concrete foundations.

Inspections of protective coatings are performed and any degradation of paint, coating, sealant, and caulking is reported through the corrective action system for evaluation. This evaluation assesses and reports the extent of any degradation - cracking, flaking, or peeling of paint, or drying, cracking or missing sealant and caulking.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M29.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancements will be made prior to entering the period of extended operation.

1. Enhance the Seabrook Station Aboveground Steel Tanks Program implementing procedures to:
   a) Include the Fire Protection Fuel Oil Tanks, Auxiliary Boiler Fuel Oil Storage Tank, and Fire Protection Water Storage tanks as part of the scope of tanks.
   b) Add paint flaking and drying, cracking, or missing sealant and caulking as examples of minor structural deficiencies.
   c) Add a requirement that discrepant conditions be reported through the station Corrective Action Program.

   *Program Elements Affected: Element 1 (Scope of Program), Element 3 (Parameters Monitored/Inspected), Element 4 (Detection of Aging Effects), Element 5 (Monitoring and Trending,) and Element 6 (Acceptance Criteria)*

2. Enhance the Seabrook Station Aboveground Steel Tanks Program implementing procedures to require the performance of an ultrasonic examination and evaluation of the internal bottom surface of the two Fire Protection Water Storage Tanks within 10 years prior to the period of extended operation.

   *Program Elements Affected: Element 1 (Scope of Program), Element 3 (Parameters Monitored/Inspected), Element 4 (Detection of Aging Effects), Element 5 (Monitoring and Trending), and Element 6 (Acceptance Criteria)*

**Operating Experience**

Seabrook Station has a comprehensive Operating Experience Program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events. Preventive Maintenance work orders are used for tracking, identifying, and repairing Aboveground Steel Tanks during surveillances. Those issues and conditions, whether external or plant specific, that are potentially significant to the Aboveground Steel Tanks at Seabrook Station are evaluated.

Seabrook Station has encountered minor exterior coating degradation on tanks within the scope of License Renewal, as noted from a screening of the Corrective Action Program and work control data bases. Periodic inspections, followed by any necessary corrective actions, have assured the continued capability of the tanks to perform their intended functions.

The aging management activities that have been applied to aboveground steel tanks at Seabrook Station include:

1. In 1995, three large above ground steel tanks, the Auxiliary Boiler Fuel Oil Storage Tank and Fire Protection Water Storage Tanks “A” and “B” were totally reconditioned to ensure their continued resistance to environmental factors that could lead to degradation and loss of function.

2. In September 1999, degradation of coatings was reported on Fire Protection Fuel Oil Tanks “A” and “B”. The tanks were surface prepped and re-coated to an upgraded condition.

3. In May 2001, in response to a condition of chipped paint and exposed, rusting surface metal around the lower manways of the Fire Protection Water Storage Tank “A”, a work order was initiated. As part of the remedial work, Fire Protection Water Storage Tank “B” was surveyed and found to have a similar condition. Both tanks were surface prepped and re-coated to effect a repair that would maintain the structural integrity of the tank.

4. In June 2001, an inspection of the internal bottom surface of the Auxiliary Boiler Fuel Oil Storage Tank (1-AB-TK-29) was performed by certified personnel under a work order and in accordance with *“Specification for Cleaning, Inspection and Repair of Bulk Fuel Oil Storage Tank 1-AB-TK-29 for North Atlantic Energy Co.”* Inspection results were captured in a report titled, *“Technical Inspection & Engineering Analysis Report”*. The report indicated minimal thickness loss on the nominal ¼” thick floor after 26 years. No degradation of the tank floor was characterized as major.

The above examples demonstrate that the Seabrook Station Aboveground Steel Tanks Program is effective in managing the aging affects during the period of extended operation.

### Conclusion

The Seabrook Station Aboveground Steel Tanks Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.18 FUEL OIL CHEMISTRY

### Program Description

The Seabrook Station Fuel Oil Chemistry Program is an existing program that manages the aging effects of loss of material due to general, pitting, crevice, galvanic, and microbiologically influenced corrosion, and due to fouling in the diesel fuel oil systems for the Emergency Diesel Generators, diesel engine driven Fire Protection system pumps, and the Auxiliary Boiler fuel oil system, through monitoring and maintenance of diesel fuel oil quality. The program complies with the Seabrook Station Technical Specifications and associated Technical Requirements.

This program manages aging effects for the Diesel Generator Fuel Oil Storage Tanks (1-DG-TK-26A & B), the Diesel Generator Fuel Oil Day Tanks (1-DG-TK-78A & B), the Diesel Fire Pump Fuel Oil Day Tanks (1-FP-TK-35A & B) and the Auxiliary Boiler Fuel Oil Storage Tank (1-AB-TK-29) and the associated piping, tubing and valves. By maintaining fuel oil chemistry, removing water and cleaning and inspecting the tanks, this program manages the aging effects to these components.

Seabrook Station monitors new fuel oil deliveries to ensure they meet the requirements of Technical Requirements Program 5.1, "*Diesel Fuel Oil Testing Program*," and Technical Requirement 7, *"Fire Suppression Water System,"* when they are added to the Diesel Generator Fuel Oil Storage Tanks and Diesel Fire Pump Fuel Oil Day Tanks.

New fuel oil samples are taken in accordance with ASTM D4057, *"Standard Practice for Manual Sampling of Petroleum and Petroleum Products"* guidelines. The sample is verified to meet the requirements of applicable ASTM standards prior to offloading to the applicable storage tank.

Seabrook Station uses ASTM Standards D4176, *"Standard Test Method for Free Water and Particulate Contamination in Distillate Fuels (Visual Inspection Procedures)"* and D2709, *"Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge"*, for the determination of water and sediment contamination in diesel fuel as specified by Seabrook Station Technical Requirements, and uses the non-modified ASTM D2276, *"Standard test Method for Particulate Contaminant in Aviation Fuel by Line*

*Sampling"* for measurement of particulates. Fuel oil sampling and analysis is performed in accordance with approved procedures for new fuel and stored fuel.

Checking for the presence of water and removing water, as necessary, eliminates the necessary environment for bacterial survival. Seabrook Station Technical Requirements require that the Diesel Generator Fuel Oil Storage Tanks and the Diesel Generator Fuel Oil Day Tanks are checked for the presence of water every 31 days and water is removed as necessary. Technical Requirement 7 requires that the Diesel Fire Pump Fuel Oil Day Tanks are checked for the presence of water at least once every 92 days and water is removed as necessary. The Auxiliary Boiler Fuel Oil Storage Tank is checked for the presence of water at least once per quarter and water is removed as necessary.

Biological growth and loss of material due to corrosion require the presence of a water interface, therefore the monitoring for and draining of water from the tanks will mitigate the related aging effects. Microbiological organisms are identified as part of the periodic particulate (water/sediment) testing for fuel oil storage tanks. If microbiological organisms are identified as part of the particulate analysis, appropriate actions will be taken as identified during the corrective action evaluation. The Technical Specification surveillance test procedures for fuel oil prescribe that if any values are outside of procedural limits, actions to restore the values to within limits will be immediately initiated.

Stability additives and corrosion inhibitors are not added to fuel oil at Seabrook Station based on the turnover rate, new fuel quality and periodic monitoring of the fuel oil storage tanks. A microbiocide is added to the Diesel Generator Fuel Oil Storage Tanks and monitored quarterly. Biocides are not added to the other tanks as a standard practice. Seabrook Station procedures allow for fuel oil stabilizers to be added to the Diesel Generator fuel oil storage tanks to improve fuel quality during long term storage.

Seabrook Station monitors fuel oil quality and the levels of water in the fuel oil, which cause the loss of material of the tank internal surfaces. The ASTM Standard D4057 is used for guidance in the Seabrook Station fuel oil sampling procedure.

Seabrook Station uses ASTM Standards D4176, *"Standard Test Method for Free Water and Particulate Contamination in Distillate Fuels (Visual Inspection Procedures)"* and ASTM D2709, *"Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge"* for determination of water and sediment contamination in diesel fuel as specified by Seabrook Station Technical Requirements.

Seabrook Station uses the non-modified ASTM D2276 for particulate testing. This method uses a 0.8 μm filter pore size versus the 3 μm size described in the modified ASTM D2276, Method A. The smaller filter pore size allows collection of particulates indicating biological degradation that may be much smaller than 3 μm.

The Seabrook Station Fuel Oil Chemistry Program complies with applicable diesel fuel oil standards as defined by Technical Specifications and Technical Requirements and performs periodic multi-level sampling which provides reasonable assurance that fuel oil contaminants are within unacceptable levels

The Diesel Generator Fuel Oil Storage Tanks, Diesel Generator Fuel Oil Day Tanks, the Diesel Fire Pump Fuel Oil Day Tanks and the Auxiliary Boiler Fuel Oil Storage Tank are drained, cleaned and inspected on a frequency of at least once every ten years. This inspection includes ultrasonic thickness measurements of the tank bottom surface to ensure that degradation has not occurred.

A sample of components in systems that contain fuel oil will also be inspected for evidence of effective management of the aging effects of loss of material in accordance with the Seabrook Station One-Time Inspection Program (B.2.1.20).

The fuel oil analyses results for the fuel oil storage tanks and the new fuel tankers are documented and reviewed against acceptance criteria specified in the Chemistry Department procedures. New fuel deliveries are sampled and verified prior to off-loading the fuel to the storage tanks. The results are also logged in the Chemistry Department data management system to provide long term trending. The frequency of sampling and trending assure timely detection of conditions conducive to corrosion of the internal surface of the diesel fuel oil tanks before the potential loss of intended function.

**NUREG-1801 Consistency**

This program, with exceptions noted below, is consistent with NUREG-1801 XI.M30.

**Exceptions to NUREG-1801**

1. NUREG-1801 XI.M30 states *"The quality of fuel oil is maintained by additions of biocides to minimize biological activity, stabilizers to prevent biological breakdown of the diesel fuel, and corrosion inhibitors to mitigate corrosion."*

Seabrook Station does not use stabilizers or corrosion inhibitors in the diesel fuel oil. Biocide is added only to the Diesel Generator Fuel Oil Storage Tanks.

Justification for the Exception

Monthly testing for and removal of water and the purchase of quality fuel oil negates the need for stabilizers or corrosion inhibitors. Seabrook Station Operating Experience has shown this to be an acceptable alternative based on favorable sample results. New fuel oil is sampled from the delivery tanker per ASTM D4057 guidelines and the sample is verified to meet the requirements of applicable ASTM standards prior to offloading to the applicable storage tank. The program manages the aging effects of the components by maintaining fuel oil chemistry, removing any accumulated water, and cleaning and inspecting the tanks. These fuel oil storage tanks are periodically drained and inspected. The fuel oil is used and topped off often enough to negate the need for stabilizers or corrosion inhibitors.

*Program Elements Affected: Element 2 (Preventive Actions).*

2. NUREG-1801 XI.M30 states *"The ASTM Standards D1796 and D2709 are used for determination of water and sediment contamination in diesel fuel. For determination of particulates, modified ASTM D2276, method A is used. The modification consists of using a filter with a pore size of 3.0 μm, instead of 0.8 μm."*

   The Seabrook Station Fuel Oil Chemistry Program does not use modified ASTM D2276, *"Standard Test Method for Particulate Contaminant in Aviation Fuel by Line Sampling"* method A to sample for particulates.

   Justification for the Exception

   Seabrook Station uses the non-modified ASTM D2276 which uses a filter pore size of 0.8μm verses the 3.0μm as used by the Modified ASTM D2276, method A. The smaller pore size retains smaller particles and is a conservative practice since the analysis for particulates is based on the total weight of particulates captured.

   *Program Elements Affected: Element 3 (Parameters Monitored/Inspected) and Element 6 (Acceptance Criteria).*

3. NUREG-1801 XI.M30 states *"The ASTM Standards D1796 and D2709 are used for determination of water and sediment contamination in diesel fuel."*

Seabrook Station does not use ASTM D1796, *"Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method (Laboratory Procedure)"*, for determination of water and sediment in diesel fuel due to the type of fuel.

<u>Justification for the Exception</u>

Seabrook Station uses the ASTM Standard D4176, *"Standard Test Method for Free Water and Particulate Contamination in Distillate Fuels (Visual Inspection Procedures)"* as well as ASTM D2709, *"Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge"* for determination of water and sediment contamination in diesel fuel as specified by Seabrook Station Technical Requirements. ASTM Standard D2709 is for testing of middle distillate fuels and ASTM Standard D1796 is for fuel oils. Both are standards for laboratory testing for water and sediment. By contrast, Seabrook Station uses ASTM Standard D4176 to perform a Clear and Bright Test of Light Fuel Oil and only ASTM Standard D2709 is used for determination of water and sediment contamination as part of a lab test. The clear and bright test can be performed in the field as well as in the lab and is an easy first screening to determine quality of the fuel oil. Seabrook Station has determined that using one lab test to analyze for water and particulate coupled with the field clear and bright test provides an acceptable approach for detecting water and particulates in the delivered Diesel Generator Fuel Oil.

*Program Elements Affected: Element 3 (Parameters Monitored/Inspected) and Element 6 (Acceptance Criteria)*

**Enhancements**

The following enhancements will be made prior to entering the period of extended operation.

1. The Seabrook Station Fuel Oil Chemistry Program will be revised to include requirements to

   a. Sample and analyze new fuel deliveries including testing for biodiesel prior to offloading to the auxiliary boiler fuel oil storage tank.

   b. Periodically sample stored fuel in the Auxiliary Boiler Fuel Oil Storage Tank.

   *Program Elements Affected: Element 1 (Scope of Program), Element 2 (Preventive Actions), Element 3 (Parameters Monitored/Inspected), and Element 5 (Monitoring and Trending)*

2. The Seabrook Station Fuel Oil Chemistry Program will be revised to include a requirement to check for the presence of water in the auxiliary boiler fuel oil storage tank at least once per quarter and to remove water as necessary.

   *Program Elements Affected: Element 2 (Preventive Actions)*

3. Seabrook Station Fuel Oil Chemistry Program will be revised to require draining, cleaning and inspection of the diesel fire pump fuel oil day tanks on a frequency of at least once every ten years.

   *Program Elements Affected: Element 1 (Scope of Program), Element 2 (Preventive Actions), Element 3 (Parameters Monitored/Inspected), and Element 5 (Monitoring and Trending)*

4. Seabrook Station Fuel Oil Chemistry Program will be revised to include ultrasonic thickness measurement of the tank bottom during the 10-year draining, cleaning and inspection of the Diesel Generator fuel oil storage tanks, Diesel Generator fuel oil day tanks, diesel fire pump fuel oil day tanks and auxiliary boiler fuel oil storage tank.

   *Program Elements Affected: Element 2 (Preventive Actions) and Element 4 (Detection of Aging Effects)*

**Operating Experience**

Seabrook Station has a comprehensive operating experience program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the fuel oil systems at Seabrook Station are evaluated. The Seabrook Station Fuel Oil Chemistry Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

1. In November 2000, a trend of increasing particulates was identified in Diesel Generator fuel oil storage tank, 1-DG-TK-26B. Test results from two samples showed particulate matter at 11.8 mg/L and 11.2 mg/L which exceeded the limit of 10 mg/L. As a corrective action, both Diesel Generator Fuel Oil storage tanks (1-DG-TK-26A and 26B) were filtered to a particulate matter of less than <0.3 mg/L. A condition report was written along with a work order to correct the issue. A cause analysis was performed. The cause analysis determined that a combination of factors were potential contributors to the high particulate count. Corrective actions included cleaning of the diesel generator day tanks at the next opportunity, more frequent replacement of the associated fuel oil filters (every 6

months), review the lube oil - fuel oil interfaces to rule out lube oil contamination of the fuel as a major contributor and to plan for future potential particulate clean-up activities. The plant did not experience a loss of intended function of the Diesel Generator due to the high particulate count.

2. A review of work orders associated with the cleaning and inspection of Diesel Generator fuel oil storage tanks and day tanks and the fire pump fuel oil day tanks indicate no degradation of the tanks. The "A" Diesel Generator fuel oil storage tank was drained and the bottom UT inspected in 1994 and 2003. The "B" Diesel Generator fuel oil storage tank was drained and the bottom ultrasonic inspection examination was performed on the tank bottom in 1994 and 2005. The "A" Diesel Generator fuel oil day tank was drained, cleaned and inspected in 2003. The "B" Diesel Generator fuel oil day tank was drained, cleaned and inspected in 2005.

3. In June 2001, an inspection of the internal bottom surface of the Auxiliary Boiler Fuel Oil Storage Tank (AB-TK-29) was performed by certified personnel under a work order and in accordance with a Seabrook Station specification for cleaning, inspection and repair of the bulk fuel oil storage tank. Inspection results were captured in a technical inspection and engineering analysis report. The report indicated minimal thickness loss on the nominal ¼" thick floor after 26 years. No degradation of the tank floor was characterized as major.

4. A review of Seabrook Station condition reports identified instances when the new fuel oil deliveries were rejected due to the presence of water.

   In December of 2004, a fuel shipment for the Emergency Diesel Generators did not meet the acceptance criteria of the clear and bright test. Samples were analyzed for water, particulate, and haze. Visible water droplets could be seen at the bottom of the clear and bright bottle. A second sample was taken and it also had visible water droplets in the sample bottle and therefore, the tanker fuel oil shipment was rejected.

   In September of 2005, a fuel shipment for the Emergency Diesel Generators did not meet the acceptance criteria for flashpoint. The flashpoint reading of 117°F was below the minimum requirement of 125°F and therefore, the tanker fuel oil shipment was rejected.

   In these instances corrective actions were taken to correct the out of specification condition prior to offloading the fuel oil into the Diesel Generator Fuel Oil Storage Tank.

5. Although not called out in NUREG-1801 XI.M30, the NRC has recently issued Information Notice 2009-02, "*Biodiesel in Fuel Oil Could Adversely*

*Impact Diesel Engine Performance"*. This document indicates that No. 2 diesel fuel could contain up to a 5 percent bio-diesel fuel (B5) blend without labeling the blend in accordance with ASTM D 975-08a, *"Standard Specification for Diesel Fuel Oils"*. Bio-diesel B5 blend: (a) can have a cleansing effect that can increase sediment that could plug filters, (b) could form "*dirty water*" which leads to algae growth, (c) is biodegradable such that long term storage is not recommended and (d) can be more susceptible to gel creation in the presence of brass, bronze and copper fittings, piping and tanks. These effects could lead to plant specific operating experience outside the bounds of industry operating experience.

Existing Seabrook Station plant procedures test for bio-diesel prior to off-load of fuel oil to the Diesel Generator fuel oil storage tanks and fire pump fuel oil day tanks. Acceptance criteria for bio-diesel is <2% (non-detectable).

The operating experiences discussed above show examples of abnormal conditions that were identified by routine monitoring activities and corrective actions that were put in place to correct or prevent reoccurrence or proactive improvements to the program. The previous examples of operating experience provide objective evidence that the Seabrook Station Fuel Oil Chemistry Program will be effective in managing aging effects through the period of extended operation.

**Conclusion**

The Seabrook Station Fuel Oil Chemistry Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.19 REACTOR VESSEL SURVEILLANCE

**Program Description**

The Seabrook Station Reactor Vessel Surveillance Program is an existing program that manages the aging effect of loss of fracture toughness due to neutron embrittlement of the low alloy steel Reactor Vessel.

The extent of Reactor Vessel embrittlement for upper-shelf energy and pressure-temperature limits for 60 years is projected in accordance with the NRC Regulatory Guide (RG) 1.99, Revision 2, *"Radiation Embrittlement of Reactor Vessel Materials"*. Seabrook Station utilizes the NUREG-1801 methodology of projecting neutron embrittlement using surveillance data. Monitoring methods are in accordance with 10 CFR 50, Appendix H, *"Reactor*

*Vessel Material Surveillance Program Requirements"*. Testing methods are in accordance with ASTM E 185-82, *"Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Power Reactors Vessels"*.

The initial surveillance program for Seabrook Station was prepared by Westinghouse in 1983 and issued as WCAP-10110, *"Public Service Company of New Hampshire, Seabrook Station Unit No. 1, Reactor Vessel Radiation Surveillance Program"*.

Aging effects are detected through testing of specimens from surveillance capsules that are periodically withdrawn from the vessel. Testing is performed to determine the increase in transition temperature, Reference Temperature – Nil Ductility Transition ($RT_{NDT}$), and the decrease in upper shelf energy for materials that closely match Reactor Vessel beltline materials. Trending is accomplished utilizing RG 1.99 methods for projection of $RT_{NDT}$ and upper shelf energy. Projection of the increase in $RT_{NDT}$ and the decrease in upper shelf energy provides early indication of whether the fracture toughness properties of the Reactor Vessel beltline materials will fail to meet regulatory requirements.

Neutron embrittlement is evaluated through surveillance capsule testing and evaluation, fluence calculations and monitoring of effective full power years. All capsules pulled to date have met the testing and reporting requirements of ASTM E185-82.

Seabrook Station falls under the NUREG-1801 XI.M31 item 6 condition of having a surveillance program that consists of capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years. The Seabrook Station Reactor Vessel Surveillance Program will withdraw one remaining capsule at an outage in which the capsule receives a neutron fluence that meets the schedule requirements of 10 CFR 50 Appendix H and ASTM E185-82 and that bounds the 60-year fluence and test the capsule in accordance with the requirements of ASTM E185-82. Any capsules remaining in the Reactor Vessel will be removed at that time unless determined that the capsule(s) might provide meaningful metallurgical data if left in place.

Although the regulatory requirements applicable to Reactor Vessel surveillance are not changed as a result of license renewal, several actions are required to show that regulatory requirements will continue to be met through the period of extended operation. For example, the neutron fluence projection used to determine compliance with the Pressurized Thermal Shock screening criteria of 10 CFR 50.61, *"Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"*, must account for the additional effective full power years to be accrued during the period of extended operation.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M31.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancements will be made prior to entering the period of extended operation.

1. The Seabrook Station Reactor Vessel Surveillance Program will be enhanced to specify that all pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage.

   *Program Elements Affected: Element 1 (Scope of Program)*

2. The Seabrook Station Reactor Vessel Surveillance Program will be enhanced to specify that if plant operations exceed the limitations or bounds defined by the Reactor Vessel Surveillance Program, such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes on the extent of Reactor Vessel embrittlement will be evaluated and the NRC will be notified.

   *Program Elements Affected: Element 1 (Scope of Program)*

3. The Seabrook Station Reactor Vessel Surveillance Program will be enhanced as necessary to ensure the appropriate withdrawal schedule for capsules remaining in the vessel such that one capsule will be withdrawn at an outage in which the capsule receives a neutron fluence that meets the schedule requirements of 10 CFR 50 Appendix H and ASTM E185-82 and that bounds the 60-year fluence, and the remaining capsule(s) will be removed from the vessel unless determined to provide meaningful metallurgical data.

   *Program Elements Affected: Element 5 (Monitoring and Trending)*

4. The Seabrook Station Reactor Vessel Surveillance Program will be enhanced to ensure that any capsule removed, without the intent to test it, is stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation.

   *Program Elements Affected: Element 5 (Monitoring and Trending)*

**Operating Experience**

1. The first surveillance capsule, "U", was removed from the Seabrook Station Reactor Vessel in August 1991 after 0.913 effective full power years of reactor operation. The second surveillance capsule, "Y", was removed in May 1997 after 5.572 effective full power years of reactor operation. The third surveillance capsule, "V", was removed in April 2005 after 12.39 effective full power years of reactor operation. The results of the capsule V analysis were submitted to the NRC as required by 10 CFR 50 Appendix H by WCAP-16526-NP *"Analysis of Capsule V from FPL Energy-Seabrook Unit 1 Reactor Vessel Radiation Surveillance Program".* This report summarizes results from all three capsules ("U", "Y", and "V") and the initial un-irradiated mechanical tests for comparison. Seabrook Station evaluation of these results is documented by an engineering evaluation and found that the Pressure/Temperature limit curves and Low Temperature Overpressure Protection set points remained valid for the 20 effective full power year period of applicability. All surveillance materials exhibited adequate upper shelf energy.

2. The Reactor Vessel lower shell plate material and beltline weld were included in the surveillance capsules as the limiting beltline materials in all three capsules. The radiation induced transition temperature shifts (delta $RT_{NDT}$) for the limiting plate and weld materials, from all three capsules, were within the standard two deviations of Regulatory Guide 1.99, Revision 2 predictions. The irradiated upper shelf energy values for the vessel weld metal and base materials samples were well in excess of the 50 ft-lb. lower limit for continued safe operation and are expected to be maintained above 50 ft-lbs. throughout vessel life as required by 10 CFR 50 Appendix G, *"Fracture Toughness Requirements".*

3. Following removal of capsule "V" during Refueling Outage 10 (April of 2005), the surveillance capsule removal schedule was revised by UFSAR Change 06-019, "*Revise Reactor Vessel Surveillance Program Withdrawal Schedule*," incorporating the actual capsule "V" fluence and effective full power years of exposure, and recommending that the extra capsules ("W" and "Z") be withdrawn within one cycle of the removal of capsule "X". This would allow for meaningful metallurgical data for approximating 60 years of plant operation. This change also recommended that these capsules be placed in storage upon removal.

4. In 2004, another nuclear facility found damage to the lower internals support flange and the surveillance capsule access plug in the lower internals flange. The surveillance capsule access plug appeared to have been partially tilted in its access hole when the upper internals were installed at a prior outage. The weight of the upper internals and Reactor

Vessel head, combined with the bolted closure force to crush the access plug to the height of the lower internals hold down spring. A review of this operating experience identified a similarity between the Seabrook Station Reactor Vessel surveillance capsule access plug locations and those at the other facility. The applicable Seabrook Station maintenance procedure was revised to address verifying with an underwater camera, that the surveillance capsule access plugs are in their proper location and seated flat against the flange prior to installing the upper internals as part of fuel loading. Although this event was not aging related, this example demonstrates the rigor that Seabrook Station maintains in responding to industry operating experience that could impact the Seabrook Station Reactor Vessel Surveillance Program.

These examples of Seabrook Station operating experience provide evidence that the current Reactor Vessel Surveillance Program is adequately monitoring the aging effect of loss of fracture toughness due to neutron embrittlement of the low alloy steel Reactor Vessel, and that Seabrook Station is maintaining an awareness and sensitivity to operating experiences throughout the industry that could impact this program.

**Conclusion**

The Seabrook Station Reactor Vessel Surveillance Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.

### B.2.1.20 ONE-TIME INSPECTION

**Program Description**

The Seabrook Station One-Time Inspection Program is a new program. This program addresses potentially long incubation periods and provides a means of verifying that aging effects are either not occurring or are progressing so slowly as to have negligible effect on the intended function of the structure or components through the period of extended operation. The Seabrook Station One-Time Inspection Program provides measures for verifying that an aging management program is not needed, for verifying the effectiveness of an existing program, or for determining that degradation is occurring which will require evaluation and corrective action.

The program elements include (a) determination of appropriate inspection sample size, (b) identification of inspection locations, (c) selection of examination techniques, including acceptance criteria, and (d) evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample includes locations where the most severe aging effect(s) would be expected to occur. Inspection methods may include visual (or remote visual), surface or volumetric examinations, or other established nondestructive examination techniques.

This Program will be used to:

- Verify the effectiveness of the Seabrook Station Water Chemistry Program (B.2.1.2) for managing the aging effects in portions of piping and components exposed to a treated water environment.
- Verification of the effectiveness of the Seabrook Station Fuel Oil Chemistry Program (B.2.1.18) for managing the aging effects of piping and components in systems that contain fuel oil.
- Verification of the effectiveness of the Seabrook Station Lubricating Oil Analysis Program (B.2.1.26) for managing the aging effects of piping and components in systems that contain lube oil.

This program will perform a one-time inspection of selected components determined to be most susceptible to the potential degradation mechanisms. The components to be inspected will be chosen from the systems within the scope of the Seabrook Station Water Chemistry Program, Seabrook Station Fuel Oil Chemistry Program, and the Seabrook Station Lubricating Oil Analysis Program. From these groups of components, a sample of the population will be selected for inspection as part of the Seabrook Station One-Time Inspection Program. The inspection population will be based on such aspects of the systems and components as similarity of materials of construction, operating environment, and aging effects. The sample size will be based on such aspects of the systems and components as the specific aging effect, location, system, and structure design, materials of construction, service environment, or previous failure history. The selection criteria will include stagnant or low-flow areas.

This program assesses aging effects of loss of material due to corrosion (general, pitting, crevice, or galvanic); loss of material due to microbiological influenced corrosion; loss of material due to fouling; reduction of heat transfer due to fouling; and cracking due to stress corrosion cracking and cyclic loading of susceptible components within License Renewal scope. This program will select the locations to be inspected, provide the inspection criteria, evaluate the results of the inspections and provide recommendations for additional inspections, as necessary. The results of these inspections will be evaluated for impact throughout the relevant systems at Seabrook Station. They will also determine the need for additional inspections to manage this aging effect.

The inspections will be scheduled as close to the end of the current operating license period as practical, with margin provided to ensure completion prior to commencing the period of extended operation. The inspection requirements may be satisfied by a review of repair or other inspection records to confirm that the component has been inspected for aging degradation and no significant degradation has occurred within ten years prior to the period of extended operation.

The Seabrook Station One-Time Inspection Program is intended to verify that aging degradation is either not occurring or is occurring at such a slow rate that the component or structure's intended function is not affected. By definition, the inspections are one-time, and therefore they do not include any methods to prevent or mitigate degradation.

The examination techniques will be visual, surface, volumetric, or other appropriately established non-destructive examination (NDE) methods. The NDE will be performed by qualified personnel following procedures consistent with ASME Code and 10 CFR 50, Appendix B, *"Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants"*.

Any indication or relevant conditions of degradation detected are evaluated. Acceptance criteria may be based on applicable ASME or other appropriate standards, design basis information, or vendor-specified requirements and recommendations.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.M32.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

There is no programmatic operating experience specifically applicable to the new one-time inspections. However, plant and industry operating experience will be considered in the selection of the initial component sample sets. The following examples demonstrate that the existing condition monitoring and condition reporting programs are effective in identifying, evaluating and correcting aging effects typical of the scope of this program.

1. In July of 2001, during replacement of Train "B" Service Water Cooling Tower pump with a new style pump, corrosion was observed on the pump discharge head flange. The discharge head was replaced with a more corrosion resistant material during the installation of the new style pump.

2. In February of 2002, two Service Air system receiver inlet isolation valves were scheduled for replacement due to seat leakage. When these valves were removed, corrosion was discovered inside the piping adjacent to the valves. Engineering was contacted to perform a minimum wall evaluation, which concluded that the measured wall thickness satisfied the piping system requirements for design pressure as well as mechanical loading.

   During internal inspection, a large amount of rust and scale was also found lying in the piping between the valves and the air receivers. All loose scale was removed from the tank and inlet piping. The inlet piping to this receiver was inspected by the system engineer and design engineering. All corrosion products appeared to be surface corrosion. No evidence of pitting or wall thinning was found.

   Additionally, when the receiver inlet isolation valves were removed, the adjacent check valves were found to be corroded. Corrective work orders were generated to replace these check valves. Regular preventive maintenance work orders were also created for the check valves to ensure their proper operation in the future.

   Several condition reports had previously documented rust in the Service Air system piping and seat leakage on valves due to rust in the system. Rust is expected in this system because the air is not dried and the piping is carbon steel. However, there was no piping integrity issue and therefore, complete piping replacement was not warranted at that time. The valves with seat leakage were gate valves used to isolate the air receivers and to split the Service Air headers. The present condition of the valves had not prevented required work from being completed. Receiver inspections could still be completed using these valves as isolation. Additionally, seat leakage on these valves was determined to have no adverse effect on the system since they are normally full open. It was concluded that if the seat leakage on these valves worsened then they could be replaced with stainless steel ball valves in the future.

3. In October of 2002, during an inspection of the internal surface of the Steam Generator Blowdown system acid tank corrosion and pitting of the tank wall base metal was noted. Tank thickness measurements were performed by ultrasonic examination and by using a pit gage. As found readings were compared to tank minimum wall thickness, which determined that base metal repair of the tank wall was not necessary. The inside surface of the

tank was coated with corrosion resistant material eliminating any further degradation in the tank.

4. During Refueling Outage 12 (Spring of 2008), internal inspections of the Train "B" Service Water strainer bypass line was performed, which identified several areas of rust staining and rust build-up. Ultrasonic thickness measurements were taken of the identified areas of concern. Subsequently, an engineering evaluation was performed, which concluded that the piping was acceptable for continued operation. The piping is currently scheduled to be repaired and/or replaced during Refueling Outage 14 (Spring of 2011).

The operating experience review revealed the aging effects falling under the Seabrook Station One-Time Inspection Program are not contributing to any adverse trend in performance or loss of component intended functions. The previous examples of operating experience provide objective evidence that the Seabrook Station One-Time Inspection Program will be effective in assuring that intended function(s) will be maintained consistent with the current licensing bases for the period of extended operation.

**Conclusion**

The Seabrook Station One-Time Inspection Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.21 SELECTIVE LEACHING OF MATERIALS

**Program Description**

The Seabrook Station Selective Leaching of Materials Program is a new program that manages the aging effect of loss of material due to selective leaching in components made of gray cast iron and copper alloys with greater than 15 percent zinc that are exposed to raw water, brackish water, treated water (including closed cycle cooling), or groundwater environment. Seabrook Station has also identified copper alloys with greater than 8 percent aluminum (e.g., aluminum bronze) as susceptible to selective leaching. Because NUREG-1801 does not include this material type, Seabrook Station has included it with the copper alloys with greater than 15 percent zinc components.

The Seabrook Station Selective Leaching of Materials Program will include a one-time inspection of selected components that may be susceptible to selective leaching. Because selective leaching is a slow acting corrosion process, the one-time inspection for selective leaching will be performed within the last five years prior to entering the period of extended operation.

Where practical, the inspection will include a representative sample of the population and focus on the bounding of lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. Twenty percent of the population with a maximum sample of 25 constitutes a representative sample size. Otherwise, a technical justification of the methodology and sample size used for selecting components for a one-time inspection will be included in the Seabrook Station Selective Leaching of Materials Program. Each group of components with different material/environment combinations will be considered a separate population.

Visual inspection and mechanical examination techniques (Brinell hardness testing or other mechanical examination techniques such as destructive testing, when appropriate, scraping, chipping or other types of hardness testing), or additional examination methods that become available to the nuclear industry, will be used to determine if selective leaching is occurring on the surfaces of a selected set of components. NUREG-1801 XI.M33, "*Selective Leaching of Materials*," recommends that visual inspections be performed with Brinell hardness testing.

Visual inspection is capable of detecting corrosion while the mechanical test techniques such as chipping, scraping or hardness testing are capable of detecting a corroded or weakened component structure.

If it is determined that selective leaching is occurring, then an engineering evaluation will be initiated to determine acceptability of the affected components for continued service. Follow-up of unacceptable inspection findings will include an expansion of the inspection sample size and location.

**NUREG-1801 Consistency**

This program, with the exception noted below, is consistent with NUREG-1801 XI.M33.

**Exceptions to NUREG-1801**

1. NUREG-1801 XI.M33 states "*One acceptable procedure is to visually inspect the susceptible components closely and conduct Brinell Hardness testing on the inside surfaces of the selected set of components to determine if selective leaching has occurred.*"

   Seabrook Station will utilize visual inspections and mechanical examination techniques [(Brinell hardness testing or other mechanical examination techniques such as destructive testing (where appropriate), scraping, chipping or other types of hardness testing)], or additional examination methods that become available to the nuclear industry, to determine if selective leaching is occurring on the surfaces of a selected set of components.

Justification for the Exception

The form and configuration of many components do not physically allow access for Brinell hardness testing. This is particularly an issue with testing from the inside surface. In such cases, additional mechanical test techniques are needed.

The visual inspection is capable of detecting corrosion while the mechanical test techniques such as chipping, scraping or hardness testing are capable of detecting a corroded or weakened component structure.

*Program Elements Affected: Element 4 (Detection of Aging Effects).*

**Enhancements**

None

**Operating Experience**

Seabrook Station has experienced instances of de-aluminization of aluminum-bronze components having an internal environment of raw sea water. Seabrook Station has already recognized this aging mechanism and is pro-active in addressing the condition as it is discovered.

1. A 1998 condition report noted that throughout the sea water piping systems at Seabrook Station there are aluminum bronze pipe fittings which show signs of leakage. The condition report notes that some aluminum-bronze fittings have been replaced with copper-nickel fittings as permitted by an existing engineering change document. The condition report also identified related programmatic issues associated with aluminum-bronze pipe fittings.

   Actions taken as a result of this condition report include:

   a. Piping specifications for pipe classes applicable to small (less than 4 inch) non-ferrous piping and fittings in seawater service, non-safety related and safety-related, were revised to specify the use of copper nickel fittings, flanges and unions for the piping systems within these two pipe classes instead of the previously specified aluminum bronze.

b. The procurement department identified all stock codes for aluminum-bronze fittings, flanges and unions specified by the piping specifications noted above and disposed of in stock items, cancelled outstanding purchase orders and set affected stock codes to the appropriate status code.

c. Planning reviewed all outstanding Service Water work requests that specified the use of aluminum-bronze and changed the stock codes to reflect the new material fittings.

d. Piping designers were directed to specify copper nickel in lieu of aluminum-bronze for the affected piping systems. Engineering also reviewed and revised outstanding piping design change documents which specified the use of aluminum-bronze, fittings, flanges and unions.

2. A December 2002 condition report documented weepage through the valve body of Service Water system valve, 1-SW-V-48. The valve was a 3 inch aluminum-bronze plug valve. The condition report noted that sand cast aluminum-bronze valves are subject to de-aluminization, and that this is a known mechanism. The de-aluminized area was localized on the valve body downstream of the plug seating assembly. An engineering evaluation determined that, although weeping, the valve maintained its structural integrity. The valve was replaced during the subsequent Refueling Outage OR9 (Fall of 2003).

3. A 2007 condition report noted that three aluminum-bronze valves were scheduled for replacement and several others identified as no longer available from the original vendor. Seabrook Station Procurement identified a replacement bar stock ball valve that would eliminate the de-aluminization issue with the previous cast aluminum-bronze valves in sea water service.

These examples demonstrate that Seabrook has identified and addressed one form of selective leaching and has taken corrective actions to monitor and refurbish susceptible materials. Appropriate guidance for evaluation, repair, or replacement is provided for locations where de-aluminization was found. The previous examples of operating experience also provide objective evidence that the Seabrook Station Selective Leaching of Materials program will be effective in identifying other forms of selective leaching and ensuring that the intended function of components susceptible to such aging effects will be maintained.

**Conclusion**

The Seabrook Station Selective Leaching of Materials Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.22 BURIED PIPING AND TANKS INSPECTION

### Program Description

The Seabrook Station Buried Piping and Tanks Inspection Program is a new program that will manage the aging effects of loss of material due to general pitting, crevice, and microbiologically influenced corrosion from the external surfaces of buried steel (including cast iron) and stainless steel components. Although the program refers to buried tanks as well as piping, Seabrook Station has no buried tanks in scope for license renewal. The systems that contain buried piping that credit this program are the Auxiliary Boiler, Control Building Air Handling, Condensate, Plant Floor Drain, Diesel Generator, Fire Protection and Service Water systems.

The Seabrook Station program will include preventive measures to mitigate corrosion and periodic inspections that manage the aging effects of corrosion on the pressure-retaining capacity of buried piping and bolting in the scope for license renewal.

The initial installation of in-scope buried steel and stainless steel piping at Seabrook Station included external coatings and wrappings. The Seabrook Station program will include provisions for visual inspections of the protective wraps and coatings on buried steel and stainless steel piping in the scope for license renewal. The visual inspections for damage will be performed when the pipes are excavated during maintenance or other activities and when a pipe is dug up and inspected. The inspections will look for evidence of damaged wrapping or coating defects, such as coating perforation, holidays, or other damage. If damage to the protective wraps or coatings is found, and the piping surface is exposed, the outer surface of the pipe will be inspected for loss of material.

Inspections intended to identify aging effects of corrosion on the pipe may also be performed by methods other than visual where such technology permits. An example of this type of emerging technology is ultrasonic inspection of pipe wall from the internal surface of the pipe designed to detect loss of material at the pipe external surface.

At least one opportunistic or focused inspection will be performed within the 10 year period prior to entering the period of extended operation. Upon entering the period of extended operation a planned inspection will be performed within ten years, unless an opportunistic inspection has occurred within that ten year period.

Opportunistic and/or focused visual inspections will be performed in areas with the highest likelihood of corrosion problems or areas with a history of corrosion problems.

The results of previous inspections will be evaluated, and used to assess the condition of the external surfaces of other buried steel and stainless steel components, and to identify susceptible locations that may warrant further inspections.

Any coating and wrapping degradations will be documented, repaired and evaluated under the Corrective Action Program.

**NUREG-1801 Consistency**

This program, with the exceptions noted below, is consistent with NUREG-1801 XI.M034.

**Exceptions to NUREG-1801**

1. NUREG-1801, XI.M34 states "*The program relies on preventive measures such as coating, wrapping and periodic inspection for loss of material caused by corrosion of the external surface of buried steel piping and tanks*".

   Seabrook Station includes stainless steel piping in the Buried Piping and Tanks Inspection Program.

   <u>Justification for the Exception</u>

   The program will inspect buried stainless steel piping when it is excavated. The inspection methods used for buried cast iron and carbon steel are applicable to buried stainless steel piping as well. Buried stainless steel piping is more resistant to pitting and crevice corrosion than carbon steels and other materials addressed in NUREG-1801 XI.M34 when exposed to soil and inspection of the buried stainless steel piping will detect unacceptable loss of material. Buried stainless steel pipes will be inspected for loss of material due to pitting and crevice corrosion and microbiologically influenced corrosion (MIC).

   *Program Elements Affected: Element 1 (Scope of Program) and Element 3 (Parameters Monitored/Inspected).*

**Enhancements**

None

**Operating Experience**

The Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues and events. Those issues and events, whether external or plant specific, that are potentially significant to the Buried Piping and Tanks Program Inspection Program are evaluated. The Buried Piping and Tanks Program Inspection Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

1. Extensive visual inspection of buried Service Water system piping interior surfaces has been conducted since Refueling Outage 4 (Fall of 1995) with no indications of pipe wall degradation. The piping is cement lined, but degradation from the exterior surface is expected to lead to staining of the cement liner as water and corrosion products reach the inner surface of the pipe wall. When staining of the cement liner is found, the liner material is removed in order to evaluate the surface condition of the underlying pipe. Ultrasonic thickness measurements are taken to determine any degradation of the pipe wall. Such information, combined with no indication of interior pipe surface degradation, would indicate wall thinning cause by external environmental conditions. To date, there has been no indication of through wall leakage in the buried Service Water piping either originating from the inside surface or the outside surface. This example demonstrates one alternate indirect inspection method used to identify exterior surface degradation.

2. In November 2000, the Auxiliary Boiler buried fuel supply line was determined to be leaking diesel fuel into the surrounding soil. A small leak was discovered in the buried carbon steel pipe in an area where the bituminous wrap had been damaged. The fuel-contaminated soil was removed and with the concurrence of the New Hampshire Department of Environmental Services, the leak was temporarily repaired. In June 2001, after an examination of the failed section of pipe and visual/ultrasonic inspections at several excavations along the piping run further pipe deterioration was discovered and it was ultimately decided that the existing pipe would not be returned to service. A design change was initiated to replace the piping with dual-wall pipe meeting newly passed state requirements. A temporary modification was created to provide fuel oil during the period of implementation of this design change.

3. In March of 2001, a service vendor noticed oil drops coming from the ground around the fuel oil pumps at the vehicle maintenance shop. After excavation, the source of the leak was found to be at a threaded

joint. An evaluation of the condition determined that the most likely cause of the pipeline leakage was the loosening of the joints over time due to temperature changes and frost heaving. The pumping station and underground piping were removed and a new pumping station and dual-wall underground piping with leak detection capability installed. This piping is not in the scope of license renewal, but the example demonstrates appropriate investigation of and response to identified degradation of buried piping.

4. A branch connection was installed in a 6 inch buried Fire Protection system line in 2007. When excavated, the existing carbon steel pipe was inspected and showed no degradation of the coating or external surfaces.

5. Following excavation to repair a Fire Protection valve in September 2008, minor damage to the external tape coat on a 12" carbon steel Fire Protection line was found. Engineering was notified and the condition documented in the Seabrook Station Corrective Action Program. An inspection report was issued, which included documentation that the coating was worn but no metal (pipe) was exposed, and that there were no signs of backfill embedded in the coating. The coating was repaired and the area backfilled. This example demonstrates the appropriate notifications and inspections utilized when opportunistic observations detect evidence of conditions that could affect the integrity of buried piping.

6. Following an EPRI workshop on buried piping, the Seabrook Station attendees initiated the development of a buried piping program. A plan was developed and appropriate actions assigned in the Seabrook Station Corrective Action Program to implement this program. Specific actions were assigned in under the Corrective Action Program in November 2007. Underground piping was identified and inventoried by the respective system engineers, and a Buried Piping System Health Report generated. This health report is issued periodically by the assigned System Engineer. Seabrook Station procedures are being developed and will form the bases for this program.

These examples of Seabrook Station operating experience provide evidence that the Buried Piping and Tanks Inspection Program will adequately monitor the aging effects and that Seabrook Station is maintaining an awareness and sensitivity to operating experiences throughout the industry that could impact this program.

### Conclusion

The Seabrook Station Aging Management Program for Buried Piping and Tanks Inspection provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.23 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL BORE PIPING

### Program Description

The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program is a new program that will manage the aging effect of cracking in stainless steel small-bore ASME Code Class 1 piping less than 4 inches nominal pipe size, including pipe, fittings, and branch connections. While the ASME Boiler and Pressure Vessel Code, Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components",* does not require volumetric examination of Class 1 small-bore piping, the Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will be used to identify cracking due to stress corrosion cracking and thermal and mechanical loading by performing volumetric examinations of selected piping.

Seabrook Station has not experienced cracking of ASME Code Class 1 small-bore piping resulting from stress corrosion or thermal and mechanical loading.

NUREG-1801 Section XI.M35, *"One-Time Inspection of ASME Code Class 1 Small-Bore Piping",* includes piping *"less than or equal to NPS 4 inch"* with a reference to ASME Section XI, Table IWB-2500-1, Examination Category BJ, item number B9-21; however, according to the ASME Code, a volumetric examination already is required for piping equal to 4 inches nominal pipe size. Consistent with the Code, NUREG-1801 Item IV.C2-1 applies the One-Time Inspection of ASME Code Class 1 Small Bore Piping Program (XI.M35) only to Class 1 piping less than 4 inches nominal pipe size. On this basis, Seabrook Station concludes that the intent of the NUREG-1801 program is not to include 4 inch pipe.

The Seabrook Station ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program, B.2.1.1, currently includes volumetric examination of welds on Class 1 pipe 4 inches nominal pipe size and larger. The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will select a sample from the total population of ASME Code Class 1 small bore (less than 4 inches nominal pipe size) piping locations based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations using the recommendations of

MRP-146, *"Material Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines"*, or later updated guidance. The inspection sample determination will include both socket welds and butt welds. If non-destructive volumetric inspection techniques have not been qualified, Seabrook will have the option to remove the weld for destructive examination.

When a one-time inspection reveals evidence of cracking due to stress corrosion cracking or thermal and mechanical loading, evaluation of the inspection results will develop appropriate corrective actions which may include periodic inspections. Flaws or indications exceeding the acceptance criteria of ASME Section XI Paragraph IWB-3400 will be evaluated in accordance with ASME Section XI Paragraph IWB-3131, and any additional examinations will be performed in accordance with ASME Section XI Paragraph IWB-2430.

A count of Class 1 welds less than 4 inches nominal pipe size noted approximately 400 welds in the Reactor Coolant (RC), Chemical and Volume Control (CS) and Safety Injection (SI) systems. Approximately 25% of these are socket welds. The number of welds on pipe less than 2 inches nominal pipe size (which includes small branch connections) is less than 50% of the total population.

This program will be an inspection activity independent of methods to mitigate or prevent degradation. No preventative actions will be required.

The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will inspect for cracking in ASME Code Class 1 small-bore piping using volumetric examination techniques available. Should, evaluation of the inspection results indicate the need for additional examinations, such examinations will be consistent with ASME Section XI, Subsection IWB.

If flaws or indications exceed the acceptance criteria of ASME Code, Section XI, Paragraph IWB-3400, they will be evaluated in accordance with ASME Code, Section XI, Paragraph IWB-3131, and additional examinations will be performed in accordance with ASME Code, Section XI, Paragraph IWB-2430.

Repairs and replacements will be performed in accordance with applicable Section XI rules and requirements.

**NUREG-1801 Consistency**

This program, with the exception noted below, is consistent with NUREG-1801 Section XI.M35.

## Exceptions to NUREG-1801

1. NUREG 1801 XI.M35 states "*Guidelines for identifying piping susceptible to potential effects of thermal stratification or turbulent penetration are provided in EPRI report 1000701, 'Interim Thermal Fatigue Management Guideline (MRP-24)', January 2001.*"

   When identifying piping susceptible to potential effects of thermal stratification or turbulent penetration, Seabrook Station will follow the guidance issued as EPRI Report 1011955, *"Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)"* issued in June of 2005 and the supplemental guidance issued in EPRI Report 1018330, *"Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines - Supplemental Guidance (MRP-146S)"* issued in December of 2008.

   <u>Justification for the Exception</u>

   EPRI *MRP-24* was an interim report. This interim report was meant to provide early feedback to PWR plant operators prior to completion of the MRP project and provide a common industry approach that may be used to assess the potential for thermal fatigue cracking in piping systems where through-wall leakage has been observed in other plants in the past.

   EPRI *MRP-146* was issued in June of 2005 to expand on an interim guideline and to provide an ongoing fatigue management program in affected lines.

   EPRI *MRP-146S* issued in December of 2008 provides supplemental guidance for assessment of normally stagnant non-isolable reactor coolant system branch lines as required by MRP-146, including an implementation schedule for requirements. Seabrook Station will review and incorporate information as it is made available by EPRI.

   *Program Elements Affected: Element 1 (Scope of Program).*

## Enhancements

None

## Operating Experience

The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program is a new program. Both plant and industry operating experience will be used to establish the program and to ensure that this inspection uses volumetric inspection techniques with demonstrated capability and a proven industry record to detect cracking in piping weld and base material. The specific examination techniques utilized will be qualified prior to performing the examinations.

The Seabrook Station Second Ten-Year Period Inservice Inspections included volumetric examination of twenty-seven 2 inch and 3 inch Class 1 butt welds. No indication of cracking was noted in any of those locations. A search of the Seabrook Station condition reports also showed no reported degradation or failures in Class 1 piping less than 4 inches nominal pipe size.

**Conclusion**

The Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### B.2.1.24 EXTERNAL SURFACES MONITORING

**Program Description**

The Seabrook Station External Surfaces Monitoring Program is an existing program that manages the aging effects of (a) hardening and loss of strength due to elastomer degradation, (b) reduction of heat transfer due to fouling, (c) loss of material due to general, pitting, crevice, galvanic, and microbiologically influenced corrosion and due to fouling, and (d) loss of material due to wear; through visual inspection and non-visual tactile inspections of external surfaces. This program consists of periodic inspections of aluminum, cast austenitic stainless steel, copper alloy, copper alloy >15% zinc (Zn), elastomer, galvanized steel, gray cast iron, nickel alloy, stainless steel and steel components. The aging effect of loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion Program, B.2.1.4. The aging effect of loss of material for buried piping and components is managed by the Buried Piping and Tanks Program, B.2.1.22. The aging effect of loss of material for supports and structural components is managed by the Structures Monitoring Program, B.2.1.31.

The Seabrook Station External Surfaces Monitoring Program utilizes periodic system inspections and walk downs to monitor for materials degradation and leakage. This program inspects components such as piping, piping components, ducting and other components, including bolting, within the scope of license renewal. Coatings deterioration is monitored as an indication of possible underlying degradation.

The Seabrook Station External Surfaces Monitoring Program conducts visual inspection of component surfaces at least once per refueling cycle. This frequency accommodates inspections of components that may be in locations that are normally only accessible during outages. The intervals of inspections may be adjusted as necessary based on plant-specific inspection results and industry experience.

For surfaces that are insulated, inspections of opportunity will be performed to assess the external condition when insulation is removed for maintenance or inspection and the external surface is exposed. The insulated piping and components of interest are those within the scope of this program with low normal operating temperatures and in indoor or outdoor environment such that the piping could be wetted under its insulation.

The Seabrook Station External Surfaces Monitoring Program will require a periodic review of documented under-insulation inspection results to verify that there were a sufficient number of inspection opportunities to provide a representative indication of system condition, and to assess the need for further inspections.

Seabrook Station does not credit the External Surfaces Monitoring Program for age managing loss of material on internal surfaces.

The Seabrook Station External Surfaces Monitoring Program is a visual monitoring and inspection program that does not include preventive actions.

Examples of inspection parameters include:

a. corrosion and material wastage (loss of material),
b. leakage from or onto external surfaces,
c. worn, flaking, or oxide-coated surfaces,
d. corrosion stains on thermal insulation,
e. insulation damage or wetting,
f. protective coating degradation (blistering, cracking and flaking),
g. cracking or flaking of non-metallic components
h. accumulation of dirt or debris contributing to loss of heat transfer
i. surface breaking flaw (i.e., cracks or surface areas that have exhibited loss of material)
j. discontinuities and imperfections

Degradation of coated metallic surfaces cannot occur without the degradation of the paint or coating. Confirmation of the integrity of the paint or coating is an effective method for managing the effects of corrosion on the steel surface. No credit is taken for coatings to protect equipment from corrosion, but loss of integrity of coating can indicate evidence of corrosion.

Metallic components including aluminum, cast austenitic stainless steel, copper alloy, copper alloy >15% Zn, galvanized steel, gray cast iron, nickel alloy and stainless steel would exhibit indications of loss of material on the surface similar to steel material and visual inspections will be capable of detecting any surface breaking flaws (i.e., cracks or surface areas that have exhibited loss of material).

The program inspects for hardening and loss of strength in components made from elastomers by visual examinations to detect discontinuities and imperfections of the surface of the component, and non-visual examinations such as tactile techniques, which include scratching, bending, folding, stretching and pressing in conjunction with the visual examinations. Scratching the material will screen for residues that may indicate a breakdown of the elastomer material, bending or folding of the component may indicate surface cracking, stretching to evaluate resistance of the elastomer material and pressing on the material to evaluate the resiliency.

The program inspects for loss of material due to wear for elastomers only. The same visual examination and tactile techniques used to detect hardening and loss of strength in elastomers are used to detect wear in the components made from elastomers.

The program inspects for loss of material due to general, crevice and pitting corrosion using visual inspection.

The program inspects for loss of material due to galvanic corrosion. This aging mechanism will be detectible as the galvanic corrosion mechanism will create a corrosion product detectable by visual inspection similar to general corrosion.

The program inspects for reduction of heat transfer due to fouling. The program is only credited for management of heat transfer degradation due to fouling of the external surface of cooling coils that are exposed to an external air environment. Visual examinations performed as part of this program are capable of identifying corrosion, discoloration and accumulation of dirt/debris which are indicative of heat transfer degradation due to fouling.

Visual inspection activities are performed and associated personnel are qualified in accordance with site controlled procedures and processes. Personnel having the responsibility to perform these system specific walkdowns will be trained and qualified to perform these inspections.

Seabrook Station has existing guidance documents that support a monitoring and trending process to track degradation. The "*Systems Walkdowns*" guideline sets the parameters and expectations for conducting system engineer walkdown. The "*Performance Monitoring Guidelines*" provides the expectations for trending, analyzing and benchmarking system performance down to the component level. The "*System Health Reports*" guideline describes how the review of the system performance is documented.

Acceptance criteria will be applied to the results of corrective action evaluations and will include design standards, procedural requirements, current licensing basis, industry codes or standards, and engineering evaluation. The results of the evaluation will determine a threshold for action.

**NUREG-1801 Consistency**

This program, with the exceptions noted below, is consistent with NUREG-1801 XI.M36.

**Exceptions to NUREG-1801**

1. NUREG 1801 XI.M36 states *"This program visually inspects the external surface of in-scope components and monitors external surfaces of steel components in systems within the scope of license renewal and subject to AMR for loss of material and leakage."*

   Seabrook Station includes components made from additional materials such as aluminum, cast austenitic stainless steel (CASS), copper alloy, copper alloy >15% Zn, elastomer, galvanized steel, gray cast iron, nickel alloy, and stainless steel and have included them in the scope of this program.

   <u>Justification for the Exception</u>

   Seabrook Station has identified components made from materials other than steel that have surfaces exposed to an external environment. These components may also be subject to potential aging effects that should be managed under a license renewal aging management program.

   *Program Elements Affected: Element 1 (Scope of Program).*

2. NUREG-1801 XI.M36 states: "*This program is credited with managing the following aging effects.*

   a. *Loss of material for external surfaces;*

   b. *Loss of material for internal surfaces exposed to the same environment as the external surface*"

   NUREG-1801 XI.M36 also states *"Therefore, this program is acceptable for use in inspecting for loss of material for general, pitting and crevice corrosion."*

Seabrook Station includes the additional aging effects of hardening and loss of strength, reduction of heat transfer, and loss of material due to galvanic corrosion and wear.

Justification for the Exception

The NUREG-1801 program limits its discussion of those aging effects (general, pitting and crevice corrosion) that are likely to occur on the external surface of steel components, which is the singular material addressed by the program. Since Seabrook Station also includes components made from aluminum, cast austenitic stainless steel, copper alloy, copper alloy >15% Zn, elastomer, galvanized steel, gray cast iron, nickel alloy, and stainless steel, the additional aging effects of hardening and loss of strength, reduction of heat transfer, and loss of material due to galvanic corrosion and wear need to be addressed.

Aluminum, cast austenitic stainless steel, copper alloy, copper alloy >15% Zn, galvanized steel, gray cast iron, nickel alloy and stainless steel components would exhibit indications of loss of material on the surface similar to steel material and visual inspections will be capable of detecting any surface breaking flaws (i.e., cracks or surface areas that have exhibited loss of material) that occur on the same side as that being examined.

The program inspects for hardening and loss of strength in components made from elastomers by visual examinations to detect discontinuities and imperfections of the surface of the component, and non-visual examinations such as tactile techniques, which include scratching, bending, folding, stretching and pressing in conjunction with the visual examinations. Scratching the material will screen for residues that may indicate a breakdown of the polymer material, bending or folding of the component may indicate surface cracking, stretching to evaluate resistance of the elastomer material and pressing on the material to evaluate the resiliency.

The program also inspects for loss of material due to wear for elastomers only. The same visual examination and tactile techniques used to detect hardening and loss of strength in elastomers are used to detect wear in the components made from elastomers.

*Program Elements Affected: Element 1 (Scope of Program) and Element 4 (Detection of Aging Effects).*

## Enhancements

The following enhancement will be made prior to entering the period of extended operation.

1. Seabrook Station procedures will be enhanced to more specifically address the scope of the program, relevant degradation mechanisms and effects of interest, the refueling outage inspection frequency, the inspections of opportunity for possible corrosion under insulation, the training requirements for inspectors and the required periodic reviews to determine program effectiveness.

   *Program Elements Affected: Element 1 (Scope of Program), Element 3 (Parameters Monitored/Inspected), Element 5 (Monitoring and Trending), and Element 6 (Acceptance Criteria)*

## Operating Experience

The existing walkdowns at Seabrook Station have been effective in identifying leakage or corrosion in systems. The following operating experience demonstrates program effectiveness:

1. In July 2002, during a system walkdown, the system engineer noted that the Primary Component Cooling Water heat exchanger outlet lines were subject to external corrosion due to condensation. Further review of the condition by the System Engineer revealed that the type of insulation installed on the subject piping was incorrect and not in accordance with the piping specification. The system engineer identified that the subject piping should have the Armoflex anti sweat insulation instead of the fiberglass thermal insulation. The system engineer also identified that in its current condition, the condensation was being absorbed by the fiberglass insulation, creating a corrosion cell on the carbon steel piping substrate. As a result, the existing fiberglass insulation was removed and replaced with anti sweat type insulation. Additionally, upon removal of the fiberglass insulation, the carbon steel piping external surfaces were inspected to ensure that no unacceptable surface corrosion had taken place.

2. In July 2004, a condition report was initiated describing Service Water lines in the intake transition structure exhibiting a heavy coating of corrosion product over the entire length of piping. Ultrasonic thickness measurements were taken on the pipe, which showed no notable wall loss from external corrosion. This issue was presented to the plant health committee which approved funding for painting the pipe. Subsequently, the piping was painted in 2008 to preclude further corrosion.

3. In August 2004, a condition report was initiated to report external surface corrosion on Diesel Generator piping. Ultrasonic thickness measurements were taken and indicated pipe wall thickness below the original installation value. An engineering evaluation was performed to assess the degraded condition. The results of the engineering evaluation determined that the applicable design code requirements for all design conditions were satisfied, and therefore, the measured reduced wall thickness was determined to be acceptable.

   The condition report evaluation determined the cause of corrosion was due to condensation dripping from the Service Water system piping located directly above the Diesel Generator piping. As part of the corrective actions, the corroded piping was cleaned and recoated under a work order. To prevent recurrence, a sheet metal diverter was installed above the Diesel Generator piping per an engineering change to prevent condensation from dripping onto the Diesel Generator piping.

4. In August 2008, "B" Emergency Feedwater pump FW-P-37B seal water flange fastener nuts were reported to exhibit evidence of corrosion. Subsequently, the condition was evaluated by engineering and the cause of the corrosion was determined to be leakage from the pump vent, which was used when refilling the pump following maintenance. A work order was generated to correct the leakage from the vent cap on the pump. As part of the extent of condition review, the "A" Emergency Feedwater pump was also inspected and a similar leakage from the vent cap for this pump was noted. A second work order was initiated to correct the leakage from the vent cap on the "A" Emergency Feedwater pump.

The above examples provide objective evidence that when surface corrosion is identified, it is entered into the corrective action process so that corrective actions will be taken to address the issues. Appropriate guidance for evaluation, repair, or replacement is provided for locations where degradation is found. The previous examples of operating experience provide objective evidence that the Seabrook Station External Surfaces Monitoring Program will be effective in ensuring that intended function(s) will be maintained.

**Conclusion**

The Seabrook Station External Surfaces Monitoring Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.25 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS

### Program Description

The Seabrook Station Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program that will manage the aging effects of (a) cracking due to stress corrosion cracking, (b) loss of material due to general, pitting, crevice, galvanic and microbiologically influenced corrosion and due to fouling (c) loss of material due to erosion and wear (d) reduction of heat transfer due to fouling, and (e) hardening and loss of strength due to elastomer degradation. This program will consist of inspections of the internal surfaces of aluminum, cast austenitic stainless steel, copper alloy, copper alloy >15% Zn, elastomer, galvanized steel, gray cast iron, nickel alloy, stainless steel, and steel piping, piping components, ducting and other components that are not covered by other aging management programs.

The program inspections will be inspections of opportunity, performed during pre-planned periodic system and component surveillances or during maintenance activities when the systems are opened and the surfaces made accessible for visual inspection. This maintenance may occur during power operations or refueling outages when many systems are opened. The visual inspections will assure that existing environmental conditions are not causing material degradation that could result in a loss of the component intended function. The program will include indication of borated water leakage on internal surfaces. The Seabrook Station program will provide for visual inspection activities performed by personnel who are qualified in accordance with site controlled procedures and processes.

On implementation of this program, the maintenance planning process will include the opportunity for an internal inspection for work orders planned for the system or components identified as requiring aging management under the Seabrook Station Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. The program will then be implemented to the maximum extent practical.

All inspection results will be sent to the associated system engineer. The results of these inspections of opportunity will be evaluated and tracked as part of the system health assessment program and integrated with the plant corrective action process. Condition reports associated with a particular system will be subject to the system engineer's review on a periodic basis.

Periodic inspections will provide for detection of aging effects prior to the loss of component function. The inspections of internal surfaces will detect the aging effects of cracking, hardening and loss of strength, loss of material, and reduction of heat transfer. For painted or coated surfaces, degradation of steel surfaces cannot occur without the degradation of the paint or coating. Confirmation of the integrity of the paint or coating is an effective method for managing the effects of corrosion on the steel surface. Paint or coating degradation will be identified by the presence of blistering, cracking, rusting, loss of adhesion, and mechanical damage. For uncoated surfaces, visual inspections will directly monitor for surface degradation including indications of general corrosion.

The program will be an inspection and condition monitoring program; therefore no preventive actions or steps exist to mitigate or prevent component degradation.

Visual inspections of internal surfaces of plant components will be performed during maintenance or surveillance activities. The presence of corrosion or fouling will be identified by visual inspection as localized discoloration and surface irregularities such as rust, scale/deposits, surface pitting, surface discontinuities and coating degradation. Metallic components including aluminum, brass or bronze, cast austenitic stainless steel, copper alloy, copper nickel and stainless steel will exhibit indications of loss of material on the surface similar to steel material and visual inspections will be capable of detecting any surface breaking flaws (i.e., cracks or surface areas that have exhibited loss of material) that occur on the same side as that being examined.

The program will be used to detect cracking due to stress corrosion cracking in a limited number of stainless steel components exposed to steam or diesel exhaust. The Auxiliary Heating Steam System has some stainless steel components in steam while the Diesel Generator and Fire Protection systems have some stainless steel components with an internal environment of diesel exhaust. The inspection techniques utilized to detect this aging effect will be either visual inspection with a magnified resolution as described in 10 CFR 50.55a (b)(2)(xxi)(A) or an ultrasonic inspection method. The inspections will be performed by qualified personnel using proven techniques in accordance with Seabrook Station procedures and processes.

The program will be used to detect hardening and loss of strength in components made from elastomers by visual examinations and non-visual examinations such as tactile techniques, which include scratching, bending, folding, stretching and pressing in conjunction with the visual examinations. Scratching the material will screen for residues that may indicate a breakdown of the elastomer material, bending or folding of the component may indicate surface cracking, stretching to evaluate resistance of the elastomer material and pressing on the material to evaluate the resiliency.

The inspection results will be reviewed as part of the system health reports. The reviews will include tracking and trending over time to determine if an appropriate number of locations and inspection intervals will be able to provide reasonable assurance that the effects of aging will be adequately managed consistent with the current licensing basis for the period of extended operation.

The system engineer review of inspection results will help ensure that the extent and schedule of inspections and testing detect component degradation prior to loss of intended function.

Acceptance criteria for indications of various corrosion mechanisms or fouling will be identified in the appropriate inspection procedure and will be part of the training/qualification program required for inspectors. Visual inspection will monitor parameters such as corrosion, corrosion byproducts, coating degradation, discoloration on the surface, scale/deposits, pits and surface discontinuities. The degree to which these conditions exist will be used to establish baseline acceptance criteria for future inspections. For painted or coated surfaces, any evidence of damaged or degraded coating may be an indicator of corrosion damage to the surface underneath. Therefore, evidence of damaged or degraded coatings will be documented and evaluated using the Seabrook Station Corrective Action Program. For materials susceptible to corrosion heavy corrosion, localized corrosion, blistering, pitting, or visible loss of material due to corrosion will be documented and evaluated using the Seabrook Station Corrective Action Program. A thin, light, even layer of oxidation provides protection against further corrosion. Oxidation is expected in some systems, and is acceptable.

Other inspection results identified as having the potential to degrade the component or system intended function will also be documented and processed in accordance with the Seabrook Station Corrective Action Program.

**NUREG-1801 Consistency**

This program, with the exceptions noted below, is consistent with NUREG-1801 XI.M38.

**Exceptions to NUREG-1801**

1. NUREG 1801 XI.M38 states *"The program visual inspections include internal surfaces of steel piping, piping elements, ducting, and components in an internal environment (such as indoor uncontrolled air, condensation, and steam) that are not included in other aging management programs for loss of material. Inspections are performed when the internal surfaces are accessible during the performance of periodic surveillances, during maintenance activities or during scheduled outages".*

   The Seabrook Station program will also apply to components made from other materials such as aluminum, cast austenitic stainless steel, copper alloy, copper alloy >15% Zn, elastomer, galvanized steel, gray cast iron, nickel alloy, and stainless steel.

Justification for the Exception

Seabrook Station has identified components made from materials other than steel having surfaces exposed to internal environments which are not covered by other aging management programs. These components may also be subject to potential aging effects and should be managed under a license renewal aging management program. Seabrook Station has included these components in the Seabrook Station Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program.

*Program Elements Affected: Element 1 (Scope of Program).*

2. NUREG-1801 XI.M38 states "*Visual inspections of internal surfaces of plant components are performed during maintenance or surveillance activities. Parameters monitored or inspected include visible evidence of corrosion to indicate possible loss of materials*".

   The Seabrook Station program will include the additional aging effects cracking, reduction of heat transfer, and hardening and loss of strength.

   Justification for the Exception

   Since NUREG-1801 Section XI.M38 limits the scope of inspections to steel components with the aging effect of loss of material, aging effects for components made from other materials having other aging effects, are identified as exceptions to the program. NUREG-1801 Section XI.M38 requirements state that the applicant should identify and justify the inspection technique used for detecting the aging effects of concern. This discussion for the additional aging effects, other than loss of material, follows.

   The Seabrook Station program will allow for visual inspection to detect reduction of heat transfer due to fouling. The heat exchangers age managed under this program do not require precise determination of heat transfer capability, and a visual inspection of the heat exchanger internals will be able to determine whether or not the overall heat transfer function of the component is degraded.

   The program will be used to detect cracking due to stress corrosion cracking in a limited number of stainless steel components exposed to steam or diesel exhaust. The Auxiliary Steam Heating System has some stainless steel components in steam while the Diesel Generator Fire Protection systems have some stainless steel components with an internal environment of diesel exhaust. The inspection techniques utilized to detect this aging effect will be either visual inspection with a magnified resolution as described in 10 CFR 50.55a(b)(2)(xxi)(A) or an ultrasonic inspection method. Note that NUREG 1801 Section XI.M32 "One Time Inspection" recommends the use of an enhanced VT-1 visual inspection or ultrasonic inspection technique as an acceptable means to detect cracking due to stress corrosion cracking. The inspections will be performed by qualified personnel using proven techniques in accordance with Seabrook Station procedures and processes.

The program will be used to detect hardening and loss of strength in components made from elastomers by visual examinations and non-visual examinations such as tactile techniques, which include scratching, bending, folding, stretching and pressing in conjunction with the visual examinations. Scratching the material will screen for residues that may indicate a breakdown of the elastomer material, bending or folding of the component may indicate surface cracking, stretching to evaluate resistance of the elastomer material and pressing on the material to evaluate the resiliency.

*Program Elements Affected: Element 3 (Parameters Monitored/Inspected).*

**Enhancements**

None

**Operating Experience**

While the Seabrook Station Inspections of Internal Surfaces in Miscellaneous Piping and Ducting Components is a new program, Seabrook Station has been completing inspections of the internal surfaces of components during the performance of maintenance activities. During the normal course of maintenance, it is a natural part of the work process to identify the as-found conditions of the equipment or system that is the subject of the work activity. The FPL/NextEra Corporate Guidance for Maintenance sets expectations for documenting as-found conditions as part of the conduct of maintenance. This documentation includes non-conformances that needed to be resolved and any condition reports written.

The Seabrook Station Corrective Action Program has been successful in identifying potentially adverse conditions that were found when components or systems were opened for maintenance or surveillance. The corrective action process drives the evaluation and actions needed to resolve the identified condition. Below are examples of equipment internal surface degradation identified during maintenance and walk down activities.

1. In May of 2001, during an inspection of the Fire Protection fire water storage tank heat exchanger, 1-FP-E-46, evidence of corrosion was found inside the shell of the heat exchanger. The corrosion was evaluated and determined to be minor and no further action was required at that time.

   In August of 2001, during an inspection of the fire water storage tank heat exchanger, 1-FP-E-47, localized pitting was discovered inside the heat exchanger shell at the steam inlet similar to the condition previously observed in 1-FP-E-46. A weld repair was performed to repair the localized area of wall damage.

   In August 2005, during a repeat inspection of the fire water storage tank heat exchanger 1-FP-E-46 localized pitting was observed inside the shell of the heat exchanger. Plant engineering was notified and an ultrasonic examination was performed. Parts of the heat exchanger were below minimum wall thickness and a base metal repair was performed prior to returning the heat exchanger to service.

2. In February 2002, during a maintenance activity on Screen Wash System valve, 1-SCW-V-3, corrosion was found on the check valve internals. The valve was replaced. An extent of condition evaluation was performed, which identified two other valves (1-SCW-V-5 and 1-SCW-V-10) that could have the same condition as 1-SCW-V-3. As a result, preventive maintenance activities were developed for the disassembly and inspection of all three valves with a frequency of every six years. Subsequently, the disassembly and inspection of 1-SCW-V-5 was performed in April 2003. The disassembly and inspection of 1-SCW-V-10 was performed in November 2004. The second disassembly and inspection of 1-SCW-V-3 was performed in December 2008.

3. During Refueling Outage 12 (Spring of 2008), an area of concrete liner degradation was detected inside the Service Water cement lined carbon steel piping during internal inspections of AMEX-10/WEKO elastomer seals. The base metal was inspected and no evidence of metal loss was detected. The effected area of the liner was repaired with an epoxy coating material prior to being returned to service.

4. In January 2009, during Train "B" Diesel Generator outage, a black material similar to sand and small corrosion products were found in the jacket water cooling piping. A material sample was sent out for analysis, the review of the report showed similar material results to previous condition report findings for the system. There was nothing in the evaluation indicating that this was a new issue. The evaluation concluded that the particles are small enough that no blockage would be expected in any of the smaller passages of the engine. This issue is currently being trended by the System Engineer.

The above operating experience examples provides objective evidence that existing maintenance activities will identify internal degradation prior to loss of system components intended functions.

**Conclusion**

The Seabrook Station Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### B.2.1.26 LUBRICATING OIL ANALYSIS

**Program Description**

The Seabrook Station Lubricating Oil Analysis Program is an existing program that performs oil condition monitoring activities to manage the aging effects of loss of material due to galvanic, general, pitting, crevice, and microbiologically influenced corrosion, and fouling and heat transfer degradation due to fouling. Seabrook Station has no components within the scope of this program with an aging mechanism of cracking.

The purpose of the Seabrook Station Lubricating Oil Analysis Program is to obtain and analyze lubricating oil samples from plant equipment to ensure that the oil quality is maintained within established limits. The frequency of monitoring will vary depending on such factors as regulatory guidance, Technical Specification requirements of the equipment being worked on, vendor recommendations, continuous versus standby use, plant and industry experience with similar equipment, engineering analysis of equipment performance, the relative importance of the equipment to plant operation/ safety, and the maintenance history of the equipment.

The Seabrook Station Lubricating Oil Analysis Program includes sampling and analysis of lubricating oil for components within the scope of license renewal and subject to aging management review, that are exposed to lubricating oil and for which pressure boundary integrity or heat transfer is required for the component to perform its intended function. The One-Time Inspection Program (B.2.1.20) is used to verify the effectiveness of this program for managing the aging effects of piping and components in systems that contain lubricating oil.

The Seabrook Station Lubricating Oil Analysis Program provides an early indication of adverse equipment condition in lubricating oil environments. This program is administered and controlled by Seabrook Station maintenance program and procedures.

For components that undergo periodic oil changes in accordance with manufacturer's recommendations, a particle count and check for water are performed to detect evidence of abnormal wear rates, moisture contamination, or excessive corrosion. For components that do not have regular oil changes, the Seabrook Station Lubricating Oil Analysis Program determines percent water, viscosity, neutralization number and fuel dilution as applicable to verify the oil is suitable for continued use. In addition, analytical ferrography and elemental analysis are performed to identify wear particles.

Periodic oil sampling and compliance with the established acceptance criteria provide assurance that oil contaminants do not exceed acceptable levels. This practice preserves an environment that is not conducive to aging mechanisms that could lead to the aging effects of loss of material and heat transfer degradation.

Oil analysis results are reviewed to determine if alert levels or limits have been reached or exceeded. This review also checks for unusual trends. The Condition Based Maintenance Engineer determines routine test requirements for each equipment type, determines additional testing when need is indicated by test results, trend test results and in conjunction with other predictive maintenance program owners, makes recommendation to system engineering for maintenance or other corrective actions.

Sample analysis reports are reviewed to verify that contaminants (water and particulates) do not exceed limits based on manufacturer's recommendations or industry standards recommended for each component type.

Particle size and count or concentration is determined in accordance with industry standard ASTM D 6224, *"Standard Practice for In-Service Monitoring of Lubricating Oil for Auxiliary Power Plant Equipment"*. ASTM D 6224 particle count is based on ISO 4406, *"Hydraulic Fluid Power - Fluids - Method for Coding the Level of Contamination by Solid Particles"*. Water and particle concentration is tested, not to exceed limits based on manufacturer's recommendation and industry standard recommendation for each component type.

Viscosity bands are based on a tolerance around the manufacturer's typical viscosity value, baseline value or point reference of the lubricating oil as recommended by the component manufacturer, analysis of equipment data or industry standards.

Metal limits as determined by spectroscopic analysis (spectral analysis or ferrography) are based on original baseline data and manufacturer's recommendations and industry standards.

## NUREG-1801 Consistency

This program, with the exception noted below, is consistent with NUREG-1801 XI.M39.

## Exceptions to NUREG-1801

1. NUREG-1801 XI.M39 states *"For components with periodic oil changes in accordance with manufacturer's recommendations, a particle count and check for water are performed to detect evidence of abnormal wear rates, contamination by moisture, or excessive corrosion. For components that do not have regular oil changes, viscosity, neutralization number, and flash point are also determined to verify the oil is suitable for continued use. In addition, analytical ferrography and elemental analysis are performed to identify wear particles".*

   Seabrook Station does not sample for flash point in lubricating oil samples.

   <u>Justification for Exception</u>

   Testing for flash point of lubricating oil is only needed for lubricating oil that could become contaminated by fuel. Lubricating oil in many applications is not subject to this contamination (such as steam driven turbines or motor driven pumps) and testing for flash point is not needed. When there is no potential for contamination the lube oil will not be tested for flash point. In fact, ASTM D6224, *"Standard Practice for In-Service Monitoring of Lubricating Oil for Auxiliary Power Plant Equipment"*, states that flash point testing is optional for diesel engines.

   When there is a potential for contamination by fuel Seabrook Station will test the samples for fuel dilution. This is equivalent to testing for flash point as either test will provide an indication of fuel in-leakage. An example of the ability to detect fuel in-leakage using this method is included in operating experience.

   *Program Elements Affected: Element 3 (Parameters Monitored/Inspected).*

## Enhancements

The following enhancements will be made prior to entering the period of extended operation.

1. The Seabrook Station Lubricating Oil Analysis Program will be enhanced to provide an attachment with required equipment, and include the lube oil analysis required, sampling frequency, and required periodic oil changes.

*Program Elements Affected: Element 3 (Parameters Monitored/Inspected) and Element 4 (Detection of Aging Effects)*

2. The Seabrook Station Lubricating Oil Analysis Program will be enhanced to sample the oil for the Switchyard $SF_6$ compressors and the Reactor Coolant pump oil collection tanks.

   *Program Elements Affected: Element 4 (Detection of Aging Effects)*

3. The Seabrook Station Lubricating Oil Analysis Program will be enhanced to require the performance of a one-time ultrasonic thickness measurement of the lower portion of the Reactor Coolant pump oil collection tanks prior to the period of extended operation.

   *Program Elements Affected: Element 4 (Detection of Aging Effects)*

**Operating Experience**

Seabrook Station has a comprehensive Operating Experience Program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events. While no instances of component failure attributed to lubricating oil contamination or degradation were identified, the following examples demonstrate that the Seabrook Station Lubricating Oil Analysis Program provides prudent recommendations and appropriate corrective actions based on oil analysis results. The program provides input to preventive and corrective maintenance activities to ensure the integrity of the components within the scope of the program.

1. In May 2001, following an oil change in Train "B" Service Water Cooling Tower fan gear boxes, both fans were operated for approximately 1 hour. Vibration data was collected while the fans were running and oil samples were taken within 10 minutes of shut-down. The oil sample taken from fan #2 gear box was much darker in color than the sample from fan #1 gear box. A ferrographic examination of the two samples showed that there was significantly more wear debris in the oil from fan #2. Both fans had similar vibration magnitudes and spectrums and were considered acceptable. The condition was entered into the Corrective Action Program and another oil change was requested.

   During the requested oil change, the gear surfaces were inspected. No abnormal wear was identified. A build up of dark residue, possibly byproducts of oil subjected to elevated temperatures, was noted around the thermowell of the gearbox heater. A sample of this residue was sent to an outside lab for analysis. The results of the analysis showed no increase in iron concentration in the oil sample but noted a polymer gel in the analysis of the residue sample. The lab noted that a friction polymer gel can be created by gear forces which would be embedded with wear particles or a thermal polymer gel can form from exposure to heat sources. Since the polymer did not have embedded wear particles, the lab concluded that the material was created by exposure to heat as initially suspected. To preclude contamination of the oil reservoir with this polymer, the gearbox was cleaned and the gearbox heater was replaced.

In June 2001, after gearbox cleaning and subsequent refilling of reservoir, new samples were taken during fan operation. The results of these samples showed normal oil signatures. The actions taken to clean the heater element and gearbox were successful in correcting the condition. Normal monitoring was re-established.

2. In May 2002, an oil sample was taken from Fire Protection pump, 1-FP-P-20A, per the predictive maintenance program. The oil analysis results indicated that the total base number had decreased indicating the oil was near the end of its life. Based on this test, the oil was changed in May 2002. Following the oil change, a review was performed to adjust the oil change frequency on the Fire Protection pumps (1-FP-P-20A and 20B). Subsequently, the preventive maintenance program was revised to change the oil change frequency on these pumps from *"upon request"* to *"every 18 months"*.

3. In May, 20004, an oil sample taken from 1-FW-P-37A pump outboard bearing indicated elevated iron content and wear particles. Subsequently, vibration data was taken and showed no abnormal trends. Based on discussions with two different laboratories and the FPL St. Lucie program supervisor, the condition report evaluation concluded that the pump outboard bearing was the source of the elevated iron and wear particles. A recommendation was made to inspect or replace the pump outboard bearing. Immediate replacement was not deemed necessary at that time. Accordingly, in December 2004, the pump outboard bearing was replaced on 1-FW-P-37A.

4. In May 2004, the oil sample taken from the "D" Primary Component Cooling Water pump motor inboard bearing indicated that the copper and particle count had increased. Evaluation of the condition determined that the cause of the problem was due to wear of the motor inboard bearing slinger ring. Subsequently, the slinger ring was replaced.

5. On May 5, 2005, an oil sample taken from the "B" Emergency Diesel Generator rocker arm lube oil tank indicated a fuel dilution of 3.7%. No detectable fuel oil should have been present in the rocker arm lube oil system.

An apparent cause evaluation was performed, which concluded that the fuel injector was found to be leaking across the injector tip to injector body mating surface. This mating surface also aligns the cooling water passages. It was determined that the condition was most likely caused by one of two factors:

a. Improper mating surface of the injector tip to injector body.

b. The injector tip nut was under torqued.

As part of the extent of condition review, all sixteen injectors on the "B" Emergency Diesel Generator were observed at full load with the rocker housing covers removed. The two suspect injectors were removed from the "B" Emergency Diesel Generator and tested. There had been no indications of fuel dilution in "A" Emergency Diesel since these nozzles were installed in October of 2003. The injector nozzle maintenance endorsed by the vendor and the owners group was performed on the injectors installed in the "A" Emergency Diesel Generator. Hence, no further actions were deemed necessary on the "A" Emergency Diesel Generator.

Corrective actions included 1) the maintenance procedure was updated to check the torque value of the injector tip nut during Emergency Diesel Generator engine injection nozzle maintenance and 2) the vendor was requested to conduct an evaluation of the failed injector to the determine the cause of this condition.

The vendor review of the condition determined that the tip nut was under torqued. This would have occurred at the vendor's factory because Seabrook Station does not take the tip off of the injector body. As stated above, Seabrook Station had already incorporated a procedure step to check the tip nut torque prior to installation into the Diesel Generator, which should prevent a recurrence of this condition.

These examples provide objective evidence that the Seabrook Station Lubricating Oil Analysis Program effectively manages aging effects in systems and components within the scope of license renewal by maintaining oil quality within established limits.

**Conclusion**

The Seabrook Station Lubricating Oil Analysis Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis during the period of extended operation.

## B.2.1.27 ASME SECTION XI, SUBSECTION IWE

### Program Description

The Seabrook Station ASME Section XI, Subsection IWE Program is an existing program and performs inspections using the same primary Inservice Inspection method as specified in ASME Section XI, Subsection IWE, *"Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants"*, visual examination (general visual, VT-3, VT-1). IWE specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

Seabrook Station is in the first inspection interval which utilizes the requirements of the 1995 Edition, including the 1996 Addenda, of ASME Section XI, Sub-Section IWE. The components managed by the program include the containment liner, electrical penetrations, mechanical penetrations (piping, ventilation, and spares), personnel lock, equipment hatch, recirculation sump, reactor pit, moisture barriers, seals, gaskets, pressure retaining bolting, and supports.

NUREG-1801, Rev 1, discusses the use of the 2001 edition including the 2002 and 2003 addenda of ASME Section XI code, but allows use of other editions of the ASME Code as long as there is justification. The Seabrook Station ASME Section XI, Subsection IWE Program for the first ten-year inspection interval effective from August 19, 2000 through August 18, 2010, approved per 10 CFR 50.55a, is based on the 1995 edition including the 1996 addenda. The next and subsequent 120-month inspection intervals for Seabrook Station will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval. The Seabrook Station ASME Section XI, Subsection IWE Program is consistent with NUREG-1801 XI.S1.

### NUREG-1801 Consistency

This program is consistent with NUREG-1801 XI.S1.

### Exceptions to NUREG-1801

None

### Enhancements

None

### Operating Experience

There is considerable industry operating experience; NRC Information Notice (INs) 86-99, *"Degradation of Steel Containments"*, 88-82, *"Torus Shells with Corrosion and Degraded Coatings and Degraded Coatings in BWR Containments"*, and 89-79, *"Degraded Coatings and Corrosion of Steel Containment Vessels"*, that described occurrences of corrosion in steel containment shells. More recently, NRC IN 97-10, *"Liner Plate Corrosion in Concrete Containment"*, identified specific locations where concrete containments are susceptible to liner plate corrosion.

Seabrook Station engineering has reviewed the containment liner issue from Beaver Valley where inspections during the Beaver Valley refueling outage RFO17 (2006) revealed degradation from the inaccessible side of the steel liner. In addition to the ASME Section XI IWE examinations, Seabrook Station performs routine containment liner visual inspections on a 40 month frequency. No potentially through-liner corrosion issues have been noted.

Several condition reports were found in a search for "*containment liner*" documenting material found to be in contact with the containment liner (e.g., scaffolding, grating hose reels, and outage contractor storage boxes) during outage activities. These were promptly and appropriately dispositioned.

Seabrook Station Nuclear Oversight performed an audit of key activities during Refueling Outage 7 (Fall of 2000). There was only one material related observation from this audit stating that there was a faulty moisture barrier documented at the minus 26' level, azimuth 250°. A condition report was initiated and the barrier repaired.

An IWE inspection in November 2000 documented an observation of debris on top of the moisture barrier behind the shield at the -26' elevation, approx. 200 to 240 degrees. The area required vacuuming in order to complete the IWE examination. This condition report documents the degree of compliance and attention to detail exhibited during implementation of this program.

A November 2001 condition report documented the opportunity that was recognized to complete IWE examinations for an area that had been inaccessible during the refueling outage window for containment liner inspection. The containment recirculation sump had been barricaded during the outage for modifications that were in progress. Upon completion of the sump modifications, the barriers were removed. Station personnel recognized that this now provided an opportunity to complete this portion of the examination. This condition report documents the understanding of the scope of this program and the ability to recognize respective windows of opportunity.

## Conclusion

The Seabrook Station ASME Section XI, Subsection IWE Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

# B.2.1.28 ASME SECTION XI, SUBSECTION IWL

## Program Description

The Seabrook Station ASME Section XI, Subsection IWL Program is an existing program that manages the aging effects of cracking, loss of bond, loss of material (spalling, scaling) due to corrosion of embedded steel, expansion and cracking due to reaction with aggregates, increase in porosity and permeability, cracking, loss of material (spalling, scaling) due to aggressive chemical attack, and increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide and invokes the requirements of ASME Section XI, Sub-Section IWL, "*Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants*". The components managed by the program include steel reinforced concrete for the Seabrook Station containment building and complies with the requirement for examination contained in 10 CFR 50.55a in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL.

NUREG-1801, Rev 1, discusses the use of the 2001 edition including the 2002 and 2003 addenda of ASME Section XI code, but allows use of other editions of the ASME Code as long as there is justification. The Seabrook Station Inservice Inspection Program Plan for the current ten-year inspection interval effective from August 19, 2000 through August 18, 2010, approved per 10 CFR 50.55a, is based on the 1995 edition including the 1996 addenda. The next and subsequent 120-month inspection intervals for Seabrook Station will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

The primary inspection methods used at Seabrook Station are VT-1C visual examination, VT-3C visual examination and alternative examination methods (in accordance with IWA-2240). The Seabrook Station ASME Section XI, Subsection IWL Program provides acceptance criteria and corrective actions for each exam type.

As discussed in the NUREG-1801, Chapter 2, plants with aggressive groundwater/soil, and/or where the concrete structural elements have experienced degradation, a plant specific aging management program to account for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

Concrete degradation due to aggressive chemical attack is an aging effect applicable to Seabrook Station. The Seabrook Station Structures Monitoring Program (B.2.1.31) addresses the plan and specific details to determine the effects of aggressive chemical attack on the concrete. An evaluation will be performed after the testing performed in the plan and, if required, actions will be provided using the corrective action process for concrete under Seabrook Station Structures Monitoring and ASME Section XI, Subsection IWL programs.

The Seabrook Station containment is a steel reinforced concrete structure. No prestressed concrete or unbonded post-tensioning systems are used in the Seabrook Station containment.

All accessible containment reinforced concrete components are within the scope of this program. Inaccessible containment concrete portions are exempted from examination (e.g., concrete covered by liner, foundation material, or backfill, or obstructed by adjacent structures or other components) as per Subsection IWL. 10 CFR 50.55a(b)(2)(viii) specifies additional requirements for inaccessible areas. Seabrook Station will evaluate the acceptability of concrete in inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. Steel liners and their integral attachments are not within the scope of Subsection IWL, but are included within the scope of ASME Section XI, Subsection IWE (B.2.1.27).

The Seabrook Station ASME Section XI, Subsection IWL Program is an inspection program and no preventive actions are specified. Seabrook Station does not credit a coating program for managing the effects of aging of concrete surfaces.

Seabrook Station procedures provide instructions to perform visual examination of the concrete surfaces of the primary containment in accordance with requirements of Subsection IWL-2500. Seabrook Station does not have concrete surfaces surrounding tendon anchors. Concrete surfaces are inspected by VT-3C visual examination for evidence of damage or degradation such as;

a. Chemical attack, abrasion or erosion sufficient to expose coarse aggregate.

b. Water flowing from, or on the surface of, the concrete (except basement Annulus).

c. Scaling and/or disintegration sufficient to expose coarse aggregate.

d. Cracks, spalls, voids or popouts.

e. Efflorescence, exudation and/or encrustation.

f. Discoloration indicative of corrosion of embedded steel.

g. Exposure of reinforcing steel.

h. Cracking, blistering and/or peeling of coatings.

The scope of examinations is in compliance with 10 CFR 50.55a and Subsection IWL to ensure that aging effects would be detected before they would compromise the design-basis requirements. The frequency of examinations is five years. All accessible concrete surfaces receive a VT-3C visual examination. Areas detected during the VT-3C exams that indicate suspect conditions, receive a more rigorous VT-1C examination. These visual examination methods and testing would identify the aging effects of accessible concrete components at Seabrook Station containment.

Except in inaccessible areas, all concrete surfaces are monitored on a regular basis by virtue of the examination requirements. Seabrook Station procedures provide monitoring and trending information over the life of the plant.

Acceptance criteria in accordance with IWL-3000 for concrete containment are provided in Seabrook Station procedures. For concrete surfaces, the acceptance criteria rely on the determination of the "*Responsible Engineer*" regarding whether there is any evidence of damage or degradation sufficient to warrant further evaluation or repair in accordance with IWL-3300. The acceptance criteria are qualitative. Seabrook Station procedures also require that the Responsible Engineer be a registered professional engineer experienced in evaluating the inservice condition of structural concrete and knowledgeable of the design and construction codes and other criteria used in design and construction of concrete containments.

Repair activities are performed on the concrete containment as specified in Subsection IWL-4000. Testing performed following repair of modifications is done in accordance with Subsection IWL-5000.

**NUREG-1801 Consistency**

NUREG-1801, Rev 1, discusses the use of the 2001 edition including the 2002 and 2003 addenda of ASME Section XI code, but allows use of other editions of the ASME Code as long as there is justification. The Seabrook

Station Inservice Inspection Program Plan for the current ten-year inspection interval effective from August 19, 2000 through August 18, 2010, approved per 10 CFR 50.55a, is based on the 1995 edition including the 1996 addenda. The next and subsequent 120-month inspection intervals for Seabrook Station will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

This program is consistent with NUREG-1801 XI.S2.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancement will be made prior to entering the period of extended operation.

1. The Seabrook Station ASME Section XI, Inservice Inspection, Subsection IWL Program implementing procedures will be enhanced to include the definition of *"Responsible Engineer"*.

   *Program Elements Affected: Element 6 (Acceptance Criteria).*

**Operating Experience**

Seabrook Station has a comprehensive Operating Experience Program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events.

Preventive maintenance work orders are used for tracking and identifying conditions identified during surveillances. Issues and events, whether external or plant specific, that are potentially significant to containment reinforced concrete at Seabrook Station, or which show deficiencies in excess of acceptance criteria are evaluated.

The Seabrook Station ASME Section XI, Subsection IWL Program is implemented through the Seabrook Station Containment Surface Inspection Program. Some of the results of the inspection conclusions are based on the following reviews:

1. Containment inspections performed during Refueling Outage 8 (Spring of 2002) were completed satisfactorily with no indication of degradation of the concrete surfaces.

2. Containment inspections performed during Refueling Outage 10 (Spring of 2005) were completed satisfactorily with no indication of degradation of the concrete surfaces.

3. Containment inspections performed during Refueling Outage 12 (Spring of 2008) were completed satisfactorily with no indication of degradation of the concrete surfaces.

The Containment Structure concrete has been found to be in good condition during inspections performed in accordance with ASME Section XI, Subsection IWL. There is sufficient confidence that the implementation of the ASME Section XI, Subsection IWL program will effectively identify degradation prior to failure. Appropriate guidance for reevaluation, repair, or replacement is provided if degradation is found.

**Conclusion**

The Seabrook Station ASME Section XI, Subsection IWL Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.29 ASME SECTION XI, SUBSECTION IWF

**Program Description**

The Seabrook Station ASME Section XI, Subsection IWF Program is encompassed by Seabrook Station *"Inservice Inspection Reference Manual"*, Inservice Inspection (ISI) Visual Examination Procedure, and Station Procedure *"Inservice Inspection of Class 1, 2, and 3 Components"*, which invoke the requirements of ASME Section XI, Sub-Section IWF, "*Requirements for Class 1,2,3 and MC Component Supports of Light-Water Cooled Power Plants*". The program specifies the percentage of supports that must be examined. For supports, other than piping supports, the supports of only one component of a group having similar design, function, and service must be examined. Supports of piping and other items exempted from volumetric or surface examination are also exempt.

The Code of Federal Regulations, 10 CFR 50.55a, *"Codes and Standards"*, requires that Inservice Inspection of ASME Code Class 1, 2, and 3 pressure retaining components, their integral attachments and supports be conducted in accordance with the latest edition of ASME Section XI approved by the NRC twelve months prior to the start of a ten-year interval. The Inservice Inspection Program for the second (2$^{nd}$) ten-year interval, which began on August 19, 2000 for Seabrook Station, implements the 1995 edition with the 1996 addenda, of ASME Section XI. The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives.

As specified by the Inservice Inspection Reference Manual, the program uses VT-3 visual examination for detection of degradation. The performance requirements for VT-3 visual examination are provided in Inservice Inspection (ISI) Visual Examination Procedure. Per this procedure, VT-3 visual examinations are conducted to determine the general mechanical and structural condition of components including bolting and their supports by verifying parameters such as clearances, settings, and physical displacement, and to detect discontinuities and imperfections, such as loss of integrity of bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

The Seabrook Station ASME Section XI, Subsection IWF Program is implemented on a 10-year cycle. The program implements the requirements for inservice inspection of ASME Class 1, 2, and 3 supports.

The sample of pipe supports to be inspected is based on Table IWF-2500-1 which specifies that 25% of the Class 1 supports for the nonexempt Class 1 piping; 15% of the Class 2 supports for the nonexempt Class 2 piping; 10% of the Class 3 supports for the nonexempt Class 3 piping; and 100% of the Class 1, 2, 3, and MC supports other than piping supports are to be examined.

The supports to be selected and examined for components (vessels, pumps, etc.), are the supports for the components that are required to be examined under IWB-2500, IWC-2500, IWD-2500, and IWE-2500.

Supports exempt from the examination requirements of IWF-2000 are those connected to piping and other items exempted from volumetric, surface, or VT-1 or VT-3 visual examination by IWB-1220, IWC-1220, IWD-1220, and IWE-1220. In addition, portions of supports that are inaccessible by being encased in concrete, buried underground, or encapsulated by guard pipe are also exempt from the examination requirements of IWF-2000, in accordance with IWF-1230.

The same supports are inspected in each 10-year inspection interval, to the extent practical. The examinations listed above are based on specific Code Editions and Addenda and may change throughout the extended period as required and/or allowed by 10 CFR 50.55a.

The Seabrook Station ASME Section XI, Subsection IWF Program is an inspection program and no preventive actions are specified.

For piping and component support inspections, unacceptable conditions, as described in Seabrook Station ASME Section XI, Subsection IWF Program, are noted for correction or further evaluation. Supports are not monitored and trended for time dependent degradation.

The Seabrook Station ASME Section XI, Subsection IWF Program for component supports utilizes the acceptance standards for visual examination specified in ASME Section XI, Subsection IWF-3410. The following conditions have been identified as unacceptable:

a. Deformations or structural degradations of fasteners, springs, clamps, or other support items;
b. Missing, detached, or loosened support items;
c. Arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on close tolerance machined or sliding surfaces;
d. Improper hot or cold positions of spring supports and constant load supports;
e. Misalignment of supports; and
f. Improper clearances of guides and stops.

Additional examinations will be performed in accordance with IWF-2430 when examinations reveal indications exceeding the acceptance standards. If a component support is accepted for continued service in accordance with IWA-2110(a)(1)(i) and IWF-2420(b) and (c) the component support will be reexamined during the next inspection period.

**NUREG-1801 Consistency**

NUREG-1801, Rev 1, discusses the use of the 2001 edition including the 2002 and 2003 addenda of ASME Section XI code, but allows use of other editions of the ASME Code as long as there is justification. The Seabrook Station Inservice Inspection Program Plan for the second ten-year inspection interval effective from August 19, 2000 through August 18, 2010, approved per 10 CFR 50.55a, is based on the 1995 edition including the 1996 addenda. The next and subsequent 120-month inspection intervals for Seabrook Station will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

This program is consistent with NUREG-1801 XI.S3.

**Exceptions to NUREG-1801**

None

## Enhancements

None

## Operating Experience

A review of the plant specific operating experience found instances of selected supports being deficient, but operable. These conditions were reported and evaluated in accordance with the Corrective Action Program. There is reasonable assurance that the Seabrook Station ASME Section XI, Subsection IWF Program will be effective through the period of extended operation. Examples supporting this conclusion include:

1. Inservice Inspection activities conducted during Refueling Outage 5 (Spring of 1997) resulted in thirteen condition reports documenting thirty-six support deficiencies that were identified in *"OR05 (Refueling Outage 5) Inservice Inspection Document Package"*. All deficiencies were evaluated and dispositioned by Engineering.

2. In 1999, an *"Inservice Inspection Examination Report"* was generated to document the results of the Inservice Inspections conducted during Refueling Outage 6 (Spring of 1999). The report identifies five supports with deficient conditions. These were reported to Engineering on condition reports. All were found to be acceptable by engineering evaluation, based on conformance with design tolerances.

3. During Refueling Outage10 (Spring of 2005), twenty eight supports were selected for Inservice Inspection. According to *"OR10 (Refueling Outage 10) Inservice Inspection Document Package"*, no deficiencies requiring further evaluation were identified.

These examples provide objective evidence that the Seabrook Station ASME Section XI, Subsection IWF Program is effective in identifying and evaluating conditions which may challenge the ability of ASME Section XI supports to perform their intended function.

## Conclusion

The Seabrook Station ASME Section XI, Subsection IWF Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.30 10 CFR PART 50 APPENDIX J

### Program Description

The Seabrook Station 10 CFR Part 50, Appendix J Program is an existing performance based containment leak rate test program as described in the Seabrook Station Technical Requirements Program and Leakage Test Reference Manual.

The Seabrook Station Containment Leakage Rate Testing Program, required by Seabrook Station Technical Specification, implements Option B of Appendix J of 10 CFR Part 50, *"Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"*. The test requirements of Appendix J provide for periodic verification by tests of the leak-tight integrity of the primary reactor containment. The purposes of the tests are to assure that 1) leakage through the containment or systems and components penetrating the containment does not exceed the allowable leakage rate specified in the Technical Specifications and Updated Final Safety Analysis Report, and 2) integrity of the containment structure is maintained during its service life.

10 CFR Part 50 Appendix J, Option B applies guidance provided in NRC RG 1.163, *"Performance-Based Containment Leak-Test Program"*, for performance based leak testing. NRC RG 1.163 accepts methods provided in NEI 94-01, *"Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J"*, for implementing performance based testing with restrictions. NRC RG 1.163 also accepts the methods and techniques for performing Type A, B and C tests contained in ANSI/ANS-56.8-1994, *"Containment System Leakage Testing Requirements"*. The Seabrook Station Leakage Test Reference is based on the guidance provided in NEI 94-01 and ANSI / ANS-56.8-1994 with the restrictions identified in RG 1.163.

The Seabrook Station 10 CFR Part 50 Appendix J Program includes integrated and local leak rate tests of components that make up the primary containment pressure boundary. The program includes Type A, B, and C type testing as described in 10 CFR Part 50, Appendix J.

The Seabrook Station 10 CFR Part 50 Appendix J Program is a containment leak rate monitoring program and does not specify preventive actions. The test requirements of Appendix J provide for periodic verification by tests of the leak-tight integrity of the primary reactor containment. The Seabrook Station 10 CFR Part 50 Appendix J Program, in conjunction with the implementation of programs ASME Section XI, Subsection IWE Program (B.2.1.27), and ASME Section XI, Subsection IWL Program (B.2.1.28), provides an aging management program that is effective at detecting degradation of the containment boundary.

Periodic integrated leakage rate tests (Type A tests) are conducted in accordance with the UFSAR Section 6.2.6, Containment Leak Rate Testing and Technical Specification 4.6.1, Primary Containment - Containment Integrity. These tests monitor leakage rates through primary containment shells, liners, penetrations, fittings, access openings, and the isolation valves.

Type B tests are required on all containment penetrations with resilient seals, gaskets, or expansion bellows. These include, but are not limited to, air locks, air lock door seals, piping penetrations with expansion bellows and blind flanges, and electrical seals.

Type C tests are required on all lines that penetrate the primary containment and present a potential leakage path between the inside and outside atmospheres of the primary containment under postulated accident conditions.

The Seabrook Station acceptance criteria for containment leakage rates are defined in plant technical specifications and technical requirements. The Seabrook Station 10 CFR Part 50 Appendix J Program ensures that the containment leakage meets the defined acceptance criteria.

During Appendix J testing, if leakage rates do not meet the acceptance criteria, corrective actions are taken in accordance with 10 CFR Part 50, Appendix J, and NEI 94-01. An evaluation is performed to identify the cause of the unacceptable performance and appropriate corrective actions are taken to restore the leakage to an acceptable level. When excessive leakage results in corrective actions to repair a degraded condition, leak rate testing is performed after completion of repairs to confirm that the deficiency has been corrected.

The Appendix J Program monitors the results of the type A, B and C leak rate tests to demonstrate that the acceptance criteria for leakage have been satisfied. The test results that exceed the performance criteria are assessed as required by 10 CFR 50.72, "*Immediate Notification Requirements for Operating Nuclear Power Reactors*" and 10 CFR 50.73, "*Licensee Event Report System.*"

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.S4.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

## Operating Experience

To date the industry wide 10 CFR Part 50, Appendix J, leak rate testing programs have been effective in preventing unacceptable leakage through the containment pressure boundary Seabrook Station implementation of Option B for testing frequency is consistent with plant operating experience. The examples below demonstrate the effectiveness of the Seabrook Station leak rate testing program in identifying and correcting conditions that could lead to unacceptable containment pressure boundary leakage.

1. During Refueling Outage 9 (Fall of 2003) performance of the as-found local leak rate test for the Containment On-line Purge penetration the leakage rate was determined to have increased from the previous test in August 2001. Immediate evaluation of this condition determined that the inside containment isolation valve was responsible for the leakage. The actuator and quick exhauster diaphragm of this valve were replaced and the penetration retested. The leak rate remained unchanged. Subsequently, the inside containment isolation valve was replaced with a spare valve and the penetration retested. The as-left local leak rate test results were acceptable and comparable to those from the test performed in 2001.

2. During Refueling Outage 10 (Spring of 2005), fifty-three penetrations were tested and the results were compared to local leak rate test results performed during previous outages. Any changes in the results were evaluated. As a result of this comparative review, there were no degrading trends attributed to the containment isolation valves. However, during the performance of the local leak rate test for Combustible Gas Control Purge Exhaust penetration, the test results indicated an increase in leakage from Refueling Outage 7 (Fall of 2000). Both valves in the penetration indicated approximately the same leakage. It was determined that this can occur when a test boundary valve is leaking. The boundary valves were checked with liquid leak detector and one valve was found to have a packing leakage. Having identified the leakage path, the penetration was accepted. The leaking test boundary valve was repacked during Refueling Outage 12 (Spring of 2008).

3. During Refueling Outage 11 (Fall of 2006), twelve Type B penetrations and twenty-one Type C penetrations were tested. There were no major issues identified during the refueling outage but there was an increasing leakage trend noted on both of the Containment On-line Purge penetrations. A condition report was generated to evaluate this change observed during Refueling Outage 11. The penetration leakage rate was well within acceptable limits and the noted increase was determined to be acceptable.

4. During Refueling Outage 12 (Spring of 2008), seven Type B penetrations and twenty-eight Type C penetrations were tested. There were no major issues identified but one of the Containment On-line Purge penetrations still showed an increase in leak rate from Refueling Outage 11 (Fall of 2006). The leakage was evaluated and determined to be still acceptable and therefore, no corrective maintenance work was performed on the valves. The Containment On-Line Purge penetrations are tested each refueling outage. The results of these tests are trended by engineering to ensure that leakage rates will not exceed acceptance criteria or challenge the ability of the penetration valves to perform their intended function.

5. A 2008 condition report identified that the containment equipment hatch airlock barrel test and the test on the other testable penetrations are performed on an every other refueling outage frequency. This frequency requires the use of the allowable grace period to perform these during a refueling outage. To ensure that the grace period is not routinely used, these tests were rescheduled to be performed every refueling outage. This action is consistent with the Appendix J Program Owner's Group Technical Position Paper 2004-03 regarding the use of the 25% grace period.

These examples provide objective evidence that the Seabrook Station 10 CFR Part 50, Appendix J program is effective in preventing unacceptable leakage through the containment pressure boundary by monitoring, testing and evaluation of system and component conditions.

**Conclusion**

The Seabrook Station 10 CFR Part 50, Appendix J Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.31 STRUCTURES MONITORING PROGRAM

**Program Description**

The Seabrook Station Structures Monitoring Program is an existing program that will be enhanced to ensure provision of aging management for structures and structural components including bolting within the scope of this program. The Structures Monitoring Program is implemented through the Seabrook Station Maintenance Rule Program, which is based on the guidance provided in NRC Regulatory Guide 1.160, Revision 2, *"Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"* and NUMARC 93-01, Revision 2, *"Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"*. The Seabrook Station Structures Monitoring Program was developed using the guidance of these two documents to monitor the condition of structures and structural components within the scope of the Maintenance Rule, such that there is no loss of structure or structural component intended function.

The Seabrook Station Structures Monitoring Program includes periodic visual inspection of structures and structural components for the detection of aging effects specific for that structure. These inspections are completed by qualified individuals at a frequency determined by the characteristics of the environment in which the structure is found. A structure found in a harsh environment is defined as one that is in an area that is routinely subject to outside ambient conditions, very high temperature, high moisture or humidity, frequent large cycling of temperatures, frequent exposure to caustic materials, or extremely high radiation levels. For structures in these harsh environments, the inspection is conducted on a five year basis (plus or minus one year due to outage schedule and two inspections within ten years). Structures not found in areas qualifying as a harsh environment are classified as being in a mild environment, and are inspected on a ten year basis (plus or minus one year due to outage schedule and two inspections within twenty years).

Individuals conducting the inspection and reviewing the results are qualified per the Seabrook Station Structures Monitoring Program, which is in accordance with the requirements specified in ACI 349.3R-96, *"Evaluation of Existing Nuclear Safety related Concrete Structures"*. Individuals conducting the inspection and reviewing the results are to possess expertise in the design and inspection of steel, concrete and masonry structures. These individuals must either be a licensed Professional Engineer experienced in this area, or will work under the direction of a licensed Professional Engineer experienced in this area.

Detection of aggressive subsurface environments will be completed through the sampling of groundwater. This procedure monitors groundwater for chloride concentration, sulfate concentration and pH on a 5 year basis

Examination of inaccessible areas, such as buried concrete foundations, will be completed during inspections of opportunity or during focused inspections. An evaluation of these opportunistic or focused inspections for buried concrete will be performed under the Maintenance Rule Program every 5 years (if no opportunistic inspection was performed during a 5-year period, a focused 5 year inspection is required) to ensure that the condition of buried concrete foundations on site is characterized sufficiently to provide reasonable assurance that the foundations on site will perform their intended function through the period of extended operation. Additional inspections may be performed in the event that an opportunistic or focused inspection or visible portions of the concrete foundation reveal degradation and will be entered into the Corrective Action Program (CAP).

Concrete structures were constructed equivalent to recommendations in ACI 201.2R, *"Guide for Making a Condition Survey of Concrete in Service"*. Loss of material due to leaching of calcium hydroxide is considered to be an aging effect requiring management for Seabrook Station. There have been indications of leaching in below grade concrete in Seabrook Station structures. 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, well-cured concrete with low permeability. These structures are designed in accordance with ACI 318 and constructed in accordance with ACI 301 and ASTM standards. Nevertheless, Seabrook Station manages loss of material due to leaching of calcium hydroxide with visual inspection through the Structures Monitoring Program.

Seabrook Station has scheduled specific actions to determine the effects of aggressive chemical attack due to high chloride levels in the groundwater. Seabrook Station has scheduled concrete testing during the second and third quarter of 2010. An evaluation will be performed based on the results of the testing and a determination of the concrete condition which may lead to additional testing or increased inspection frequency. Testing of concrete may consist of the following:

a. concrete core samples

b. penetration resistance tests

c. petrographic analysis of the concrete core samples

d. visual inspection of rebar as they are exposed during the concrete coring

Seabrook will evaluate the results of the testing and, if required, undertake additional corrective actions in accordance with the Structures Monitoring Program CAP.

The Seabrook Station Structures Monitoring Program does not credit protective coatings for management of aging effects on structures and structural components within the scope of this program.

There are no preventative actions specified in the Seabrook Station Structures Monitoring Program, which includes implementation of NUREG-1801 XI.S5, XI.S6, and XI.S7. These are monitoring programs only.

The parameters monitored in the Seabrook Station Structures Monitoring Program are in agreement with ACI 349.3R-96 and ASCE 11-90, "*Structural Condition Assessment of Buildings*".

Concrete deficiencies are classified using the criteria specified in the Seabrook Station Structures Monitoring Program, which is based on the guidance provided in ACI 201.1R-2, "*Guide for Making a Condition Survey of Concrete in Service*".

As noted in the Seabrook Station response to NRC IN 98-26, *"Settlement Monitoring and Inspection of Plant Structures Affected by Degradation of Porous Concrete Subfoundations"*, porous concrete was not used in the construction of building sub-foundations at Seabrook.

Monitoring of structures and structural components in the scope of the Seabrook Station Structures Monitoring Program is performed in compliance with Regulatory Position 1.5 of NRC Regulatory Guide 1.160 The condition of all structures within the scope of this program is assessed on a periodic basis as specified by 10 CFR 50.65. Structures that do not meet their design basis at the time of inspection due to the extent of degradation, or that may not meet their design basis at the next normally scheduled inspection due to further degradation without intervention are entered into the Corrective Action Program and evaluated for corrective action and/or additional inspections as delineated in 10 CFR 50.65(a)(1). In addition, structures may also be scheduled for follow-up inspections following the completion of any corrective actions to that structure.

The condition of any structure subject to additional inspections or corrective actions is recorded through Seabrook Station Structures Monitoring Program reports to provide a basis for scheduling additional inspections and any required corrective actions in the future, as specified the Seabrook Station Structures Monitoring Program.

Structures that are determined to be acceptable under the Maintenance Rule structural inspections are monitored as specified in 10 CFR 50.65(a)(2).

Evaluations of a structure's condition assess the extent of any degradation of the structural member in accordance with industry standards and the judgment of the qualified individuals performing the inspections.

The acceptance guidelines in the Seabrook Station Structures Monitoring Program are a three-tier hierarchy similar to that described in ACI 349.3R-96, which provides quantitative degradation limits. Under this system, structures are evaluated as being acceptable, acceptable with deficiencies, or unacceptable. Evaluations of a structure's condition are completed according to the guidelines set forth in the Seabrook Station Structures Monitoring Program.

**XI.S5 – Masonry Wall Program**

The existing Seabrook Station Structures Monitoring Program also integrates the required elements of NUREG-1801 XI.S5 Masonry Wall Program.

There are no block or concrete masonry walls utilized in any category I structures at Seabrook Station. Therefore, requirements of NUREG-1801 XI.S5 including NRC IE Bulletin (IEB) 80-11, *"Masonry Wall Design"* and IN 87-67, *"Lessons Learned From Regional Inspections of License Actions in Response to IE Bulletin 80-11"* are not applicable at Seabrook Station. The masonry walls in the structures/buildings (Fire Pump House; Nonessential switchgear room; Turbine Building, and Yard Structure Station Blackout) performing non-(a)(1) functions are monitored under the Structures Monitoring Program portion of the Maintenance Rule Program, including the attributes as specified in XI.S5.

The primary parameter monitored for masonry walls is cracking of the walls.

Monitoring for cracking and degradation of masonry walls at Seabrook Station is completed through periodic visual inspections conducted under the Seabrook Station Structures Monitoring Program.

Acceptance criteria for non-safety-related masonry walls within the scope of this program are based upon the extent of cracking or other wall degradation. The ability of the masonry wall to perform its intended function is evaluated in the Structures Monitoring Program.

**XI.S7 – Inspection of Water Control Structures Associated with Nuclear Power Plants**

The existing Seabrook Station Structures Monitoring Program also integrates the required elements of NUREG-1801 XI.S7 and RG 1.127, *"Inspection of Water Control Structures Associated with Nuclear Power Plants"*. The recommendations of Regulatory Guide 1.127 are met by implementing an appropriate inservice inspection and surveillance program for the flood protective structures and the Water Control Structures.

Flood protective structures (stone revetments, reinforced concrete vertical seawall and the sheet pile retaining wall) are within the scope of this program and are inspected under the Maintenance Rule Program.

Water Control Structures (includes Service Water Cooling, Service Water Pump House and Circulating Water Pump House Building (below elevation

21'-0), and Intake & Discharge Transition Structures) are inspected under the Maintenance Rule Program.

Periodic visual inspections of Water Control Structures and Flood Protective Structures are conducted by the Seabrook Station Structures Monitoring Program. The visual inspections will detect cracking; movement due to settlement, heaving or deflection; conditions at junctions with abutments and embankments, erosion, cavitation, seepage, and leakage.

Monitoring for aging effects on flood protective structures and Water Control Structures is completed through periodic visual inspections conducted under the Seabrook Station Structures Monitoring Program.

The acceptance of the flood protective structures and Water Supply Structures condition is evaluated according to the criteria set forth in the Seabrook Station Structures Monitoring Program. This criterion includes the evaluation of concrete deficiencies in accordance with ACI 201.1R-2 and as described above in the Structures Monitoring Program, which is in accordance with ACI 349.3R-96.

There are no earthen structures at Seabrook Station within the scope of this program.

**NUREG-1801 Consistency**

This existing program is consistent with NUREG-1801, Chapter XI, Programs XI.S5, XI.S6, and XI.S7.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancements will be made prior to entering the period of extended operation.

1. Enhance procedure to add the aging effects, additional locations, inspection frequency and ultrasonic test as follows:

    a. Elastomers: Loss of sealing, leakage, deterioration for CEVA seals only, Aluminum: Cracking, Non-Metallic Fire Proofing: Abrasion and Flaking, Lubrite: Corrosion, Distortion, Dirt etc.

Additional Locations

a. Overhead and Fuel Handling Cranes and NUREG-0612 Cranes, All supports, Tanks 1-FP-TK-35-A, 1-FP-TK-35-B, 1-FP-TK-36-A, 1-FP-TK-36-B, 1-FP-TK-29, and their supports and foundations, Fire Pump House Boiler Building,

b. Safety-Related and Non-Safety-Related Electrical Cable Manhole, Duct Bank Yard Structures

c. A below grade inspection for buried concrete is required at least once every 5 years and it may either opportunistic or focused inspection due do aggressive groundwater inleakage.

Ultrasonic Testing Inspections

a. Perform Ultrasonic testing inspections and evaluation of the internal bottom surface of the two Fire Protection Water Storage Tanks within 5 years prior to the period of extended operation (in support of Aboveground Steel Tanks program).

*Program Elements Affected: Scope of Program (Element 1),* Parameters Monitored or Inspected (Element 3), Monitoring and Trending (Element 5).

2. Enhance procedure to include inspection of opportunity when planning excavation work that would expose inaccessible concrete.

   a. Procedure SH 6.4, "Dig Safe" will be enhanced to include inspection of opportunity when planning excavation work that would expose inaccessible concrete.

   *Program Elements Affected: Element 1 (Scope of Program), Element 3 (Parameters Monitored/Inspected).*

**Operating Experience**

1. *XI.S6 – Structures Monitoring Program*

   Structures Monitoring Program is an existing program that was established in 1996 in response to the Regulatory requirements resulting from the establishment of the Maintenance Rule, 10 CFR 50.65. Upon completion of this program, a baseline inspection was completed for both online and outage-accessible structures found in mild and harsh environments at Seabrook Station.

   According to the Periodic Assessments of the Maintenance Rule Program *"October 2004 through March 2006"*, and *"April 2006 through March*

*2008"*, the baseline inspection was performed at Seabrook Station from November 1995 to June 1996 of online-accessible structures included concrete, masonry, and steel structures used to house plant equipment. This included the walls, floors, and roofs of buildings, along with the pedestals of mechanical and electrical equipment. The second series of inspections was conducted in 2001, with the most current inspections completed in 2006. The results of this inspection are identified under various preventive maintenance work orders for structures monitoring walkdowns.

The Seabrook Station Maintenance Rule Program is reviewed by an Expert Panel at least once every refueling cycle, not to exceed an interval of 24 months between assessments, in accordance with 10 CFR 50.65(a)(3).

Ground water testing performed in November 2008 and September 2009 found pH values between 6.01 and 7.51, chloride values between 19 ppm and 3900 ppm, and sulfate values between 10 ppm and 100 ppm. Aggressive chemical attack becomes a concern when environmental conditions exceed threshold values (Chlorides > 500 ppm, Sulfates >1500 ppm, or pH < 5.5). Seabrook Station is not located in areas exposed to sulfate attack, nor is it located near industrial plants whose emissions could alter environmental parameters, but is exposed to chloride attack.

Seabrook Station has experienced groundwater infiltration through cracks, capillaries, pore spaces, seismic isolation joints and construction joints in the concrete walls and floor slabs of below-grade concrete structures.

Various materials and methods have been utilized for remediation of groundwater infiltration. For example, Seabrook Station installed below-grade dewatering wells that are included in the Dewatering System. These wells were designed to pump down groundwater levels to reduce the static hydraulic head on the outside concrete. The wells were successful at reducing inleakage. Other remediation methods include the application of waterproofing materials, such as:

a. Vandex waterproofing

b. De Neef Chemical Corporation's Denepox I-40, Denepox I-300, and Denepox Rapid Gel epoxy.

c. De Neef Chemical Corporation's polyurethane resin systems including: Hydro Active Cut, Hydro Active Flex, Hydro Active Flex SLV, and Hydro Active Sealfoam.

d. Aqua-Tech W.P. for waterproofing exterior concrete surfaces above grade.

Currently, three cementitious products manufactured by Xypex Chemical Corporation are approved and authorized for use at Seabrook Station. The Xypex products are: Xypex Concentrate, Xypex Modified and Xypex Patch'N Plug. The Xypex products have had limited success in reducing groundwater infiltration.

An Engineering Evaluation was performed in 1987 for "Corrosion of Concrete Reinforcing Steel Due to Groundwater Inleakage". This evaluation was performed to evaluate the long term corrosion effects on reinforcing steel due to cracking of concrete walls and inleakage of groundwater. The Engineering Evaluation concluded that inleakage through cracks in concrete walls at Seabrook Station would not lead to corrosion of reinforcing steel on either a short-term or a long-term basis. The Engineering Evaluation found that the environment of the concrete remains non-corrosive due to the low chloride concentration in the concrete placed during construction and the low concentration of chlorides in the groundwater on site. In addition, the Engineering Evaluation concluded that site conditions are such that, even if the passive environment of the concrete had been impaired, corrosion of reinforcing steel would not occur due to the lack of necessary oxygen to complete the electro-chemical process.

The groundwater testing discussed above indicated chloride levels in the aggressive range for concrete. As a result, additional testing, as described below, is planned to validate the existing Engineering Evaluation. Specifically, Seabrook Station will perform concrete testing during the second quarter of 2010, in areas that have experienced groundwater infiltration, to determine the effects of aggressive chemical attack on structural concrete. The concrete testing will consist of the following:

a. Concrete core samples

b. Penetration resistance testing

c. Petrographic analysis on concrete core samples

d. Visual inspection of rebar as they are exposed during the concrete coring

The testing is intended to demonstrate whether the concrete and reinforcing steel show any indication of degradation that could, in time, lead to a loss of capacity of the concrete to perform its intended functions. Based on the results of the testing, the Seabrook Station Structures Monitoring Program will evaluate the need for additional measures, such as additional testing of the type listed above, the imposition of a five-year frequency of the type of testing listed above, or other remedial actions. Other actions may include the implementation of an aggressive crack and void sealing program and/or water sealing of entire walls. All actions will be based on the guidelines of ACI-349.3R-96 and evaluated by a registered professional engineer, knowledgeable in the design, evaluation, and in-service inspection of concrete structures and performance requirements of nuclear safety-related structures.

The actions discussed above provide reasonable assurance that the Structures Monitoring Program will manage the aging effects of concrete and reinforcing steel such that they will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation

2. *XI.S5 – Masonry Wall Program*

   There are no block or concrete masonry walls utilized in any category I structures at Seabrook Station. The masonry walls in the structures/buildings (Fire Pump House; Nonessential switchgear room; Turbine Building, and Yard Structure SBO) performing non-(a)(1) functions are monitored as required. NRC IE Bulletin 80-11 and NRC IN 87-67 require actions to *"to ensure that the evaluation basis for a wall is not invalidated through a physical plant change or system reclassification"*. The design control process assures no changes are made to the status of any block walls. Visual inspection of non-safety related masonry includes examination for any cracking and degradation of masonry walls within the areas listed in the Monitoring Checklist of Structures Monitoring Program portion of the Maintenance Rule Program.

3. *XI.S7 – Inspection of Water Control Structures Associated with Nuclear Power Plants*

   During refueling Outage 11 (Fall of 2006), divers discovered concrete debris in the Circulating Water Pump House Forebay during the scheduled inspection. Site engineering evaluated the impact on the Circulation Water structure and concluded that there is no effect on the structural integrity of the Circulating Water Pump House. During this inspection no other concrete degradation was reported.

   The condition of the Water Control Structures and Flood Protection Structures at Seabrook Station has been assessed through visual inspection conducted through Structure Monitoring Program procedures in the Maintenance Rule Program.

The Maintenance Rule inspections began in 1996 with the initial baseline inspection, as described previously.

The deficiencies that were recorded during these baseline inspections were evaluated by Seabrook Station personnel in accordance with the Maintenance Rule Program. The evaluation included a review of the deficiency report and initiation of any necessary corrective actions.

A review of both operating experience at Seabrook Station and the results of the Maintenance Rule Program inspections did not indicate any adverse degradation of Water Control Structures or Flood Protection Structures.

**Conclusion**

The Seabrook Station Aging Management Program for Structures Monitoring provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operations.

## B.2.1.32 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS

**Program Description**

The Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program is a new program that will manage the aging effects of embrittlement, cracking, discoloration or surface contamination leading to reduced insulation resistance or electrical failure of accessible cables and connections due to exposure to an adverse localized environment caused by heat, radiation or moisture in the presence of oxygen. This program applies to accessible cables and connections installed in in-scope structures.

The program consists of a visual inspection of the cable or connections exterior surfaces. Accessible electrical cables and connections exposed to adverse localized environments or ambient conditions in excess of 60-year service limiting environments will be visually inspected for signs of accelerated age related degradation.

Connections included in the scope of this program are splices, terminal blocks, connectors, and the insulation portion of fuse blocks.

This program considers the technical information and guidance provided in the following:

a. NUREG/CR-5643, *"Insights Gained From Aging Research"*

b. IEEE Std. P1205, *"IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations"*

c. SAND96-0344, *"Aging Management Guidelines for Commercial Nuclear Power Plants – Electrical Cable and Terminations"*

d. EPRI TR-109619, *"Guideline for the Management of Adverse Localized Equipment Environments"*

An Adverse Localized Environment is a condition in a limited plant area that is significantly more severe than the specified service environment (i.e. temperature, radiation, or moisture) for the cable or connections. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability.

Seabrook Station has determined that Polyvinyl Chloride has the limiting temperature and radiation thresholds of all cable and connection material. The ambient temperature and radiation threshold values for Polyvinyl Chloride are 112°F (44.2°C) and $2 \times 10^7$ Rads. Seabrook Station has chosen these limiting temperature and radiation values to define the threshold for adverse localized environments.

The scope of the Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program includes insulated cable and connections identified in walkdowns that will be performed utilizing the guidance provided in EPRI TR-109619. The inspection area includes all buildings or structures that are in-scope for License Renewal. The walkdown of the in-scope buildings or structures will include a temperature measuring method such as thermography to assist in identifying adverse localized environments. Accessible insulated cable and connections within the in-scope buildings or structures which contain temperatures or radiation values that are equal to or exceed 112°F (44.2°C) or $2 \times 10^7$ Rads identified during the walkdown will be inspected for surface anomalies.

The non-EQ insulated cables and connections managed by this program will include power, instrumentation, control, and communication applications located in accessible adverse localized environments throughout the plant. Accessible is defined as those cables and connections that can be viewed from ground level and without opening electrical enclosures.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is an inspection program and contains no actions to prevent or mitigate aging degradation.

This program will visually inspect accessible electrical cables and connections installed in adverse localized environments at least once every 10 years. The first inspection for license renewal is to be completed before the period of extended operation.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program requires that accessible insulated cables and connections are free from unacceptable, visual indications of cable and connection surface anomalies, such as embrittlement, discoloration, cracking, or surface contamination, which suggest that conductor insulation or connection insulation degradation exists. The Seabrook Station program defines an unacceptable indication as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program requires that all unacceptable visual indications of cable and connection surface anomalies be entered into the Corrective Action Program and be subjected to an engineering evaluation. The engineering evaluation will consider the age and operating environment of the component, as well as the severity of the anomaly and whether such an anomaly has previously been correlated to degradation of conductor insulation or connections. If cable and connection surface anomalies are detected, the corrective actions may include, but are not limited to, testing, shielding or otherwise changing the environment, or relocation or replacement of the affected cable or connection. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.E1.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Plant-specific and industry wide operating experience was considered in the development of this program. The review of plant-specific and industry-wide operating experience ensures that the corresponding NUREG-1801, Chapter XI.E1 Program will be an effective Aging Management Program for the period of extended operation.

1. This program considered the operating experience in NUREG-1801, Chapter XI, Section E1. Industry operating experience noted in NUREG-1801 has shown that adverse localized environments caused by heat, radiation or moisture for electrical cables and connections have been shown to exist and have been found to produce degradation of insulating materials that is visually observable. These visual indications, such as color changes or surface cracking, can be used as indicators of degradation. This demonstrates that Seabrook Station considered industry operating experience in the preparation of this program.

2 As nuclear plants approach or enter the period of extended operation, several utilities have begun to implement their Electrical Aging Management Programs. Seabrook Station License Renewal personnel are actively involved in industry groups such as the NEI License Renewal Electrical Working Group. Experience of other utilities is shared in this working group. As an example, several plants presented the results of their GALL Section XI.E1 programs at the 2009 and 2010 License Renewal Electrical Working Group meeting. This operating experience demonstrates that Seabrook Station actively participates in industry activities relative to the subject program.

3. Plant operating experience has shown that the Corrective Action Program has addressed issues of cable degradation in recent years. Cables have been identified with degraded insulation, primarily as a result of exposure to excessive localized overheating. For example, insulated cable associated with heater terminations was found to be degraded. Also, power cords to various non-essential instruments and wiring internal to lighting fixtures were found to be cracked. Seabrook Station is continuing to monitor for degraded connections and is systematically replacing the power cords and internal lighting wire. This operating experience demonstrates that plant-specific operating experience was used in the development of this program.

4. Seabrook Station Plant Engineering Guidelines for system walkdowns includes steps that prompt engineers to observe the condition of cable and connections. This operating experience demonstrates that Seabrook Station monitors the condition of accessible cables and connections.

The above operating experience demonstrates Seabrook Station's consideration of industry operating experience, its involvement in industry activities directly related to the subject program, and that plant processes are being implemented to manage the aging of accessible cables and connections.

**Conclusion**

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### B.2.1.33 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS

**Program Description**

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a new program that will manage the aging effects of reduced insulation resistance due to exposure to adverse localized environments caused by heat, radiation, or moisture in the presence of oxygen, causing increased leakage currents. This program applies to sensitive instrumentation cable and connection circuits with low-level signals in the in-scope portions of in-core neutron flux monitoring cable in the Nuclear Instrumentation System. These cables are not included in the Seabrook Station EQ Program.

This program considers the technical information and guidance provided in the following:

a. NUREG/CR-5643, *"Insights Gained From Aging Research"*

b. IEEE Std. P1205, *"IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations"*

c. SAND96-0344, *"Aging Management Guidelines for Commercial Nuclear Power Plants – Electrical Cable and Terminations"*

d. EPRI TR-109619, *"Guideline for the Management of Adverse Localized Equipment Environments"*

These in-scope in-core neutron flux monitoring cables and connections in the Nuclear Instrumentation System are routed inside containment and potentially exposed to moisture, radiation, and high temperatures. The high voltage low-level signal instrumentation circuits from the Radiation Monitoring System are not included in this program. These cables are included in the Seabrook Station EQ program.

The Seabrook Station program contains no actions to prevent or mitigate aging degradation.

The Seabrook Station program specifies insulation resistance tests to be performed on in-scope cable to determine the cable insulation condition. Insulation resistance testing of cable systems is a proven method for detecting deterioration of the insulation system. The frequency of the tests on these cables will be based on engineering evaluation, but the test frequency will be at least once every ten years. The first test will be completed before entering the period of extended operation. The program will develop acceptable insulation resistance values. The acceptable insulation resistance values for the test will provide reasonable assurance that the cable will perform its intended function.

Unacceptable test results are entered into the Corrective Action Program. The program's corrective actions component requires that an engineering evaluation to be performed when the test acceptance criteria are not met in order to ensure that the intended functions of the electrical cable system can be maintained consistent with the current licensing basis. The evaluation considers the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and likelihood of recurrence.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.E2.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Plant-specific and industry wide operating experience was considered in the development of this program. The review of plant-specific and industry-wide operating experience ensures that the corresponding NUREG-1801, Chapter XI.E2 Program will be an effective Aging Management Program for the period of extended operation.

1. Industry operating experience that forms the basis for this program is included in the operating experience element of the corresponding NUREG-1801, Chapter XI Program. Industry operating experience noted in NUREG-1801 has shown that exposure of electrical cables to adverse localized environments caused by heat, radiation, or moisture can result in

reduced insulation resistance. Reduced insulation resistance causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in insulation resistance is a concern for circuits with sensitive high voltage, low-level signals such as nuclear instrumentation circuits since it may contribute to signal inaccuracies. Insulation resistance testing is an acceptable method for determining the cables condition. Seabrook Station considered this industry operating experience in the development of the program.

2. Plant specific operating experience was reviewed. Insulation resistance tests have been performed on the in-scope sensitive instrumentation cable and connections.

   In 2008, testing was performed on all in-core neutron flux monitoring cables and connections. The test results documented a less than expected insulation resistance reading between the inner and outer shield. The low insulation resistance reading was attributed to the connector design. The design issue was resolved and retesting found the cable and connection to be acceptable. Although this example is not representative of age related degradation, it does demonstrate that the test method is an acceptable for identifying degraded conditions.

The above plant specific operating experience shows that insulation resistance testing is an effective method to determine the acceptability of cables and connections used in nuclear instrumentation circuits

### Conclusion

The Seabrook Station Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.34 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS

### Program Description

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program that will manage the aging effects of localized damage and breakdown of insulation leading to electrical failure of inaccessible medium voltage cables due to adverse localized environments caused by exposure to significant moisture and voltage.

Seabrook Station defines an adverse localized environment for medium-voltage cables as exposure to moisture for more than a few days while energized at the system voltage for more than 25 percent of the time.

The Seabrook Station program includes periodic inspections of manholes containing in-scope medium voltage cables. The inspection focuses on water collection in cable manholes, and draining water, as needed. The frequency of manhole inspections for accumulated water and subsequent pumping will be based on inspection results. The objective of the inspections is to keep the cables from becoming submerged thereby minimizing their exposure to significant moisture. To meet this objective, adjustments in inspection frequency may be required. The maximum time between inspections will be no more than two years. The first inspections will be completed prior to entering the period of extended operation.

In addition to periodic manhole inspections, in-scope, medium-voltage cables exposed to significant moisture and energized at the system voltage for more that 25 percent of the time are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, "*Effects of Moisture on the Life of Power Plant Cables*" or other testing that is state-of-the-art at the time the test is performed. Cable testing will be performed prior to entering the period of extended operation and at least every 10 years thereafter.

Development of this program considers the technical information and guidance provided in the following:

a. NUREG/CR-5643, *"Insights Gained From Aging Research"*

b. IEEE Std. P1205, *"IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations"*

c. SAND96-0344, *"Aging Management Guidelines for Commercial Nuclear Power Plants – Electrical Cable and Terminations"*

d. EPRI TR-109619, *"Guideline for the Management of Adverse Localized Equipment Environments"*

Seabrook Station defines significant moisture as periodic exposures to moisture that last more than a few days (e.g., cable in standing water). Seabrook Station considers periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) as not being significant. Significant voltage exposure is defined as being subjected to system voltage for more than twenty-five percent of the time.

The Seabrook Station program includes periodic actions taken to prevent cables from being exposed to significant moisture, such as inspecting for water collection (and draining if needed) in manholes that contain in-scope inaccessible medium-voltage cables.

The Seabrook Station program acceptance criteria for the electrical cable test is defined by the specific type of test performed and the specific cable tested. If water is found in manholes, the water will be drained and the inspection frequency will be increased.

Unacceptable tests or inspections will be entered into the Corrective Action Program. The corrective action will include an engineering evaluation when the cable testing test acceptance criteria are not met to determine the acceptability of the cable to perform its intended function consistent with the current licensing basis. The evaluation will also consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and the likelihood of recurrence. The corrective action process will include a determination as to whether the same condition or situation is applicable to other inaccessible, in-scope, medium-voltage cables.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.E3.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Plant-specific and industry wide operating experience was considered in the development of this program. The review of plant-specific and industry-wide operating experience ensures that the corresponding NUREG-1801, Chapter XI.E3 Program will be an effective Aging Management Program for the period of extended operation.

1. The Seabrook Station program considered NUREG-1801 as part of the operating experience review. NUREG-1801 compiled the industry operating experience for inaccessible medium voltage cables. This information is current through the September 2005 issue date of the NUREG. This demonstrates that Seabrook Station considered industry operating experience in the formation of this aging management program.

2. Seabrook Station reviewed NRC Generic Letter 2007-01, "*Inaccessible or Underground Power Cable Failures That Disable Accident Mitigation Systems or Cause Plant Transients*". The Generic Letter informed licensees of failure of certain power cables can affect the functionality of multiple accident mitigation systems or cause plant transients. As part of the Generic Letter, the NRC provided examples of medium voltage cable failures at other utilities. In response to Generic Letter 2007-01, Seabrook Station described periodic testing of representative low voltage cables, testing of medium voltage cables and periodic inspections of manholes. The response concluded that no failures have occurred in power cables in the scope of Maintenance Rule. This operating experience demonstrates Seabrook's involvement in regulatory activities relative to inaccessible cables.

3. Seabrook Station performed reviews of plant specific operating experience. The review focused on test data of in-scope cables and manhole inspections.

   In 1994, a commitment was made to inspect 10 percent of the safety related manholes every five years. This commitment is reiterated in the Seabrook Station response to Generic Letter 2007-01.

   In 2009, a fleet procedure was issued which provided a dewatering strategy for electrical cables. The strategy is that all medium voltage cables important to generation and nuclear safety are to be maintained in a dry (not submerged) condition. The fleet procedure states that the inspection frequency should be based on operating experience that has demonstrated successful methods for keeping the cable dry. Seabrook Station has issued guidelines to implement the fleet procedure and has begun the process of complying with the fleet procedure.

   Seabrook Station performed inspections in late 2009 and early 2010 of all safety related manholes containing medium voltage cables. Water was removed from the manholes. The inspection frequency was increased as required to prevent the medium voltage cables from becoming submerged.

   Results of tests performed in 2008 were reviewed. All in-scope cables met the acceptance criteria of the test performed.

This operating experience demonstrates that Seabrook Station is proactively managing the water levels in manholes containing safety related medium voltages cables and testing medium voltage cables.

The above operating experience demonstrates that Seabrook Station considered industry operating experience while preparing this program, and participated in regulatory activities related to inaccessible medium voltage cables. The operating experience also demonstrates that Seabrook Station is proactive in managing the aging of inaccessible medium voltage safety related cables.

**Conclusion**

The Seabrook Station Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.35 METAL ENCLOSED BUS

**Program Description**

The Metal Enclosed Bus (MEB) Program is a new program that will manage the following aging effects of in-scope metal enclosed busses:

a. Loosening of bolted connections due to thermal cycling and ohmic heating

b. Hardening and loss of strength due to elastomer degradation

c. Loss of material due to general corrosion

d. Embrittlement, cracking, melting, swelling, or discoloration due to overheating or aging degradation

This new program will be implemented prior to entering the period of extended operation and at least once every 10 years thereafter.

The in-scope MEB's are the non-safety related 4160V non-segregated bus ducts 1-ED-BD-1 thru -4, the non-safety related 13800V non-segregated bus ducts 1-ED-BD-5 thru -8, the safety related 4160V non-segregated bus ducts 1-EDE-BD-9 thru -12 and the non-safety related 25kV isolated phase bus duct 1-ED-BD-13.

Aging management of the exterior housing and elastomers of the in-scope MEB's is included in the Structures Monitoring Program. The inspection frequency and acceptance criteria for these components will be defined by the Structures Monitoring Program.

The internal portions of the in-scope metal enclosed bus enclosures will be visually inspected for aging degradation of insulating material and for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion. The bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The isolated phase bus conductor is not insulated. The internal bus supports will be visually inspected for structural integrity and signs of cracks. The accessible bus sections will be inspected for loose connections using thermography from outside the metal enclosed bus while the bus is energized.

The program requires that bolted connections be below the maximum allowed temperature for the application, and free of unacceptable visual defects.

Unacceptable thermography heat signatures or unacceptable visual indications are entered into the Corrective Action Program.

The corrective action process will perform further investigations and evaluations when inspections do not meet the acceptance criteria. Corrective actions applied to inspections that do not meet the acceptance criteria may include but are not limited to cleaning, drying, increased inspection frequency, replacement, or repair of the affected MEB components. If an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible MEBs.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.E4.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Plant-specific and industry wide operating experience was considered in the development of this program. The review of plant-specific and industry-wide operating experience ensures that the corresponding NUREG-1801, Chapter XI.E4 Program will be an effective Aging Management Program for the period of extended operation.

1. Industry operating experience that forms the basis for the Seabrook Station Program is included in the operating experience element of the corresponding NUREG-1801, Chapter XI Program. Industry operating experience noted in NUREG-1801 has shown that failures have occurred on MEBs caused by cracked insulation and moisture or debris buildup internal to the MEB. Industry operating experience noted in NUREG-1801 has also shown that MEB exposed to appreciable ohmic or ambient heating during operation may experience loosening of bolted connections related to the repeated cycling of connected loads or of the ambient temperature environment. This condition can occur in heavily loaded circuits (i.e., those exposed to appreciable ohmic heating or ambient heating) that are routinely cycled. This operating experience demonstrates that Seabrook Station considered industry operating experience in the development of this program.

2. INPO issued a Special Event Report, SER 5-09, "6.9-kV Nonsegregated Bus Failure and Complicated Scram." The event report documents the catastrophic failure of a 6.9-kV nonsegregated bus. The cause of the event was attributed to the overheating of the center bus bar at the flex connection. This operating experience demonstrates that Seabrook Station considered industry operating experience in the development of this program.

3. Seabrook Station performs periodic visual inspections and infrared thermography tests on all in-scope non-segregated bus and the isolated phase bus.

   In 2005, during the inspection of a non- segregated phase bus, white corrosion was found on a bolted connection surface near a flat washer. In addition, a green residue was noted on the surface area of the bus near the connection area. The connection was broken to facilitate a complete inspection of the connection for additional corrosion. The connection was remade and successfully tested. The same duct was noted that an expansion joint was not sealing. The deficiency was corrected. This operating experience demonstrates the effectiveness of visual inspections.

   In 2007, a condition report was issued to document the results of thermography inspections performed after power up-rate modifications. The Condition Report reports increasing temperature on the generator neutral bus conductor. The root cause analysis prescribed several remedies including design changes for the connections, increased ventilation and air flow. Subsequent thermography inspections found the bus temperatures to be within the design limits. Although this operating experience is not age related it demonstrates the effectiveness of thermography inspections.

In 2008, a condition report was issued to document an inspection of the Isolated Phase Bus. During the inspection, a black substance on the outside of the duct was discovered. The substance was determined to be elastomer deterioration. Work orders were initiated to inspect bus elastomers and replace as necessary. This operating experience demonstrates the effectiveness of visual inspections.

A review of recent inspection data for the other in-scope metal enclosed bus did not reveal any other anomalies.

4. Recent MEB thermography inspection results were reviewed. No anomalies were identified. This operating experience demonstrates that Seabrook Station is proactive by inspecting for loose connections.

The above operating experience demonstrates that Seabrook Station considered industry and plant specific operating experience in the development of this program, that visual inspection is an acceptable method in finding anomalies and that the plant is proactive by inspecting for loose connections.

### Conclusion

The Seabrook Station Metal Enclosed Bus Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.36 FUSE HOLDERS

### Program Description

The Fuse Holders Program is a new program that will manage the aging effects of thermal fatigue in the form of high resistance due to corrosion or oxidation of in-scope metallic clamps of fuse holders.

The Seabrook Station program, *"Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"*, will manage the aging of insulating material but not the metallic clamps of the fuse holders.

The program will perform tests on the in-scope fuse holders (metallic clamps). The test will be a proven test such as thermography, or contact resistance which detects thermal fatigue in the form of high resistance caused by corrosion or oxidation. The type test performed will be determined prior to the initial test. The first test will be completed prior to entering the period of extended operation and at least once every 10 years thereafter.

The program applies to the metallic portions of fuse holders located outside of active devices and considered susceptible to aging effects. Fuse holders inside an active device (e.g., switchgears, power supplies, power inverters, battery chargers, and circuit boards) are not within the scope of this program. The fuse holders in the scope of this program have been identified and are located in 19 fuse panels.

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

This program inspects for high resistance due to corrosion and oxidation on the metallic clamp portion of the fuse holder.

The Seabrook Station analysis shows that the aging effects due to thermal fatigue in the form of high resistance caused by ohmic heating, thermal cycling or electrical transients, mechanical fatigue caused by frequent removal/replacement of the fuse or vibration do not require aging management.

The program will define the acceptance criteria for each specific type of test and inspection performed and the specific type of fuse holder tested. If thermography is used, a heat signature should not indicate abnormal temperatures for the application. If contact resistance test is used, the resistance value should be appropriate for the application.

Unacceptable inspection or test results will be entered into the Corrective Action Program. An engineering evaluation will be performed to ensure that the intended functions of the fuse holders can be maintained consistent with the current licensing basis. The evaluation considers the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective action necessary, and the likelihood of recurrence.

**NUREG-1801 Consistency**

The program is consistent with NUREG-1801 XI.E5.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Plant-specific and industry wide operating experience was considered in the development of this program. The review of plant-specific and industry-wide operating experience ensures that the corresponding NUREG-1801, Chapter XI.E5 Program will be an effective Aging Management Program for the period of extended operation.

1. Industry operating experience that forms the basis for this program is included in the operating experience element of the corresponding NUREG-1801, Chapter XI Program. NUREG-1801 notes that loosening of fuse holders and corrosion of fuse clips are aging mechanisms that, if left unmanaged, can lead to a loss of electrical continuity function. Also, as stated in NUREG-1760, "*Aging Assessment of Safety-Related Fuses Used in Low and Medium Voltage Applications in Nuclear Power Plants*", fuse holders experience a number of age-related failures. The major concern is that failures of a deteriorated fuse holder might be induced during accident conditions since they are not subject to the environmental qualification requirements of 10 CFR 50.49. This operating experience demonstrates that Seabrook Station considered industry operating experience in the development of this program.
2. Seabrook Station routinely performs infrared thermography tests on numerous pieces of equipment including electrical connections as part of the preventive maintenance program.

   In 2008, during a routine infrared thermography test, a fuse holder was found to be 50° F hotter than similar fuses in the cabinet. The problem was diagnosed as a defective fuse holder. The documentation identified this anomaly as defective equipment and not age related. This operating experience demonstrates the effectiveness of the thermography test.
3. Seabrook Station has performed thermography or resistance tests on fuses located in the in-scope fuse panels. All recent tests were reviewed and found to be satisfactory. This operating experience demonstrates that Seabrook Station is proactive in efforts to detect heat due to increased resistance of fuse holders.
4. In 2009, members of the Seabrook Station License Renewal team performed a walkdown of the in-scope Train "B" fuse cabinets. The walkdown was performed to assess the current condition of the in-scope fuse panels. The walkdown results concluded that fuse blocks showed no signs of excessive heating, discoloration, corrosion, degradation or looseness. However, a Condition Report was written to document the presence of a residue on the fuse cabinet mounting bolts. The evaluation of

the anomaly concluded that residue on the bolts had no affect on the fuse holders. This operating experience confirms that the current condition of the in-scope fuse holders viewed during the walkdown is free of corrosion and oxidation.

### Conclusion

The Seabrook Station Fuse Holders Program provides reasonable assurance that the aging effects will be adequately managed such that components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.1.37 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS

### Program Description

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new, one-time testing program that will be used to verify that the aging effect of loosened bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination corrosion and oxidation on the metallic portion of electrical cable connections does not require management.

A representative sample of cable connections within the scope of license renewal will be selected for one-time testing prior to the period of extended operation. The scope of this sampling program will consider application (medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc). The technical basis for the sample selection will be documented.

The specific type of test performed will be a proven test for detecting loose connections, such as thermography or contact resistance measurement, as appropriate for the application.

The program scope includes external connections terminating at active or passive devices. Wiring connections internal to an active assembly are considered a part of the active assembly and therefore not within the scope of the program. The program does not include high-voltage (>35 kV) switchyard connections. Seabrook Station cable connections covered under the EQ program are not included in the scope of this program.

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

The Seabrook Station program will perform tests on a representative sample of electrical connections within the scope of license renewal at least once prior to the period of extended operation to confirm that there are no aging effects requiring management during the period of extended operation. The testing methods will include either thermography or contact resistance testing. The test will not remove connection insulation such as heat shrink tape, sleeving, insulating boots, etc. The program is a one-time inspection which provides additional confirmation to support industry operating experience that shows electrical connections have not experienced a high degree of failures, and that existing installation and maintenance practices are effective.

The acceptance criteria for bolted connections will meet the criteria as defined for the specific type of test performed and the specific type of cable connections tested. If thermography is used, a heat signature should not indicate abnormal temperatures for the application. If contact resistance test is used, the resistance value should be appropriate for the application.

If test acceptance criteria are not met, the results will be entered into the Corrective Action Program. The corrective action process will be used to perform an evaluation that will consider the extent of the condition, the indications of aging effect, and changes to the one-time inspection program. Corrective actions may include, but are not limited to sample expansion, increased inspection frequency, and replacement or repair of the affected cable connection components.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 XI.E6 as modified by LR-ISG-2007-02.

**Exceptions to NUREG-1801**

None

**Enhancements**

None

**Operating Experience**

Plant-specific and industry wide operating experience was considered in the development of this program. The review of plant-specific and industry-wide operating experience ensures that the one-time inspection corresponding to NUREG-1801, Chapter XI.E6 Program will confirm the absence or presence of age-related degradation of cable connections caused thermal cycling, ohmic heating, corrosion and oxidation.

1. Seabrook Station routinely performs infrared thermography tests on numerous pieces of equipment, including electrical connections, as part of the preventive maintenance program.

   In 2002, during an infrared thermography inspection of a 480 volt circuit breaker, a hot connection was found. The connection was approximately 150° F hotter than similar connections. Seabrook Station procedures required that the connection be corrected within one week. Infrared thermography was used to monitor the connection on a daily basis until corrective action could be taken to correct the hot connection. This operating experience shows that the maintenance practices at Seabrook Station are effective in identifying electrical connection anomalies prior to loss of intended function.

   In 2005, an infrared thermography inspection identified heating on three connections in a control panel. The connections were 30°F to 50°F higher than expected. Seabrook Station procedures require that this condition be corrected in 12 weeks. The hot connections were repaired. The connections were found to be tight and the hot spot was attributed to defective connectors. This operating experience shows that the maintenance practices at Seabrook Station are effective in identifying electrical connection anomalies prior to loss of intended function.

The above examples of operating experience provides evidence that the testing method utilized by Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements will be effective in finding loose electrical connections prior to failure. Additionally it shows that Seabrook Station's experience is in alignment with the industry in that electrical connections have not experienced a high degree of failures.

**Conclusion**

The Seabrook Station Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.2 PLANT SPECIFIC AGING MANAGEMENT PROGRAMS

### B.2.2.1 345 kV $SF_6$ BUS

**Program Description**

The Seabrook Station 345kV $SF_6$ Bus Program is a new plant-specific program that will manage the following aging effects on the 345kV $SF_6$ Bus:

a. Loss of pressure boundary due to elastomer degradation

b. Loss of material due to pitting, crevice and galvanic corrosion

c. Loss of function due to unacceptable air, moisture or sulfur dioxide ($SO_2$) levels

Sulfur Hexafluoride ($SF_6$) is an inert gas used to insulate the bus conductor.

The program will inspect for corrosion on the exterior of the bus duct housing, test for leaks at elastomers and periodically test gas samples to determine air, moisture and $SO_2$ levels.

Program Elements

The following provides the results of the evaluation of each program element against the 10 elements described in Appendix A of NUREG-1800 Rev. 1, *"Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants"*.

Element 1 - Scope of Program

The Seabrook Station program is a new plant specific program that applies to $SF_6$ bus within the scope of license renewal.

The in-scope Seabrook Station $SF_6$ bus segments are those included in the recovery path for a Station Black Out event. Two possible recovery paths are identified. The components are shown on Figure 2.5-1.

The first path includes the $SF_6$ bus from 345 kV Power Circuit Breakers 11 and 163 to the Generator Step Up transformer. This path continues from the Generator Step Up transformer to Unit Auxiliary Transformers via the Isolated Phase Bus.

The second path includes the $SF_6$ bus from 345 kV Power Circuit Breakers 52 and 695 to the Reserve Auxiliary Transformers.

The purpose of the program is to maintain the pressure boundary formed by the bus exterior metal housing and elastomers. The program also monitors critical $SF_6$ parameters such as air, moisture and $SO_2$ levels.

The integrity of the elastomers will be routinely monitored by performing leak tests as part of the $SF_6$ Bus Program.

Element 2 - Preventive Actions

The Seabrook Station program performs tests and inspections. No actions are taken as part of this program to prevent or mitigate aging degradation.

Element 3 - Parameters Monitored/Inspected

Critical parameters of the $SF_6$ bus system are mechanical integrity of the system to maintain a pressure boundary, maintain acceptable air, moisture and $SO_2$ levels.

The Seabrook Station program maintains the integrity of the $SF_6$ pressure boundary. The program includes pressure monitoring of the $SF_6$ gas to insure that adequate insulating properties are maintained.

The Seabrook Station program performs periodic tests on samples of the $SF_6$ gas. These tests determine the air, moisture and $SO_2$ levels The $SO_2$ measurements provide an indication of arcing internal to the bus.

The Seabrook Station program performs inspections for loss of materials on the exterior surfaces of the duct.

Element 4 - Detection of Aging Effects

The Seabrook Station program tests samples of the $SF_6$ gas to determine if the insulating properties are adequate. These tests are focused on air, moisture and $SO_2$ levels. The $SO_2$ measurements provide an indication of arcing internal to the bus. The gas is sampled prior to entering the period of extended operation and at least once every six months thereafter.

This Seabrook Station program maintains the pressure boundary by monitoring the pressure of $SF_6$ gas and inspecting for leaks. The system $SF_6$ bus is inspected for leaks prior to entering the period of extended operation and at least once every six months thereafter.

This Seabrook Station program performs visual inspections on the exterior surfaces of the duct prior to entering the period of extended operation and at least once every six months thereafter.

Element 5 - Monitoring and Trending

The Seabrook Station program does include trending actions of the $SF_6$ properties. Trending provides additional data which can be analyzed to determine the rate of change in the measured parameter.

Element 6 - Acceptance Criteria

The Seabrook Station program performs leak tests, tests the quality of $SF_6$ gas, and inspects for loss of material.

The Seabrook Station program maintains the pressure boundary by inspecting for leaks and monitoring $SF_6$ gas pressure. The minimum acceptable pressure value is sufficient to provide adequate insulation between the conductor and the exterior housing. The $SO_2$ measurements of the $SF_6$ gas provide an indication of partial discharge occurring internal to the bus. Any indication of the presence of $SO_2$ will be evaluated by Engineering. The evaluation will provide any corrective actions required.

A dew point check is used to determine the moisture content of the $SF_6$ gas. The maximum allowable dew point measurement is below the dew point value that would lead to breakdown of the insulation.

A purity check is used to determine the air content of the $SF_6$ gas. The maximum allowable air content is below the value that would lead to breakdown of the insulation.

The presence of pitting, crevice and galvanic corrosion will be detected by visual inspections on the exterior surfaces of the duct. Engineering evaluations will be performed if corrosion is found on the $SF_6$ duct. The evaluation will include a determination of the ability of the remaining wall thickness to maintain the required pressure boundary.

Element 7 - Corrective Actions

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 7 Corrective Actions.

Element 8 - Confirmation Process

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 8 Confirmation Process.

Element 9 - Administrative Controls

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 9 Administrative Controls.

Element 10 - Operating Experience

Seabrook Station routinely performs monitoring and test activities for various parameters of the $SF_6$ bus. The inspections and tests are performed as part of the preventive maintenance activities. Results that are not acceptable are documented in the Corrective Action Program.

1. Seabrook Station relied on a review of the Corrective Action Program database to provide the basis of this review.

   In 2001 EPRI conducted $SF_6$ leak inspections at Seabrook Station. This operating experience demonstrates Seabrook's involvement in the industry to increase the reliability of the $SF_6$ switchyard.

   In 2008 an increase in the level of $SO_2$ was found in a sample of the $SF_6$ gas. An engineering evaluation attributed the increase to thermal cycling or partial discharge which occurs with normal switch operation. The $SF_6$ gas was filtered. Follow up tests were within acceptable limits. This operating experience demonstrates the ability of the Station to detect and analyze anomalies prior to loss of intended function.

2. Seabrook Station is performing extensive modifications to increase the reliability of the $SF_6$ switchyard. The first stages of the modifications were completed in 2009. Seabrook Station has additional long term plans to perform additional modifications to increase the reliability of the $SF_6$ switchyard. These modifications include realignment of high voltage breaker scheme, installation of new breakers, and bus sections. This operating experience demonstrates Seabrook Station's effort to upgrade the reliability of the switchyard.

3. $SF_6$ emissions are not currently subject to federal regulations, but are regulated under New Hampshire Air Toxic rules and subject to emission inventory reporting requirements under Seabrook Station's Title V Permit. Seabrook Station has partnered with Environmental Protection Agency (EPA) voluntary $SF_6$ Emission Reduction Partnership to reduce $SF_6$ emissions.

   Since 1999 Seabrook Station has submitted annual reports to the EPA to provide updates of the previous year's results. To meet the EPA goals aggressive maintenance and leak detection activities aimed at reducing $SF_6$ emissions are performed. In addition to environmental stewardship, maintaining low $SF_6$ leakage enhances the ability of the $SF_6$ bus to perform its intended function. This operating experience demonstrates that maintenance activities are effective in maintaining a safe environment and reliable switchyard.

The above operating experience demonstrates efforts to improve reliability of the $SF_6$ switchyard, involvement with industry, and ability to detect gas deficiencies.

**Exceptions to NUREG-1800**

None

**Enhancements**

None

**Conclusion**

The Seabrook Station 345kV $SF_6$ Bus Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.2.2 BORAL MONITORING

**Program Description**

The Seabrook Station Boral Monitoring Program is an existing, plant specific program that manages the aging effects of reduction of neutron absorbing capacity due to Boral degradation and changes in dimensions and loss of material due to general corrosion of the Boral neutron absorbing material in the Spent Fuel storage racks. The program assures the Boral neutron absorbers in the Spent Fuel racks maintain the validity of the criticality analysis in support of the rack design. The program relies on representative coupon samples mounted in a coupon "*train*" located in the Spent Fuel Pool to monitor performance of the absorber material without disrupting the integrity of the storage system. The program is performed on a fuel cycle basis so that these coupons receive the maximum available radiation exposure from each core off-load to the Spent Fuel Pool. The coupons are examined and evaluated prior to the next core off-load after most of the available coupon exposure has been realized. This typically occurs 2 to 4 months prior to the next refueling outage. Coupon samples are removed from the Spent Fuel Pool on a prescribed schedule and physical, chemical and neutronic absorptive properties are measured. From these data, the physical condition and neutron-absorbing capacity of the Boral in the storage cells are assessed. The coupons are placed back into the coupon train after inspection and testing. The train is returned to the Spent Fuel Pool 1 to 2 months prior to the next core off-load to support proper rehydration and acclimation of the coupons to the Spent Fuel Pool conditions. Location of the coupon train in the Spent Fuel

Pool and the loading pattern for spent fuel in the upcoming outage are determined at this time in order to optimize exposure to the Boral coupons.

Reduction of neutron-absorbing capacity, change in dimensions, and loss of material due to the effects of the Spent Fuel Pool environment are aging effects requiring management for Boral exposed to a treated borated water environment as described in Draft LR-ISG-2009-01, *"Staff Guidance Regarding Plant-Specific Aging Management Review and Aging Management Program for Neutron-Absorbing Material in Spent Fuel Pools"*. Because NUREG-1801, Section XI does not contain an Aging Management Program for monitoring the aging effects on Boral, the guidance provided in Draft LR-ISG-2009-01 was used to evaluate the Seabrook Station Boral Monitoring Program. Seabrook Station has reviewed the final LR-ISG-2009-01, "*Aging Management of Spent Fuel Pool Neutron-Absorbing Materials Other than Boraflex*", dated May 4, 2010 and determined that the Seabrook Station Boral Monitoring Program Meets the requirements of the final guidance.

The Boral Monitoring Program is used to manage the aging effect of reduction of neutron-absorbing capacity. Inspection of the surveillance coupons for evidence of change in dimensions and loss of material due to the effects of the Spent Fuel Pool environment is included in the Boral Monitoring Program and serves as the indicator of similar degradation of the actual Boral sheets in the Spent Fuel racks.

Boral sheets consist of a core of uniformly distributed boron carbide in an alloy 1100 aluminum matrix with a thin aluminum clad on both sides. The core "*cermet*" (ceramic-metallic) is slightly porous and exposed to the wet pool environment along the sheared edges of each sheet.

Two degradation mechanisms have been observed and identified in the Seabrook Station Boral Monitoring Program. The first mechanism indicates blisters in the Boral clad formed as the clad separates at the clad-cermet interface under the influence of internal gas pressure generated within the cermet porosity under Spent Fuel Pool conditions. A Part 21 (2003-0022-00) notification was initiated by Seabrook Station on discovery of these blisters. The second observed degradation process involves aluminum corrosion within the Boral clad and core matrix. Corrosion through the clad and into the cermet matrix frees boron carbide particulate to be eroded away from the racks by pool water.

The Seabrook Station Boral Monitoring Program will continue to monitor the neutron absorbing capacity of the Boral sheets in the Spent Fuel Pool through the period of extended operation. Should this program show, through evaluation of the Boral coupons, that this capacity has degraded such that the subcriticality margin in the pool becomes challenged, appropriate evaluation and actions will be determined in accordance with the Seabrook Station Corrective Action Program.

Program Elements

The following provides the results of the evaluation of each program element against the 10 elements described in Appendix A of NUREG-1800 Rev. 1, *"Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants"*.

Element 1 - Scope of Program

The Seabrook Station Spent Fuel Pool is divided into two regions with twelve free standing and self-supporting racks. Region 1 has six racks with Boral as the neutron absorber that allow space for 576 fuel assemblies and Region 2 has six racks with Boraflex that allow space for 660 fuel assemblies. The Boraflex utilized in the Region 2 racks is not credited in the criticality analyses and is therefore not within the scope of License Renewal. The scope of this Seabrook Station Boral Monitoring Program is the management of the reduction of neutron-absorbing capacity of the Boral sheets in Region 1. This is accomplished by monitoring neutron-absorbing capacity, inspecting for changes in dimensions and inspecting for loss of material due to general corrosion caused by the effects of the Spent Fuel Pool environment on representative Boral coupons.

Element 2 - Preventive Actions

This is a condition monitoring and inspection program and therefore there are no preventive actions required.

Element 3 - Parameters Monitored/Inspected

Seabrook Station utilizes standard Boral coupons of the same design as, and traceable to, the specific Boral heat lot material used in the fabrication of the Spent Fuel Racks. The standard coupons are placed in the Spent Fuel Pool for monitoring of the aging effects. Control coupons were supplied in addition to standard coupons to benchmark coupon initial conditions, monitor possible uncontrolled changes in Boral material that are unrelated to the Spent Fuel Pool conditions, and to demonstrate comparisons between different examination techniques and service contractors. The control coupons are kept out of the Spent Fuel Pool unless needed to replace a standard coupon that has become unavailable.

The program monitors changes in the following physical properties of the Boral material by visual examination of the Boral coupons by Seabrook Station personnel. Selected control coupons are used for comparison.

a. Blistering, pitting, cracks, corrosion and spalling, loss of material, and other damage or condition

b. Dimensional measurements (length, width and thickness)

Two or more Boral coupons are selected each refueling outage for examination by an outside contractor. Neutron attenuation, neutron radiography examination and other nondestructive examinations are performed. The Boral monitoring cycle engineering evaluation provides guidance in determining the specific examinations to be performed during a given monitoring cycle.

A control coupon with traceability to the Boral sheets installed in the Spent Fuel Pool racks is also examined by neutron attenuation and radiography for comparison.

Contractor examinations support the determination of boron-10 ($B^{10}$) areal density and the boron distribution within the Boral coupons. This information is used to evaluate any reduction of neutron-absorbing capacity noted in the Boral coupons. These examinations also provide information related to the size and volume of blisters on the coupons. Boral blisters can result in small but quantifiable reactivity effects due to moderator or dissolved boron displacement. This information is used to determine the overall effect of accrued blistering of the Boral sheet surfaces on the Spent Fuel Pool criticality analysis.

<u>Element 4 - Detection of Aging Effects</u>

The program monitors coupon samples located in the Spent Fuel Pool to determine the condition of the neutron absorber material without disrupting the integrity of the Spent Fuel storage system. The program measures certain physical and chemical properties of these sample coupons each refueling outage as described above. From these data, the stability and integrity of the Boral in the Spent Fuel racks are assessed relative to any reduction in the neutron-absorbing capacity of the Boral sheets and to any degradation of the sheets as a result of corrosion or blistering.

The Seabrook Station Boral Monitoring Program maintains the coupon train within the Spent Fuel Pool positioned such that the coupons experience the same conditions as the Boral panels built into the actual fuel racks. The coupons are mounted in stainless steel jackets and stainless steel coupon train mimicking the construction of the fuel racks. This realism in the program is designed to recreate the Spent Fuel Pool environment for known effects and potential effects that may be unknown at this time. Early Boral coupon programs in the industry did not recognize that the radiation and electrolytic effects, water temperature, thermodynamic response of the coupon in its jacket, gamma heating of the coupon and jacket as well as radiolytic chemical species generated in the racks all contribute to the reactions affecting the Boral material.

Element 5 - Monitoring and Trending

Neutron attenuation tests are trended to ensure that degradation does not challenge the assumptions within the Spent Fuel Pool Criticality Analysis of record. Observable loss in neutron attenuation ability, if any, is projected to determine when neutron attenuation may fall below acceptance criteria. Size and weight measurements determine the extent of shrinkage or loss of material. These data are trended for indications of degradation. Blister shape and size are recorded and trended to determine whether new blisters are forming, the rate of growth of existing blisters, and the rate of increase in blister thickness. By the Seabrook Station design, all blister growth is directed into the flux trap space between fuel cells and not into the cell. This design, combined with the thickness of the cell box wall (90 mils), precludes blister impingement on or interference with the fuel assembly.

Element 6 - Acceptance Criteria

Acceptance criteria for the following properties are applied to each exposed Standard Boral coupon inspected. Failure to meet an acceptance criterion is addressed by subsequent engineering evaluations.

a. *Voided Blister Displacement* - The total blister void volume for all blisters present on both sides of a coupon will be less than a 45 mil uniform void over the area of the coupon. The rate of change in blister displacement provides indication of availability of sufficient margin to avoid exceeding the 45 mil uniform void prior to the next Boral coupon examination.

b. *Boron Carbide Loss* - $B^{10}$ areal density measured by thermal neutron attenuation will be greater than 0.02 $gm/cm^2$ as specified within the criticality analysis and material specification. The rate of change in boron carbide loss provides indication of availability of sufficient margin to maintain the 0.02 $gm/cm^2$ $B^{10}$ areal density beyond the next Boral coupon examination.

c. *Boron Carbide Redistribution* - Boron carbide distribution will be uniform as observed by thermal neutron radiography. Thinned or depleted areas will satisfy the criterion for boron carbide loss discussed above.

The purpose of the Seabrook Station Boral Monitoring Program is to ensure that degradation does not challenge the design bases and assumptions within the Spent Fuel Pool Criticality Analysis of record. The design of the Region 1 Spent Fuel racks containing Boral as a neutron absorbing material assures a $K_{eff} < 0.95$ (5% subcriticality margin).

Element 7 - Corrective Actions

A set of remedial action recommendations is prepared as required to maintain the acceptable Boral function within the Spent Fuel Pool. These remedial action recommendations are prescribed based on current Boral coupon inspections. Specific recommendations for the selection of coupons for examination and inspection during the next monitoring cycle are specified.

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 7 Corrective Actions.

Element 8 - Confirmation Process

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 8 Confirmation Process.

Element 9 - Administrative Controls

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 9 Administrative Controls.

Element 10 - Operating Experience

1. During planned work involving an inspection of the Spent Fuel Pool Boral coupon tree in 2003, an unexpected blistering of the Boral material was identified when one of the Boral coupons was examined.

   The condition report evaluation concluded that the effect on the current Spent Fuel Pool criticality analysis was insignificant and the current blistered condition was acceptable as is. The evaluation stated that the degree of Boral blistering was expected to increase with repeated exposure to gamma energies present during offload; as such a Boral-monitoring program was established to evaluate future changes in the Boral material. Since the Boral monitoring program would not gather any additional data on the blistering events until after the next core offload, a water reduction in the flux trap equal to 90 mils was analyzed and applied to the revised criticality analyses to formally accommodate any increased blistering at offload. The revised type determination curves are conservative to the existing curves at all points, and were implemented prior to core offload. The type determination curves with the 90 mil allowance were included to accommodate any future blistering. This allowance is used as an acceptance criterion for the Boral monitoring program. Other acceptance criteria will include the $Boron^{10}$ areal density.

2. As of January 2003, a Boral Monitoring Program had not yet been formally established following implementation of the engineering change to incorporate Boral instead of Boraflex in the second set of fuel racks. Although no commitment had been made to implement such a program, Seabrook Station opted to establish a Boral coupon monitoring program as a good practice.

3. During the Cycle 10 monitoring program (Spring of 2005), aluminum cladding oxidation and spalling was observed on Boral coupons. Photos of these coupons taken in the previous monitoring cycle were reviewed and showed oxidation but no evidence of spalling. The progression and effect of this oxidation and spalling was evaluated and predicted to remain within the program acceptance criteria through the next coupon examination in Cycle 11, when the material would be re-evaluated.

   The Boral oxidation and spalling condition was described and posted with INPO as operating experience on August 26, 2005.

   The Cycle 11 examinations (Fall of 2006) indicated continued aluminum cladding oxidation on most coupons. The potential degradation of neutron absorbing capacity due to continued, and eventually through-wall, oxidation and spalling was evaluated by observing previously dissected blisters on special coupon A131. Blisters on this coupon had been intentionally dissected to investigate the effect on the Boral should a blistered area break through. By dissecting the blisters, the cermet compound was now exposed directly to the Spent Fuel Pool water. The altered coupon A131 with dissected blisters had been exposed to the Spent Fuel Pool conditions for approximately 3 years, and was then indicating measurable change in $B^{10}$ areal density in the bare cermet.

   The results of the Cycle 11 examinations indicated continued aluminum cladding oxidation. The Boral coupons did, however, remain well within the areal density specification. The change in $B^{10}$ areal density was just above the lower limit of detection by visual examination. The corrosion process appeared to be proceeding very slowly.

   The potential for measurable $B^{10}$ loss in the unaltered coupons was reasonably expected within the next few cycles. Therefore long term $B^{10}$ areal density monitoring, via neutron attenuation, was also implemented to ensure conformance to Boral specifications.

These operating experience items illustrate the effectiveness of the Seabrook Station Boral Monitoring Program in identifying and addressing issues that may impact the neutron absorbing capacity of the Boral materials. Aging mechanisms of reduction of neutron absorbing capacity and loss of material are adequately managed through the Seabrook Station Boral Monitoring Program as evidenced by this type of operating experience.

**Exceptions to NUREG-1800**

None

**Enhancements**

None

**Conclusion**

The Seabrook Station Aging Management Program for Boral Monitoring provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### B.2.2.3 NICKEL-ALLOY NOZZLES AND PENETRATIONS

**Program Description**

The Nickel-Alloy Nozzles and Penetrations Program is an existing plant specific program that manages the aging effect of cracking due to primary water stress corrosion cracking (PWSCC) of nickel based alloy pressure boundary and structural components exposed to reactor coolant.

<u>Program Elements</u>

The following provides the results of the evaluation of each program element against the 10 elements described in Appendix A of NUREG-1800 Rev. 1, *"Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants".*

<u>Element 1 - Scope of Program</u>

The Seabrook Station Nickel-Alloy Nozzles and Penetrations Program is described in the Seabrook Station Reference Manual – RCS Materials Degradation Management Reference.

The scope of the program includes;

a. Pressurizer Nozzles

b. Steam Generator Channel Head Drain Tube and Welds
c. Reactor Vessel Core Support Pad/Lug, and Clevis Inserts
d. Reactor Vessel Hot Leg Nozzles
e. Reactor Vessel Cold Leg Nozzles
f. Reactor Vessel Bottom Mounted Instrumentation Penetrations

The program does not include Steam Generator tubes or secondary side components (included in the Steam Generator Tube Integrity Program, B.2.1.10), Reactor Vessel Internals (included in the PWR Vessel Internals Program, B.2.1.7), or control rod drive mechanism nozzles and reactor head vent nozzle (included in the Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program, B.2.1.5).

The program is based upon the industry guidance provided in EPRI MRP-126, *"Materials Reliability Program: Generic Guidance for Alloy 600 Management"*, NEI 03-08, *"Guideline for the Management of Materials Issues"*, EPRI MRP-139 Rev 1, *"Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline"*, and ASME Section XI including Code Case N-722, *"Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Division 1"* as approved for incorporation under 10CFR 50.55a with conditions in accordance with the requirements in paragraphs (g)(6)(ii)(E).

The Seabrook Station Nickel Alloy Nozzles and Penetrations Program complies with applicable NRC Orders and implements applicable NRC Bulletins, Generic Letters and staff-accepted industry guidelines.

<u>Element 2 - Preventive Actions</u>

The Seabrook Station Nickel-Alloy Nozzles and Penetrations Program considers various mitigative and repair options to ensure that nickel-alloy components continue to perform their intended functions during the period of extended operation. Some of the currently available mitigation techniques include mechanical stress improvement, induction heat stress improvement, weld overlay, mechanical nozzle seal assembly, zinc injection, abrasive water jet, nickel plating or replacement with Alloy 690/52/152 components.

Most mitigative actions implemented by the industry since the mid-1990s have utilized primary water stress corrosion cracking resistant Alloy 690/52/152 materials.

Additional preventive measures to mitigate primary water stress corrosion cracking are in accordance with the Seabrook Station Water Chemistry

Program (B.2.1.2). The Water Chemistry Program manages aging effects by controlling concentrations of known detrimental chemical species such as chlorides, fluorides, sulfates and dissolved oxygen below the levels known to cause degradation. The program includes specifications for chemical species, sampling and analysis frequencies and corrective actions for control of water chemistry. This program conforms to the EPRI "*Pressurized Water Reactor Primary Water Chemistry Guidelines.*"

Element 3 - Parameters Monitored/Inspected

The Seabrook Station Nickel-Alloy Nozzles and Penetrations Program incorporates the inspection schedules and frequencies for the nickel-alloy components in accordance with the Seabrook ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program and, where applicable, ASME Code Case N-722, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(E).

The program administers component evaluations, examination methods, scheduling, and site documentation as required to comply with regulatory, code or industry commitments related to Nickel Alloy issues.

The Seabrook Station Nickel-Alloy Nozzles and Penetrations Program monitors for cracking due to primary water stress corrosion cracking of Alloy 600/82/182 materials exposed to reactor coolant. The program performs condition monitoring examinations of the lower Reactor Vessel head surface and each bottom-mounted instrumentation tube penetration. These examinations monitor for through-wall cracks that may exist in the nozzles or their associated partial penetration J-groove welds. For other in-scope pressure boundary components, the program monitors for evidence of Reactor Coolant leakage which may manifest itself in the form of boric acid residues or corrosion products.

The core support pads/lugs and clevis insert are monitored for evidence of cracking. They are identified in the Seabrook Station Inservice Inspection Reference Manual as category B-N-2 Welded Core Support Structures and Interior Attachments to Reactor Vessels that are VT-3 inspected once per interval.

Element 4 - Detection of Aging Effects

The Seabrook Station Nickel-Alloy Nozzles and Penetrations Program utilizes visual and volumetric examination techniques to detect cracking in Alloy 600/82/182 materials. This Program implements the inspection of the Alloy 600/82/182 materials through the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program.

The Seabrook Station Nickel-Alloy Nozzles and Penetrations Program uses a number of inspection techniques to detect cracking due to primary water stress corrosion cracking. These include surface examinations, volumetric examinations, and bare metal visual examinations.

Bare metal visual examinations are similar to VT-2 examinations but require removal of insulation to allow direct access to the metal surface while pressurized or not pressurized. The nickel alloy components have been ranked based on susceptibility, safety, and economic consequences of degradation/failure. Where applicable, EPRI MRP-139 PWSCC susceptibility categories have been assigned to the components. This information is contained in the Seabrook Station Reference Manual - RCS Materials Degradation Management Reference and includes the categorization, description of the weldments and the examination extent and schedule.

Detection of cracking due to PWSCC is used to ensure that nickel alloy components meet required design attributes and maintain their availability to perform their intended function as designed when called upon. This program will detect age-related degradation prior to component failure. When required, repair or mitigation is used to ensure that components will meet the design requirements required to perform their intended function.

<u>Element 5 - Monitoring and Trending</u>

The Seabrook Station Nickel-Alloy Nozzles and Penetrations Program ranked the Alloy 600/82/182 locations based on four main criteria: PWSCC susceptibility (e.g., operational time and temperature), failure consequence, leakage detection margin, and radiation dose rates. Additionally, material heat susceptibility and other industry experience were considered.

The program incorporates the inspection schedules and frequencies for the nickel-alloy components in accordance with the Seabrook ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program and, where applicable, ASME Code Case N-722, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(E).

Flaw indications detected during the required examinations are dispositioned in accordance with the Acceptance Criteria and Corrective Actions program elements discussed below.

In accordance with ASME Code Case N-722, visual examinations of highly susceptible Alloy 600/82/182 pressure retaining components are required during each refueling outage. Other Alloy 600/82/182 pressure retaining components that are considered less susceptible to primary water stress corrosion cracking are required to be examined by visual examinations every other refueling outage or once per interval.

The Alloy 600 inspections and schedule are specified in the Seabrook Station Reference Manual - RCS Materials Degradation Management Reference.

a. Pressurizer Nozzles:

In accordance with 10 CFR 50.55a(g)(6)(ii)(E)(1), the inspection requirements of ASME Code Case N-722 do not apply to components with pressure retaining welds fabricated with Alloy 600/82/182 materials that have been mitigated by weld overlay. The six pressurizer nozzles that were mitigated with full structural weld overlays will revert to regular 10 year Inservice Inspection period following volumetric inspection in the first or second outage following installation. The near term schedule for their inspection calls for ultrasonic examinations during Refueling Outages 14 (Spring of 2011) and 17 (Fall of 2015).

b. Steam Generator Channel Head Drain Tube and Welds:

The Steam Generator channel head (bowl drains) tube and welds are subject to a bare metal visual inspection every refueling outage.

c. Reactor Vessel Core Support Pads/Lugs, Clevis Inserts:

The Reactor Vessel core support pad/lugs, clevis inserts are subject to visual examination of accessible welds under the Seabrook Station ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. No additional or augmented inspections are required under the Seabrook Station Nickel-Alloy Nozzles and Penetrations Program at this time.

d. Reactor Vessel Hot Leg Nozzles:

The Reactor Vessel hot legs were ultrasonically examined during Refueling Outage 13 (Fall of 2009). Visual and Ultrasonic Thickness inspections are specified in the Seabrook Station Reference Manual - RCS Materials Degradation Management Reference. The near term inspection schedule for the hot leg nozzles for hot legs A, B, and C, calls for bare metal visual inspections during Refueling Outages 14 (Spring of 2011), 15 (Fall of 2012), 17 (Fall of 2015), and 18 (Spring of 2016) and ultrasonic examinations during Refueling Outage 16 (Spring of 2014). The nozzle for hot leg D underwent a mechanical stress improvement process during Refueling Outage 13 (Fall of 2009). Near term inspections for this nozzle call for ultrasonic inspections during Refueling Outages 14 (Spring of 2011), 16 (Spring of 2014), and 18 (Spring of 2016).

e. Reactor Vessel Cold Leg Nozzles:

The Reactor Vessel cold legs were ultrasonically examined during Refueling Outage 13 (Fall of 2009). Future visual and ultrasonic examinations are contained in the Seabrook Station Reference Manual - RCS Materials Degradation Management Reference. The near term inspection schedule for the cold leg nozzles A, B, C, and D, calls for a bare metal visual inspection during refueling outage 16 (Spring of 2014) and ultrasonic examination during refueling outage 17 (Fall of 2015).

f. Reactor Vessel Bottom Mounted Instrumentation Penetrations:

ASME Code Case N-722 requires bare metal visual inspection of the Reactor Vessel bottom head penetrations every other refueling outage.

Element 6 - Acceptance Criteria

For the Reactor Vessel core support pads/lugs and clevis inserts, the Seabrook ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program conducts visual VT-1 examination of the accessible welds. The Seabrook ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program require that indications and relevant conditions detected during examination be evaluated in accordance with ASME Section XI, Paragraph IWB-3520.1.

The acceptance standards of ASME Section XI, Paragraph IWB-3522 are also applied to relevant indications identified during system pressure testing performed in accordance with ASME Section XI, Table IWB-2500-1, Examination Category B-P.

Reactor Vessel bottom head bare metal visual examinations are performed in order to identify very small volumes of boric acid that may result from Alloy 600 PWSCC. The acceptance criteria for this examination is the lack of any relevant indication, namely evidence of any leakage arising from the penetration to head interface, and the lack of any boric acid accumulations on the carbon steel head surfaces that may result in corrosion. The acceptance standards are in accordance with ASME Section XI, Paragraph IWB-3522 per ASME Code Case N-722, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(E).

The acceptance criteria, against which the need for corrective actions are evaluated, ensure that the component intended functions are maintained under all current licensing basis design conditions during the period of extended operation.

Element 7 - Corrective Actions

Indications are evaluated per the acceptance criteria, which determine relevant flaw indications that are unacceptable for further service. Unacceptable flaw indications are corrected through implementation of appropriate repair/replacement activities.

If visual examination of the Reactor Vessel instrumentation tube penetrations (bottom head) in accordance with ASME Code Case N-722 identifies leakage or evidence of cracking, additional actions will be performed as specified in paragraphs 10 CFR 50.55a(g)(6)(ii)(E)(2) through (4).

If PWSCC related indications are detected in the pressurizer surge nozzle full structural weld overlay, the repair/replacement activity will include removal of the weld overlay and the original dissimilar metal weld.

Repair/replacement activities comply with ASME Section XI as invoked by 10 CFR 50.55a or approved ASME Code Cases as referenced in the latest version of NRC Regulatory Guide 1.147, *"Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."* Proposed alternative repair/replacement activities, if any, will be submitted to the NRC for review and approval in accordance with 10 CFR 50.55a (a)(3)(i) or 10 CFR 50.55a(a)(3)(ii).

Identified flaw indications are entered into the Seabrook Station Corrective Action Program for appropriate disposition. A repair, replacement or evaluation is performed for all flaw indications that exceed the acceptance criteria.

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 7 Corrective Actions.

<u>Element 8 - Confirmation Process</u>

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 8 Confirmation Process.

<u>Element 9 - Administrative Controls</u>

The FPL/NextEra Energy Quality Assurance Program and Nuclear Fleet procedures will be utilized to meet Element 9 Administrative Controls.

<u>Element 10 - Operating Experience</u>

A review of operating experience for the Seabrook Station Nickel-Alloy Nozzles and Penetrations Program identified no adverse trends with program performance. The review of operating experience, as discussed below, indicates that the Nickel-Alloy Nozzles and Penetrations Program is effective in utilizing inspections, mitigation techniques, and repair/replacement activities.

1. During Refueling Outage 7 (Fall of 2000), insulation was removed from the Reactor Vessel hot leg nozzles and a bare metal visual examination

conducted. This was in response to through-wall leakage discovered at V.C.Summer. No evidence of pressure boundary leakage was observed.

2. During Refueling Outage 9 (Fall of 2003), inspection of the lower Reactor Vessel head bottom mounted instrumentation penetrations was conducted. No evidence of boric acid leakage was observed on the 58 bottom mounted instrumentation penetrations. Tape residue and boric acid trails were observed on the bottom head surface. Subsequently, the tape residue was evaluated by Engineering and determined to be acceptable. Additionally, three small boric acid drip ends were obtained for analysis, which revealed the boric acid to be greater than 7 years old. This was consistent with previous analysis and known cavity seal ring leakage prior to installation of the permanent cavity seal ring. The three subject locations were cleaned and inspected. The inspection results revealed no areas of metal degradation.

3. During Refueling Outage 10 (Spring of 2005), bare metal visual inspection of six Pressurizer butt welds and four Steam Generator bowl drain connections were conducted. The Pressurizer butt welds included a surge nozzle on the bottom, three safety nozzles, one relief nozzle, and a spray nozzle on top. The four Steam Generator bowl drains and four Pressurizer nozzles showed no evidence of leakage. Two nozzle welds on the Pressurizer exhibited white residue believed to be liquid penetrant developer. These two nozzles were Liquid Penetrant tested to verify no weld indications existed.

4. During Refueling Outage 11 (Fall of 2006), bare metal visual inspection was performed on six Pressurizer butt welds, four Steam Generator bowl drains, eight Reactor Vessel butt welds, Reactor Vessel O-ring leak taps, and Reactor Vessel Bottom Mounted Nozzles. None of these inspections showed evidence of leakage.

5. During Refueling Outage 12 (Spring of 2008), the six pressurizer nozzles were mitigated by installing full structural weld overlays using Alloy 52M weld material. Additionally, Bare Metal Visual examinations were performed on the four Steam Generator bowl drains with satisfactory results.

6. During Refueling Outage 13 (Fall of 2009), all eight Reactor Vessel nozzle butt welds were volumetrically inspected to satisfy the requirements of ASME Section XI and EPRI MRP-139 examination requirements. This meets the mandatory inspection deadline of December 31, 2009, for the hot legs and December 31, 2010, for the cold legs. During the inspection an axial flaw indication was found on Reactor Vessel loop "D" hot leg nozzle in the alloy 82/182 material connected to the inner diameter. That nozzle was

subsequently mitigated by the Mechanical Stress Improvement Process. Other alloy 600 locations inspected in Refueling Outage 13 were the Reactor Vessel bottom mounted nozzles, Reactor Vessel o-ring leak off lines, and the steam generator bowl drains. These inspections were satisfactory

The "D" hot leg nozzle axial flaw indication was identified as an inside diameter connected planar flaw in the Reactor Vessel outlet nozzle-to-safe-end dissimilar metal weld located at vessel orientation 158°. This approximately 21% through-wall flaw exceeded the acceptance standards contained in ASME Section XI, Table IWB-3410-1.

The ASME Code allows flaw acceptance if they meet specific analytical analysis. A Seabrook Station document *"Seabrook Reactor Vessel Outlet Nozzle Dissimilar Metal Weld Flaw Evaluation"* provides an analysis that concluded the flaw has an allowable service life of just under 36 months to remain in compliance with ASME Section XI. However, Seabrook Station elected to mitigate the subject location during Refueling Outage 13 to prevent flaw propagation through the next operating cycle. The mitigation technique of mechanical stress improvement process was chosen. This is a proven and accepted process, which prevents flaw initiation and flaw propagation.

The cause evaluation concluded that based on published information, the likely cause of the flaw was the susceptible Alloy 600 material being exposed to primary system fluid at hot leg temperature for 16.53 effective full power years.

*Corrective actions taken included:*

a) A detailed review of the previous 1999 electronic ultrasonic data for the Reactor Vessel outlet nozzle-to-safe-end weld at 158° was performed to determine if the newly detected flaw was present in the 1999 data. This review was performed using the same data analysis software that was used for the 2009 examination. After a thorough review of the data by the vendor, EPRI, and the Seabrook Station Level III UT examiner, it was determined that the newly reported flaw was not present in the 1999 data; and

b) An extent of condition review was performed. The Seabrook Station Reference Manual - RCS Materials Degradation Management Reference was previously issued to address Seabrook Station's plan to mitigate these welds. The manual lists the Alloy 600 locations (including the 4 hot and 4 cold legs) and inspection schedules. The Seabrook Station manual also incorporates the guidance published in EPRI MRP-139. The six butt welds on the Pressurizer had already been mitigated during Refueling Outage 12

(Spring of 2008) by full structural weld overlay. This process adds non-susceptible weld material over the susceptible area essentially rendering a new pressure boundary. Future plans include mitigation of remaining butt welds on the Reactor Vessel nozzle to safe-end welds using an approved mitigation technique. Currently, mechanical stress improvement process and weld overlay are approved techniques.

The above operating experience examples provide objective evidence that the Seabrook Station Nickel-Alloy Nozzles and Penetrations Program effectively monitors the condition of components within the license renewal boundary and ensures aging effects are acceptably managed.

**Exceptions to NUREG-1800**

None

**Enhancements**

None

**Conclusion**

The Seabrook Station Nickel-Alloy Nozzles and Penetration Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.3 NUREG-1801 CHAPTER X AGING MANAGEMENT PROGRAMS

### B.2.3.1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

**Program Description**

The Metal Fatigue of the Reactor Coolant Pressure Boundary Program is an existing program that includes measures to prevent fatigue cracking caused by anticipated cyclic strains in metal components of the Reactor Coolant pressure boundary. This is accomplished by monitoring and tracking critical thermal pressure transients for select Reactor Coolant system components ensuring the number of design transient cycles is not exceeded during the operating life. Based on design basis screening criteria, a list of fatigue-sensitive components is developed and maintained. Fatigue-sensitive components include locations such as the Reactor Vessel Shell and Lower Head, Reactor Vessel Inlet and Outlet Nozzles, Pressurizer Surge Line (Hot Leg and Pressurizer Nozzles), Reactor Coolant Piping Charging System Nozzle, Reactor Coolant Piping Safety Injection Nozzle, and Residual Heat Removal (RHR) System Class 1 Piping.

The Metal Fatigue of Reactor Pressure Boundary Program is a preventive program that monitors and tracks the number of critical thermal and pressure transients to ensure that the cumulative usage fatigue (CUF) for select reactor coolant system components remain less than 1.0 through the period of extended operation. The program determines the number of transients that occur and updates 60-year projections on an annual basis. The program is credited with monitoring CUF of the reactor vessel, the pressurizer, the steam generators, Class 1 and non-Class 1 piping, and Class 1 components subject to the reactor coolant, treated borated water, and treated water environments. The program will use fatigue monitoring software to monitor the number of cycles a system or component endures. Pre-established cycle limits will identify components approaching design limits. Components approaching design limits will be reanalyzed, inspected, repaired or replaced in accordance with applicable design codes.

NUREG/CR-6260, *"Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components"*, provides specific guidance to address environmental effects and recommendations for selection of critical components in high-fatigue usage locations that should be monitored. Formulas for calculating the environmental correction factors for carbon and low alloy steel, and stainless steel are contained in NUREG/CR-6583, *"Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-*

*Alloy Steels",* and NUREG/CR-5704, *"Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels",* respectively.

Seabrook Station has ensured the environmental effect on fatigue sensitive locations are addressed for the period of extended operation. Where specific component monitoring is required, software will track individual cycles and events. Locations with CUF approaching the design limit will be reanalyzed, inspected, repaired, or replaced as necessary in accordance with applicable design codes. Corrective action may encompass one of several activities:

1. Reanalyze affected component(s) for an increase in the number of that specific transient while accounting for other component-affecting plant transients that may be projected not to achieve their analyzed levels.

2. Perform a fracture mechanics evaluation of a postulated flaw in affected plant components, which, when coupled with an inservice inspection program, will serve to demonstrate flaw tolerant behavior.

3. Repair the affected component.

4. Replace the affected component.

The evaluation of environmental fatigue effects for the Reactor Vessel Shell and Lower Head and Reactor Vessel Inlet and Outlet Nozzles found that the CUF will remain below the ASME code allowable fatigue limit of 1.0 using the maximum applicable $F_{en}$, when extended to 60 years. The evaluation of fatigue effects for these locations has thereby been validated for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i), including effects of the reactor coolant environment.

The remainder of these locations, RCS Pressurizer Surge Line Nozzle, RCS Charging Nozzle, RCS Safety Injection Nozzle, and RCS Residual Heat Removal System Class 1 Piping, were analyzed in accordance with NB-3200, based on Seabrook Specific conditions and will be monitored for fatigue usage including environmental effects by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Pre-established action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor, including environmental effects, exceeds the ASME Code limit of 1.0.

Two locations that were analyzed to have a CUF, including environmental effects, greater than 1.0, during the period of extended operation, are the Surge Line Hot Leg Nozzle-to-Pipe Weld and the Charging Nozzle near Blend Radius. At least two (2) years prior to entering the period of extended operation, for the plant-specific locations listed in NUREG/CR-6260 for newer vintage Westinghouse plants, Seabrook Station will implement the following.

1. Seabrook Station will update the fatigue usage calculations using refined fatigue analyses, if necessary, to determine acceptable CUFs (i.e., less than 1.0) when accounting for the effects of the reactor water environment. This includes applying the appropriate $F_{en}$ factors to valid CUFs determined from an existing fatigue analysis valid for the period of extended operation or from an analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). Formulas for calculating the environmental life correction factors for carbon and low alloy steels are contained in NUREG/CR-6583 and those for austenitic stainless steels are contained in NUREG/CR-5704. NUREG/CR-6909 includes alternate formulas for calculating environmental life correction factors, in addition to updated fatigue design curves.

2. If acceptable CUFs cannot be demonstrated for the selected locations, then additional plant-specific locations will be evaluated. For the additional plant-specific locations, if CUF including environmental effects are greater than 1.0, then Corrective Actions will be initiated. Corrective Actions will include inspection, repair, or replacement of the affected locations before exceeding a CUF of 1.0 or the effects of fatigue will be managed by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

The Metal Fatigue of Reactor Coolant Pressure Boundary Program will maintain fatigue within the design code limit through the period of extended operation. The software program will monitor cycles to verify cycle limits are maintained below limits specified in the UFSAR.

The program includes generation of a periodic fatigue monitoring report, including a listing of transient events, cycle summary event details, cumulative usage factors, a detailed fatigue analysis report, and a cycle projection report. If the fatigue usage for any location has had an unanticipated increase based on cycle accumulation trends or if the number of cycles is approaching their limit, the corrective action program is used to evaluate the condition and determine the corrective action. Acceptable corrective actions include a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the period of extended operation, inspection of the component, repair of the component, and replacement of the component. Corrective actions include a review of additional affected reactor coolant pressure boundary locations.

**NUREG-1801 Consistency**

This program is consistent with NUREG-1801 X.M1.

**Exceptions to NUREG-1801**

None

**Enhancements**

The following enhancements will be made prior to entering the period of extended operation.

1. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional transients beyond those defined in the Technical Specifications and UFSAR.

   *Program Elements Affected: Element 3 (Parameters Monitored/Inspected).*

2. The program will be enhanced to use a software program to count transients to monitor cumulative usage on select components.

   *Program Elements Affected: Element 1 (Scope of Program), Element 3 (Parameters Monitored/Inspected), Element 5 (Monitoring and Trending), and Element 6 (Acceptance Criteria).*

**Operating Experience**

Demonstration that the aging effects are effectively managed is achieved through objective evidence that shows aging effects and mechanisms are being adequately managed. The following examples provide objective evidence that the Metal Fatigue of Reactor Coolant Pressure Boundary program will be effective in assuring that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

1. NRC Bulletin 88-11, issued in December 1988, requested utilities to establish and implement a program to confirm the integrity of the pressurizer surge line. The program required both visual inspection of the surge line and demonstration that the design requirements of the surge line are satisfied, including the consideration of stratification effects.

   In 1989, Seabrook Station responded to the NRC Bulletin 88-11, by analyzing and demonstrating the acceptability of stress fatigue in the pressurizer surge line. Seabrook Station reviewed the pressurizer surge line temperature and displacement data collected during the first operating cycle and verified that this data was enveloped by the thermal stratification design transients used in the structural and fatigue evaluation of the pressurizer surge line. The results indicated that the cumulative usage factor met the acceptance criteria of less than 1.0.

This example provides objective evidence that the existing Metal Fatigue of Reactor Coolant Pressure Boundary program is capable of utilizing industry information in preventing cumulative fatigue damage of sample reactor coolant system components.

2. NRC Bulletin 88-08 was issued June 22, 1988 with supplements in 1988 and 1989 because of observed pipe cracking due to valve leakage in unisolable lines. The Bulletin required that licensees identify potential locations that might be subject to high stresses due to leaking valves, inspect the potential locations, and to assure that susceptible locations will not fail for the remaining life of the unit.

   Seabrook Station evaluated the possibility of fluid in-leakage by identifying seven piping sections that are unisolable from the Reactor Coolant System and pressurized by the charging pumps. These areas were evaluated for the effects of thermal stresses due to leaking valves that could potentially experience in-leakage to the Reactor Coolant System. Four of the lines are the High Head Safety Injection lines and the other three are the Charging System lines. In 1988, Seabrook Station performed a one-time non-destructive examination for the four High Head Safety Injection lines, showing acceptable results. Non-destructive examinations were not considered to be necessary for the three Charging System lines because they had not yet been subjected to excessive thermal cycling at that time.

   Additionally, a temperature monitoring program for the High Head Safety Injection and Charging System lines was deployed in 1989. In this program, temperature detectors were installed on the unisolable piping sections to detect adverse temperature distributions, appropriate temperature limits were established, requirements for periodic review of the temperature instrument values were established, and action limits put into place in the event of exceeding the temperature limit.

   Seabrook Station evaluated the possibility and effects of fluid out-leakage and concluded that unisolable piping sections connected to the Reactor Coolant System at Seabrook Station are not subject to stresses from thermal stratification or temperature oscillations resulting from the mechanism described in NRC Bulletin. This example provides objective evidence that the existing Metal Fatigue of Reactor Coolant Pressure Boundary program is capable of utilizing industry information to determine components and locations subject to thermal and cyclic fatigue

3. To validate that transient cycle design limits are not exceeded, the Seabrook Station Engineering Department tracks and reports cumulative cycles in a quarterly surveillance report. The report, performed under the cycle counting procedure, details components surveyed, transients counted, and their design and report limits. All monitored cycles have been within their limit, with sufficient margin in the 40-year design limits. This example provides objective evidence that the existing Metal Fatigue of Reactor Coolant Pressure Boundary program is capable of monitoring aging effects associated with metal fatigue of reactor coolant system components.

4. To support the 60-year TLAAs associated with metal fatigue of the reactor coolant system pressure boundary components, Seabrook Station analyzed the projected cumulative usage factor, incorporating the environmental fatigue effects for seven (7) NUREG/CR-6260 locations; Reactor Vessel Shell and Lower Head, Reactor Vessel Inlet Nozzle, Reactor Vessel Outlet Nozzle, Surge Line Hot Leg Nozzle, Charging System Nozzle, Safety Injection Nozzle, and the RHR Suction Nozzle. The analyses found the environmentally-adjusted cumulative usage factors will exceed 1.0 for 60 years of service for the surge line hot leg nozzle and the charging nozzle. Enhancements to the program will identify degradation prior to failure. Guidance for evaluation, inspection, repair, and/or replacement is provided for locations where degradation is identified.

   There is sufficient confidence that implementation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program will effectively identify degradation and therefore prevent failure.

**Conclusion**

The Metal Fatigue of Reactor Coolant Pressure Boundary Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B.2.3.2 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS

### Program Description

The Seabrook Station Environmental Qualification (EQ) of Electric Components program is an existing program, implemented through station procedures and preventive maintenance tasks. The EQ program complies with 10 CFR 50.49, *"Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants".* The EQ program manages component thermal, radiation and cyclic aging through the use of 10 CFR 50.49(f) qualification methods. All EQ equipment is included within the scope of License Renewal. Qualified lives are determined for equipment within the scope of the EQ program and appropriate actions such as replacement, refurbishment or re-evaluation are taken prior to the end of the qualified life of the equipment so that the aging limit is not exceeded. Refer to Section 4.4 for a discussion of EQ program reanalysis attributes.

The Environmental Qualification (EQ) of Electric Components program addresses the low voltage I&C cable issues, consistent with those described in the closure of Generic Safety Issue 168 (GSI 168), *"Environmental Qualification of Electrical Equipment".*

### NUREG-1801 Consistency

This program is consistent with NUREG-1801 X.E1.

### Exceptions to NUREG-1801

None

### Enhancements

None

### Operating Experience

The Seabrook Station EQ program is an existing program and has been maintained by on-site Engineering personnel since its inception. Seabrook Station has a comprehensive Operating Experience Program that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Seabrook Station Corrective Action Program is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to Seabrook Station are evaluated. The EQ Coordinator is responsible for reviewing the disposition of such documents as well as subsequent assignment of actions to be taken at Seabrook Station and, confirming that completion of the actions has satisfactorily addressed potential EQ aging issues. The EQ coordinator participates in NUGEQ, the industry EQ group. The EQ coordinator reviews design changes for impact on the EQ program.

EQ System Health Reports are issued quarterly. The parameters monitored within the health reports include EQ regulatory compliance, self-assessments, corrective actions, documentation updates, equipment failures and outage activities. Areas for improvement and conditions requiring action are addressed in action plans.

Data loggers are used in select locations to monitor area temperatures to confirm assumptions and to adjust qualified lives, including both reduction and extension.

The EQ Program documentation underwent a significant review and updating as part of the power uprate project in the 2005-2006 timeframe.

The following are representative condition report samples which indicate that critical aspects of the EQ program are being routinely monitored and evaluated.

a. A condition report addressed extending EQ activities based on actual time of energization of solenoids.

b. A condition report addressed reducing solenoid qualified lives based on temperature monitoring.

c. A condition report addressed documents the failure analysis for an EQ component which failed prior to its designed end of life.

d. A condition report addressed addresses the potential loss of the environmental seal due to the twisting of a transmitter's electronics housing.

e. A condition report addressed addresses potentially different grease being used on EQ fan motor bearings.

f. A condition report addressed addresses the inclusion of a new cable EQ test report into the EQ file.

The operating experience of the EQ program did not show any adverse trend in performance. The problems identified would not cause significant impact to the safe operation of the plant, and adequate corrective actions were taken to prevent recurrence. The key elements of the EQ program are being monitored and effectively implemented. There is sufficient confidence that the implementation of the EQ program will effectively manage the aging of components. Guidance for the re-evaluation, refurbishment or replacement is provided. Periodic self-assessments of the EQ program are performed to identify the areas that need improvement to maintain the performance of the program.

**Conclusion**

The Seabrook Station Environmental Qualification (EQ) of Electric Components Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

# APPENDIX C

# NOT USED

# APPENDIX D

# TECHNICAL SPECIFICATION CHANGES
# (NOT USED)

# APPENDIX E

## ENVIRONMENTAL REPORT FOR SEABROOK STATION

(SEE ADDITIONAL BINDER)