3.5 AGING MANAGEMENT OF CONTAINMENT, STRUCTURES, AND COMPONENT SUPPORTS

3.5.1 INTRODUCTION

Section 3.5 provides the results of the aging management reviews (AMRs) for those structural components and commodities identified in Section 2.4, Scoping and Screening Results - Structures, subject to AMR. The structures or structural commodities are described in the indicated sections.

- Containment (including Containment Vessel, Shield Building, and Containment internal structures) (Section 2.4.1)
- Auxiliary Building (Section 2.4.2)
- Intake Structure, Forebay, and Service Water Discharge Structure (Section 2.4.3)
- Borated Water Storage Tank Level Transmitter Building (Section 2.4.4)
- Miscellaneous Diesel Generator Building (Section 2.4.5)
- Office Building (Condensate Storage Tanks) (Section 2.4.6)
- Personnel Shop Facility Passageway (Missile Shield Area) (Section 2.4.7)
- Service Water Pipe Tunnel and Valve Rooms (Section 2.4.8)
- Station Blackout Diesel Generator Building (including Transformer X-3051 and Radiator Skid Foundations) (Section 2.4.9)
- Turbine Building (Section 2.4.10)
- Water Treatment Building (Section 2.4.11)
- Yard Structures (Section 2.4.12)
- Bulk Commodities (Section 2.4.13)

Table 3.5.1, Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801, provides the summary of the programs evaluated in NUREG-1801 that are applicable to structural component and commodity groups in this section. 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 AMR for Containment, Structures, and Component Supports.

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

The materials from which specific components and commodities are fabricated, the environments to which they are exposed, the aging effects requiring management, and the aging management programs (AMPs) used to manage these aging effects are provided for each of the above structures and structural components in the following sections.

3.5.2.1.1 Containment (including Containment Vessel, Shield Building, and Containment internal structures)

Materials

Containment structural components subject to aging management review are constructed of the following materials:

- Aluminum
- Carbon steel
- Concrete
- Elastomer
- Galvanized steel
- Stainless steel
- Lubrite® sliding surfaces

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Containment structural components subject to aging management review are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Raw water
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

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

- Change in material properties
- Cracking
- Loss of material
- Loss of mechanical function

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Containment structural components:

- 10 CFR Part 50, Appendix J Program
- Boric Acid Corrosion Program
- Cranes and Hoists Inspection Program
- Fire Protection Program
- Inservice Inspection (ISI) Program – IWE
- Inservice Inspection (ISI) Program – IWF
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.2 Auxiliary Building

Materials

Auxiliary Building structural components subject to AMR are constructed of the following materials:

- Aluminum
- Boral®
- Carbon steel
- Concrete

- Concrete blocks
- Galvanized steel
- Stainless steel

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Auxiliary Building structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Raw water
- Soil
- Treated borated water

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

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

- Change in material properties
- Cracking
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Auxiliary Building structural components:

- Boral® Monitoring Program
- Boric Acid Corrosion Program
- Cranes and Hoists Inspection Program
- Fire Protection Program
- Leak Chase Monitoring Program

- Masonry Wall Inspection
- PWR Water Chemistry Program
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.3 Intake Structure, Forebay, and Service Water Discharge Structure

Materials

Intake Structure, Forebay, and Service Water Discharge Structure structural components subject to AMR are constructed of the following materials:

- Carbon steel
- Concrete
- Concrete blocks
- Galvanized steel
- Earthen

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Intake Structure, Forebay, and Service Water Discharge Structure structural components subject to AMR are exposed to the following environments:

- Soil
- Air-indoor
- Air-outdoor
- Water-flowing
- Raw water

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Intake Structure, Forebay, and Service Water Discharge Structure structural components require management:

- Loss of material
- Cracking

- Change in material properties
- Loss of form

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Intake Structure, Forebay, and Service Water Discharge Structure structural components:

- Water Control Structures Inspection
- Fire Protection Program
- Cranes and Hoists Inspection Program
- Masonry Wall Inspection

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.4 Borated Water Storage Tank Level Transmitter Building

Materials

Borated Water Storage Tank Level Transmitter Building structural components subject to AMR are constructed of the following materials:

- Aluminum
- Carbon steel
- Concrete

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Borated Water Storage Tank Level Transmitter Building structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Borated Water Storage Tank Level Transmitter Building structural components require management:

- Change in material properties
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Borated Water Storage Tank Level Transmitter Building structural components:

- Boric Acid Corrosion Program
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.5 Miscellaneous Diesel Generator Building

Materials

Miscellaneous Diesel Generator Building structural components subject to AMR are constructed of the following materials:

- Carbon steel
- Concrete
- Concrete blocks

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Miscellaneous Diesel Generator Building structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Miscellaneous Diesel Generator Building structural components require management:

- Change in material properties
- Cracking
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Miscellaneous Diesel Generator Building structural components:

- Fire Protection Program
- Masonry Wall Inspection
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.6 Office Building (Condensate Storage Tanks)

Materials

Office Building (Condensate Storage Tanks) structural components subject to AMR are constructed of the following materials:

- Aluminum
- Carbon steel
- Concrete
- Concrete blocks
- Porcelain

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Office Building (Condensate Storage Tanks) structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Office Building (Condensate Storage Tanks) structural components require management:

- Change in material properties
- Cracking
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Office Building (Condensate Storage Tanks) structural components:

- Fire Protection Program
- Masonry Wall Inspection
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.7 Personnel Shop Facility Passageway (Missile Shield Area)

Materials

Personnel Shop Facility Passageway (Missile Shield Area) structural components subject to AMR are constructed of the following materials:

- Carbon steel
- Concrete

- Galvanized steel

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Personnel Shop Facility Passageway (Missile Shield Area) structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Personnel Shop Facility Passageway (Missile Shield Area) structural components require management:

- Change in material properties
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following program is credited for managing the effects of aging on the Personnel Shop Facility Passageway (Missile Shield Area) structural components:

- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.8 Service Water Pipe Tunnel and Valve Rooms

Materials

Service Water Pipe Tunnel and Valve Rooms structural components subject to AMR are constructed of the following material:

- Concrete

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Service Water Pipe Tunnel and Valve Rooms structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Raw water
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Service Water Pipe Tunnel and Valve Rooms structural components require management:

- Change in material properties
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Service Water Pipe Tunnel and Valve Rooms structural components:

- Fire Protection Program
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.9 Station Blackout Diesel Generator Building (including Transformer X-3051 and Radiator Skid Foundations)

Materials

Station Blackout Diesel Generator Building (including Transformer X-3051 and Radiator Skid Foundations) structural components subject to AMR are constructed of the following materials:

- Carbon steel
- Concrete
- Concrete blocks

- Galvanized steel

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Station Blackout Diesel Generator Building (including Transformer X-3051 and Radiator Skid Foundations) structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Raw water
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Station Blackout Diesel Generator Building (including Transformer X-3051 and Radiator Skid Foundations) structural components require management:

- Change in material properties
- Cracking
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Station Blackout Diesel Generator Building (including Transformer X-3051 and Radiator Skid Foundations) structural components:

- Masonry Wall Inspection
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.10 Turbine Building

Materials

Turbine Building structural components subject to AMR are constructed of the following materials:

- Carbon steel
- Concrete
- Concrete blocks
- Galvanized steel

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Turbine Building structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Raw water
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

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

- Change in material properties
- Cracking
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Turbine Building structural components:

- Fire Protection Program

- Masonry Wall Inspection
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.11 Water Treatment Building

Materials

Water Treatment Building structural components subject to AMR are constructed of the following materials:

- Carbon steel
- Concrete
- Concrete blocks
- Galvanized steel

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Water Treatment Building structural components subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Raw water
- Soil

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with the Water Treatment Building structural components require management:

- Change in material properties
- Cracking
- Loss of material

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on the Water Treatment Building structural components:

- Fire Protection Program
- Masonry Wall Inspection
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.12 Yard Structures

Materials

Structural components of yard structures subject to AMR are constructed of the following materials:

- Carbon steel
- Concrete
- Concrete blocks
- Earthen
- Galvanized steel

Materials for bulk commodity components are addressed in Section 3.5.2.1.13.

Environments

Structural components of yard structures subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Concrete
- Raw water
- Soil
- Structural backfill

Environments for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Effects Requiring Management

The following aging effects associated with structural components of evaluated yard structures require management:

- Change in material properties
- Cracking
- Loss of material
- Loss of form

Aging effects requiring management for bulk commodity components are addressed in Section 3.5.2.1.13.

Aging Management Programs

The following programs are credited for managing the effects of aging on yard structures' structural components:

- Boric Acid Corrosion Program
- Fire Protection Program
- Masonry Wall Inspection
- Structures Monitoring Program

Aging management programs for bulk commodity components are addressed in Section 3.5.2.1.13.

3.5.2.1.13 Bulk Commodities

Materials

Structural components of bulk commodities subject to AMR are constructed of the following materials:

- Aluminum
- Carbon steel
- Concrete
- Elastomer
- Fire barrier materials (Ceramic fiber/ 3M Interam/ Isolatek/ Mandoseal/ Monokote)
- Galvanized steel

- Insulation materials (Calcium Silicate/ fiberglass/ aluminum jacketing/ stainless steel mirror insulation)
- Stainless steel

Environments

Structural components of bulk commodities subject to AMR are exposed to the following environments:

- Air-indoor
- Air-outdoor
- Soil
- Treated water

Aging Effects Requiring Management

The following aging effects associated with structural components of evaluated bulk commodities require management:

- Change in material properties
- Cracking
- Delamination
- Loss of material
- Separation

Aging Management Programs

The following programs are credited for managing the effects of aging on bulk commodities:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Fire Protection Program
- Structures Monitoring Program
- PWR Water Chemistry Program
- Inservice Inspection (ISI) Program – IWF

3.5.2.2 Aging Management Review Results for Which Further Evaluation is Recommended by NUREG-1801

For the Davis-Besse containment, structures, and component supports, those items requiring further evaluation are addressed in the following sections.

3.5.2.2.1 PWR and BWR Containments

3.5.2.2.1.1 Aging of Inaccessible Concrete Areas

Increases in porosity and permeability, cracking, 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 could occur in inaccessible areas of pressurized water reactor (PWR) and boiling water reactor (BWR) concrete and steel containments.

At Davis-Besse, the Inservice Inspection (ISI) Program – IWL does not apply since the Davis-Besse Containment is a free-standing, steel containment vessel.

Aggressive Chemical Attack and Corrosion of Embedded Steel

The below-grade environment at Davis-Besse is aggressive (Chlorides > 500 ppm and Sulfates > 1,500 ppm). Sampling results indicated a groundwater pH minimum value of 6.9, a chloride content maximum value of 2,870 ppm, and a sulfate content maximum value of 1,700 ppm.

In addition, portions of the containment structures are located below the normal groundwater level. The plant structures have been provided with waterproofing on the exterior portions of the below-grade structures. Water leakage (above and below grade) has been observed at the plant. Once the concrete has cracked, a path is available for water to reach the reinforcing steel, initiating corrosion in rebar that can result in reduced available reinforcing area and spalling of concrete.

Therefore, increases in porosity and permeability, cracking, 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 are applicable for Davis-Besse containment concrete in inaccessible areas.

The Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components. In addition, the Shield Building concrete is managed by the 10 CFR Part 50, Appendix J Program's Containment Vessel and Shield Building Visual Inspection.

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 Structures Monitoring Program*

Cracks and distortion due to increased stress levels from settlement could occur in PWR and BWR concrete and steel containments. Also, reduction of foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundations could occur in all types of PWR and BWR containments.

Settlement

At Davis-Besse, cracking and distortion due to settlement are not aging effects requiring management for containment concrete components because, based on settlement analyses, it is estimated that maximum settlements of Class I structures (e.g., Containment or Shield Building) will be less than 1/8 inch. Therefore, further evaluation of increased stress levels due to settlement is not required.

Porous Concrete Subfoundations

The Davis-Besse Containment does not have a porous concrete subfoundation. Therefore, further evaluation for aging effects due to erosion of porous concrete is not required.

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

Reduction of strength and modulus of concrete due to elevated temperatures could occur in PWR and BWR concrete and steel containments.

Elevated Temperatures

Elevated temperature for the Containment basemat is well below the allowable limits of 150°F general and 200°F local and therefore the aging effect for this mechanism is not applicable. Elevated temperature is an issue of concern in the upper regions of the Containment internal structures. Concrete inside containment and the concrete foundation of the Shield Building that supports containment are evaluated in Section 3.5.2.2.2.3.

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

Loss of material due to general, pitting and crevice corrosion could occur in steel elements of accessible and inaccessible areas for all types of PWR and BWR containments.

Corrosion in inaccessible areas of steel containment liner

At Davis-Besse, loss of material due to corrosion in steel elements of accessible areas is managed by the Inservice Inspection (ISI) Program – IWE, the 10 CFR Part 50, Appendix J Program, the Boric Acid Corrosion Program, and the Structures Monitoring Program.

Information Notice (IN) 2004-09 "Corrosion of Steel Containment and Containment Liner," references the corrosion identified, in the Cycle 13 refueling outage, on the Davis-Besse containment vessel as one of the industry occurrences that led to the issuance of the IN. The IN discussion refers to an amendment to Section 50.55a of Title 10 of the Code of Federal Regulations (10 CFR 50.55a) (61 FR 41303). This amendment requires inservice inspections be performed in accordance with the ASME Code, Section XI, Subsections IWE and IWL. The Davis-Besse containment vessel was subject to a corrosion investigation during the Cycle 13 refueling outage.

The containment vessel corrosion investigation used ultrasonic (UT) thickness measurements of the vessel as one of the investigation methods. The UT measurements verified that the minimum recorded vessel wall thickness (1.404 inches) was greater than the minimum required wall thickness (1.35 inches), as documented in a plant calculation.

The containment vessel is inspected in accordance with the requirements of IWE of the ASME Code Section XI. These inspections include a visual examination of the entire accessible internal surface of the containment vessel every 3-1/3 years as well as visual inspection of the internal moisture barrier at the concrete-to-steel interface. The internal moisture barrier is inspected each refueling outage. The interior and exterior moisture barriers were installed to protect uncoated portions of the vessel and to minimize exposure to water. These inspections exceed the ASME Code Section XI inspection frequency requirements.

The containment vessel area behind the interior concrete structure has been designated as an area susceptible to corrosion and the Augmented Examination requirements of IWE have been imposed. The Augmented Examinations are scheduled to be completed during the Cycle 17 refueling outage.

Loss of material due to corrosion in steel elements of inaccessible areas is managed by the Inservice Inspection (ISI) Program – IWE with Augmented Examination and the 10 CFR Part 50, Appendix J Program.

The continued monitoring of the Containment for loss of material due to general, pitting, and crevice corrosion through the Inservice Inspection (ISI) Program – IWE and the 10 CFR Part 50, Appendix J Program provides reasonable assurance that loss of material in inaccessible areas of Containment is insignificant and will be detected prior to a loss of an intended function.

3.5.2.2.1.5 Loss of Prestress due to Relaxation, Shrinkage, Creep, and Elevated Temperature

Loss of prestress forces due to relaxation, shrinkage, creep, and elevated temperature for PWR prestressed concrete containments and BWR Mark II prestressed concrete containments is a Time- Limited Aging Analysis (TLAA) as defined in 10 CFR 54.3.

Davis-Besse has a free-standing steel containment vessel with no prestressed tendons. The Davis-Besse containment design eliminates loss of prestress forces as an applicable aging effect.

3.5.2.2.1.6 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. Time-limited aging analyses are required to be evaluated in accordance with 10 CFR 54.21(c)(1).

Fatigue TLAAs evaluated for the Davis-Besse Containment are for the containment vessel shell, piping penetrations of the containment vessel, and the permanent reactor cavity seal plate (also known as, permanent canal seal plate (PCSP)). The evaluations of the fatigue TLAAs are addressed in Section 4.

3.5.2.2.1.7 Cracking due to Stress Corrosion Cracking (SCC)

Cracking due to stress corrosion cracking of stainless steel penetration sleeves, penetration bellows, and dissimilar metal welds could occur in all types of PWR and BWR containments. Cracking due to SCC could also occur in stainless steel vent line bellows for BWR containments.

Stress corrosion cracking

Stress corrosion cracking requires a combination of a corrosive environment, susceptible materials, and high tensile stresses. To be susceptible to SCC, stainless steel must be subject to both high temperature (> 140°F) and an aggressive chemical environment. SCC is not an applicable effect for the Davis-Besse stainless steel penetration sleeves and bellows because these stainless steel components are not subject to an aggressive chemical environment.

3.5.2.2.1.8 Cracking due to Cyclic Loading

See Section 3.5.2.2.1.6 that addresses Fatigue TLAAs evaluated for the Davis-Besse containment vessel shell, piping penetrations of the containment vessel, and the permanent reactor cavity seal plate (also known as, permanent canal seal plate (PCSP)).

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

Loss of material (scaling, cracking, and spalling) due to freeze-thaw could occur in PWR and BWR concrete containments.

Freeze-Thaw

Davis-Besse does not have a concrete containment. Davis-Besse has a free-standing steel containment vessel; therefore, loss of material (scaling, cracking, and spalling) from a concrete containment due to freeze-thaw is not applicable to Davis-Besse.

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*

Cracking due to expansion and reaction with aggregate, and increase in porosity and permeability due to leaching of calcium hydroxide could occur in concrete elements of PWR and BWR concrete and steel containments.

Reaction with Aggregate

Davis-Besse design specifications require that concrete aggregates conform to ASTM International (ASTM) Standard Specification C 33 and that the potential reactivity of aggregates be acceptable based on testing in accordance with ASTM Standard Test Method for Potential Alkali-Silica Reactivity of Aggregates (Chemical Method) (ASTM C 289).

Concrete structures and components at Davis-Besse are designed in accordance with American Concrete Institute ACI 318-63 and constructed in accordance with ACI 301-66 using ingredients conforming to ACI and ASTM standards thereby precluding the expansion and reaction with aggregate aging mechanism.

Leaching of Calcium Hydroxide

Change in material properties due to leaching of calcium hydroxide is an aging effect requiring management for concrete components because water leakage (above and below grade) has been observed at Davis-Besse from operating experience.

The Davis-Besse Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components. In addition to aging management by the Structures Monitoring Program, the Shield Building concrete is managed by the 10 CFR Part 50, Appendix J Program's containment vessel and Shield Building Visual Inspection.

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

NUREG-1801 recommends further evaluation of certain structure/aging effect combinations if they are not covered by the structures monitoring program.

The following aging effects (for NUREG-1800 items (1) through (4)) do not require further evaluation because the components are evaluated under the Structures Monitoring Program:

- Corrosion of embedded steel
- Aggressive chemical attack
- Loss of material due to corrosion
- Freeze-Thaw

The Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components. In addition, loss of material due to corrosion is managed by the Boric Acid Corrosion Program within areas that contain borated systems.

(5) Reaction with Aggregate

Davis-Besse design specifications require that concrete aggregates conform to ASTM C 33 and that the potential reactivity of aggregates be acceptable based on testing in accordance with ASTM Standard Test Method for Potential Alkali-Silica Reactivity of Aggregates (Chemical Method) (ASTM C 289).

Concrete structures and components at Davis-Besse are designed in accordance with ACI 318-63 and constructed in accordance with ACI 301-66 using ingredients conforming to ACI and ASTM standards thereby precluding the expansion and reaction with aggregate aging mechanism.

(6) Settlement

Cracking due to settlement is not an aging effect requiring management for concrete components because, based on settlement analyses, it is estimated that maximum settlements of Class I and major Class II structures founded on bedrock (i.e., Containment, Shield Building, Auxiliary Building, Turbine and Office Buildings, Intake Structure, and Valve Room No. 1) will be less than 1/8 inch and that settlements of Class I structures founded on till deposit and granular fill (Borated Water Storage Tank Foundation, SW Pipe Tunnel, and Valve Room No. 2) will be less than 1/4 inch. Therefore, further evaluation of increased stress levels due to settlement is not required.

(7) Porous Concrete Subfoundations

There are no Davis-Besse structures that have a porous concrete subfoundation. Therefore, further evaluation for aging effects due to erosion of porous concrete is not required.

(8) Lock up due to Wear of Sliding Support Surfaces

Lubrite® (plates, bearings, or blocks) is provided to reduce friction for certain support assemblies in Davis-Besse in-scope structural components.

Aging degradation of supports designed with or without sliding connections is managed by the Inservice Inspection (ISI) Program – IWF and the Structures Monitoring Program. Therefore, further evaluation of Lubrite® aging effects is not required.

3.5.2.2.2.2 Aging Management of Inaccessible Areas

3.5.2.2.2.2.1 Below-Grade Inaccessible Concrete Areas – Freeze-Thaw

Loss of material (spalling, scaling) and cracking due to freeze-thaw could occur in below-grade inaccessible concrete areas of Groups 1-3, 5 and 7-9 structures.

Freeze-Thaw

Davis-Besse is located in an area in which weathering conditions are considered severe (weathering index over 500 day-inch/yr).

Concrete structures and components at Davis-Besse are designed in accordance with ACI 318-63 and constructed in accordance with ACI 301-66 using ingredients conforming to ACI and ASTM standards. Concrete constructed to these criteria has a low water-to-cement ratio of less than 0.45 and an air entrainment between 3 and 6% and provides a good quality, dense, low permeability concrete.

Loss of material and cracking due to freeze-thaw are aging effects requiring management for concrete components exposed to weather because isolated instances of freeze-thaw damage have been observed at the plant from operating experience.

As described above, the design and construction of the concrete for Groups 1, 3, and 5 structures are in accordance with ACI Standards that preclude significant loss of material (spalling, scaling) and cracking due to freeze-thaw. The inspection of exposed above-grade concrete of Groups 1, 3 and 5 structures is an indicator for inaccessible concrete and inspection of the above-grade concrete provides reasonable assurance that degradation of inaccessible structures will be detected before a loss of an intended function. Operating experience review has not identified significant loss of material and cracking due to freeze-thaw of below-grade structures concrete.

In the event inspection of above-grade concrete structures identifies significant concrete degradation due to freeze-thaw, corrective actions will be initiated to evaluate the condition of inaccessible portions of structures and determine if excavation of concrete for inspection is warranted.

Therefore, loss of material (spalling, scaling) and cracking due to freeze-thaw are not aging effects requiring management for Davis-Besse below-grade inaccessible concrete components.

However, the Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components, in accordance with NRC position on managing concrete, even though the aging management review did not identify aging effects requiring management. The Structures Monitoring Program will include examination of exposed concrete for age-related degradation when a below-grade in-scope concrete component becomes accessible through excavation.

3.5.2.2.2.2.2 Below-Grade Inaccessible Concrete Areas – Expansion and Reaction with Aggregates

Cracking due to expansion and reaction with aggregates could occur in below-grade inaccessible concrete areas for Groups 1-5 and 7-9 structures.

Reaction with Aggregates

Davis-Besse design specifications require that concrete aggregates conform to ASTM C 33 and that the potential reactivity of aggregates be acceptable based on testing in accordance with ASTM Standard Test Method for Potential Alkali-Silica Reactivity of Aggregates (Chemical Method) (ASTM C 289).

Concrete structures and components at Davis-Besse are designed in accordance with ACI 318-63 and constructed in accordance with ACI 301-66 using ingredients conforming to ACI and ASTM standards thereby precluding the expansion and reaction with aggregate aging mechanism.

Therefore, cracking due to expansion and reaction with aggregates is not an aging effect requiring management for the below-grade inaccessible concrete components.

However, the Structures Monitoring Program is credited for aging management of these mechanisms and effect for the affected concrete structures and structural components, in accordance with the NRC position on managing concrete, even though the aging management review did not identify aging effects requiring management. The Structures Monitoring Program will include examination of exposed concrete for age-related degradation when a below-grade in-scope concrete component becomes accessible through excavation.

3.5.2.2.2.2.3 Below-Grade Inaccessible Concrete Areas – Settlement and Erosion

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 could occur in below-grade inaccessible concrete areas of Groups 1-3, 5 and 7-9 structures.

Settlement

Cracking due to settlement is not an aging effect requiring management for concrete components below grade because based on settlement analyses, it is estimated that maximum settlements of Class I and major Class II structures founded on bedrock (i.e., Containment, Shield Building, Auxiliary Building, Turbine and Office Buildings, Intake Structure, and Valve Room No. 1) will be less than 1/8 inch and that settlements of Class I structures founded on till deposit and granular fill (Borated Water Storage Tank Foundation, SW Pipe Tunnel, and Valve Room No. 2) will be less than 1/4 inch. Therefore, further evaluation for the effects of settlement is not required.

However, the Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components, in accordance with the NRC position on managing concrete, even though the aging management review did not identify aging effects requiring management. The Structures Monitoring Program will include examination of exposed concrete for age-related degradation when a below-grade in-scope concrete component becomes accessible through excavation.

Porous Concrete Subfoundations

Davis-Besse does not have porous concrete subfoundations for Groups 1-3, 5 and 7-9 structures. Therefore, further evaluation for aging effects due to erosion of porous concrete is not required.

3.5.2.2.2.2.4 Below-Grade Inaccessible Concrete Areas – Aggressive Chemical Attack and Corrosion of Embedded Steel

Increase in porosity and permeability, cracking, 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 could occur in below-grade inaccessible concrete areas of Groups 1-3, 5 and 7-9 structures.

Aggressive Chemical Attack and Corrosion of Embedded Steel

At Davis-Besse, concrete components below grade are exposed to an aggressive groundwater environment. In addition, portions of the structures at the plant are located below the normal groundwater level. The plant structures have been provided with waterproofing on the exterior portions of the below-grade structures. Water leakage (above and below grade) has been observed at the plant from operating experience. Once the concrete has cracked, a path is available for water to reach the reinforcing steel, initiating corrosion in rebar that can result in reduced available reinforcing area and spalling of concrete.

Therefore, increase in porosity and permeability, cracking, 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 are aging effects requiring management for the below-grade inaccessible concrete components.

The Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components. Although there is no evidence that the aggressive groundwater has contributed to structural degradation, a special provision in the Structures Monitoring Program will be implemented to monitor below-grade inaccessible concrete components before and during the period of extended operation.

3.5.2.2.2.2.5 Below-Grade Inaccessible Concrete Areas – Leaching of Calcium Hydroxide

Increase in porosity and permeability, and loss of strength due to leaching of calcium hydroxide could occur in below-grade inaccessible concrete areas of Groups 1-3, 5 and 7-9 structures.

Leaching of Calcium Hydroxide

At Davis-Besse, change in material properties due to leaching of calcium hydroxide is an aging effect requiring management for concrete components below grade because water leakage (above and below grade) has been identified in the plant operating experience.

Therefore, increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide are aging effects requiring management for the below-grade inaccessible concrete components.

The Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components.

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

Reduction of strength and modulus of concrete due to elevated temperatures could occur in PWR and BWR Group 1-5 concrete structures, and further evaluation is recommended if any portion of the safety-related and other concrete structures exceeds specified temperature limits, i.e., general area temperature greater than 66°C (150°F) and local area temperature greater than 93°C (200°F).

Elevated Temperatures

Davis-Besse in-scope Group 1, 3, and 5 concrete structures and concrete components are not exposed to temperatures that exceed the limits associated with aging degradation due to elevated temperature. The general air temperatures in safety-related and other structures are maintained below the 150ºF threshold for these aging effects to be applicable.

For the Group 4 structures, several localized areas in the upper regions of the Containment internal structures have maximum temperatures exceeding 150ºF. Only one of those areas exceeded 200ºF. The primary shield wall temperature calculations addressed the effect that a bounding temperature of up to 207ºF would have on the mechanical properties of reinforced concrete and quantified the impact to the upper portion of the primary shield wall during plant operation. The calculations concluded the elevated temperature will not influence the capacity of the primary shield wall to support mechanical loading due to low mechanical stresses in that area. Consistent with NUREG-1801, higher localized temperatures are allowed in the concrete if plant specific calculations are provided. Therefore, the conditions identified in NUREG-1801 are satisfied and loss of material, cracking, and change in material properties due to elevated temperature are not aging effects requiring management for concrete. High temperature piping penetrations contained in the Containment are not in direct contact with concrete and are insulated.

Therefore, reduction of strength and modulus of concrete due to elevated temperatures are not aging effects requiring management for the concrete components at Davis-Besse.

3.5.2.2.2.4 Aging Management of Inaccessible Areas for Group 6 Structures

3.5.2.2.2.4.1 Below-Grade Inaccessible Concrete Areas – Aggressive Chemical Attack and Corrosion of Embedded Steel

Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack; and cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel could occur in below-grade inaccessible concrete areas of Group 6 structures.

Aggressive Chemical Attack and Corrosion of Embedded Steel

Davis-Besse concrete components below grade are exposed to an aggressive groundwater environment. In addition, portions of the structures at the plant are located below the normal groundwater level. The plant structures have been provided with waterproofing on the exterior portions of the below-grade structures. Water leakage (above and below grade) has been observed at the plant from operating experience. Once the concrete has cracked, a path is available for water to reach the reinforcing steel, initiating corrosion in rebar that can result in reduced available reinforcing area and spalling of concrete.

Therefore, increase in porosity and permeability, cracking, 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 are aging effects requiring management for the water control structures' concrete.

The Water Control Structures Inspection is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components.

3.5.2.2.2.4.2 Below-Grade Inaccessible Concrete Areas – Freeze-Thaw

Loss of material (spalling, scaling) and cracking due to freeze-thaw could occur in below-grade inaccessible concrete areas of Group 6 structures.

Freeze-Thaw

At Davis-Besse, loss of material (spalling, scaling) and cracking due to freeze-thaw are aging effects requiring management for concrete components exposed to raw water because the concrete located in water control structures may become saturated and could be susceptible to freeze-thaw.

The Water Control Structures Inspection is credited for aging management of these effects and mechanism for the affected concrete structures and structural components.

3.5.2.2.2.4.3 Below-Grade Inaccessible Concrete Areas – Expansion and Reaction with Aggregate and Leaching of Calcium Hydroxide

Cracking due to expansion and reaction with aggregates and increase in porosity and permeability, and loss of strength due to leaching of calcium hydroxide could occur in below-grade inaccessible reinforced concrete areas of Group 6 structures.

Reaction with Aggregates

Davis-Besse design specifications require that concrete aggregates conform to ASTM C 33 and that the potential reactivity of aggregates be acceptable based on testing in accordance with ASTM Standard Test Method for Potential Alkali-Silica Reactivity of Aggregates (Chemical Method) (ASTM C 289).

Concrete structures and components at Davis-Besse are designed in accordance with ACI 318-63 and constructed in accordance with ACI 301-66 using ingredients conforming to ACI and ASTM standards thereby precluding the expansion and reaction with aggregate aging mechanism.

Therefore, cracking due to expansion and reaction with aggregates is not an aging effect requiring management for the below-grade inaccessible concrete components.

Leaching of Calcium Hydroxide

Change in material properties due to leaching of calcium hydroxide is an aging effect requiring management for concrete components below grade because water leakage (above and below grade) has been observed at the plant from operating experience.

Therefore, increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide are aging effects requiring management for the below-grade inaccessible concrete components.

The Water Control Structures Inspection is credited for aging management of these effects and mechanisms for the affected concrete structures and structural components.

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

Cracking due to stress corrosion cracking and loss of material due to pitting and crevice corrosion could occur for Group 7 and 8 stainless steel tank liners exposed to standing water.

At Davis-Besse, no tanks with stainless steel liners are included in the structural reviews for aging management. Tanks subject to aging management review are evaluated with the respective mechanical systems.

3.5.2.2.2.6 Aging of Supports Not Covered by Structures Monitoring Program

NUREG-1801 recommends further evaluation of certain component support/aging effect combinations if they are not covered by the structures monitoring program.

Each of the following is within the scope of the Structures Monitoring Program. Therefore, further evaluation is not required. In addition, loss of material due to corrosion is managed by the Boric Acid Corrosion Program within areas that contain borated systems.

- Building concrete around support anchorages
- HVAC duct supports
- Instrument supports
- Non-ASME mechanical equipment supports
- Non-ASME supports
- Electrical panels and enclosures

3.5.2.2.2.7 Cumulative Fatigue Damage Due to Cyclic Loading

Fatigue of component support members, anchor bolts, and welds for Groups B1.1, B1.2, and B1.3 component supports is a TLAA as defined in 10 CFR 54.3 only if a current licensing basis (CLB) fatigue analysis exists.

No Davis-Besse CLB fatigue analysis exists for component support members, anchor bolts, or welds for Groups B1.1, B1.2, and B1.3.

3.5.2.2.3 Quality Assurance for Aging Management of Nonsafety-Related Components

See Appendix B, Section B.1.3, for a discussion of FirstEnergy Nuclear Operating Company quality assurance procedures and administrative controls for aging management programs.

3.5.2.3 Time-Limited Aging Analyses

The time-limited aging analyses identified below are associated with the Containment, Structures, and Component Supports commodities. The section of the application that contains the time-limited aging analysis review results is indicated in parentheses.

- Metal Fatigue (Section 4.6, containment vessel shell, piping penetrations of the containment vessel, and the permanent reactor cavity seal plate (also known as, permanent canal seal plate (PCSP))

3.5.3 CONCLUSIONS

The Containment, Structures, and Component Supports subject to AMR have been identified in accordance with the criteria of 10 CFR 54.21. The aging management programs selected to manage the effects of aging on structural components and commodities are identified in the following tables and Section 3.5.2.1. A description of the aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Containment, Structures, and Component Supports will be 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 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
PWR Concrete (Reinforced and Prestressed) and Steel Containments BWR Concrete and Steel (Mark I, II, and III) Containments					
3.5.1-01	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 the environment is non-aggressive. A plant specific program is to be evaluated if environment is aggressive.	Yes, plant-specific, if the environment is aggressive	Not applicable. Aging of concrete containment elements exposed to weather is addressed in Item Numbers 3.5.1-23 and 3.5.1-24. Further evaluation is documented in Section 3.5.2.2.1.1.
3.5.1-02	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	Not applicable. Davis-Besse does not employ a de-watering system for any of the site structures. Cracking and distortion due to settlement are not aging effects requiring management for containment concrete components based on settlement analyses. Further evaluation is documented in Section 3.5.2.2.1.2.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-03	Concrete elements: foundation, sub-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 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	Not applicable. The containment base foundation slabs are not constructed of porous concrete below-grade and are not subject to flowing water, thereby precluding these aging effects and mechanisms. Davis-Besse does not employ a de-watering system for any site structures. Further evaluation is documented in Section 3.5.2.2.1.2.
3.5.1-04	Concrete elements: dome, wall, basemat, ring girder, buttresses, containment, concrete fill-in annulus (as applicable)	Reduction of strength and modulus of concrete due to elevated temperature	A plant-specific aging management program is to be evaluated	Yes, plant-specific if temperature limits are exceeded	Not applicable. Elevated temperature for the Containment basemat is well below the allowable limits of 150°F general and 200°F local and therefore the aging effect for this mechanism is not applicable. Further evaluation is documented in Section 3.5.2.2.1.3.
3.5.1-05	BWR only—not used				

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-06	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. Loss of material is managed by the Inservice Inspection (ISI) Program – IWE with Augmented Examination and the 10 CFR Part 50, Appendix J Programs. Further evaluation is documented in Section 3.5.2.2.1.4.
3.5.1-07	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	Not applicable. This item applies to prestressed concrete containments. Davis-Besse Containment is a PWR steel containment. Refer to Section 3.5.2.2.1.5 for further information.
3.5.1-08	BWR only—not used				
3.5.1-09	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. This item is a TLAA. Further evaluation is documented in Section 3.5.2.2.1.6.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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 effects is to be evaluated	Not applicable. These components are not exposed to an aggressive environment that would support stress corrosion cracking. Further evaluation is documented in Section 3.5.2.2.1.7.
3.5.1-11	BWR only—not used				
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, and supplemented to detect fine cracks	Yes, detection of aging effects is to be evaluated	Not applicable. Cumulative fatigue damage is a TLAA for some components as identified in Item Number 3.5.1-09. Further evaluation is documented in Section 3.5.2.2.1.8.
3.5.1-13	BWR only—not used				

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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	Not applicable. Davis-Besse containment is a PWR steel containment. The Shield Building which completely encloses the steel containment vessel is the only part of the containment structures that is exposed to weather. Further evaluation is documented in Section 3.5.2.2.1.9.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-15	Concrete elements: walls, dome, basemat, ring girder, buttresses, containment, concrete fill-in annulus (as applicable)	Cracking due to expansion and reaction with aggregate; increase in porosity, permeability due to leaching of calcium hydroxide	ISI (IWL) for accessible areas. None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R.	Yes, if concrete was not constructed as stated for inaccessible areas	Not applicable. Cracking due to expansion and reaction with aggregate, and increase in porosity is not an aging effect requiring management due to the quality of concrete used in construction. Davis-Besse Containment is a PWR steel containment. However, change in material properties due to leaching of calcium hydroxide is an aging effect requiring management for the Shield Building concrete. Cracking is managed by the Structures Monitoring Program for the affected concrete structures and structural components. Further evaluation is documented in Section 3.5.2.2.1.10.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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. The subject aging effects are a result of cracking and change in material properties. Seals and gaskets for the Personnel Air Lock, Emergency Air Lock, and Equipment Hatch are evaluated with the host component. See Item Number 3.5.1-17. Cracking and change in material properties which result in loss of sealing and leakage through containment are managed by the Inservice Inspection (ISI) Program – IWE and the 10 CFR Part 50, Appendix J Program.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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. Locks, hinges and closure mechanisms are evaluated with the host component. Loss of leak tightness in closed position of the Personnel Air Lock and the Emergency Air Lock is managed by the Inservice Inspection (ISI) Program – IWE and the 10 CFR Part 50, Appendix J Program. Plant Technical Specifications ensures that access airlocks maintain leak tightness in the closed position.
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. Loss of material for the applicable components is managed by the Inservice Inspection (ISI) Program – IWE and the 10 CFR Part 50, Appendix J Program.
3.5.1-19	BWR only—not used				
3.5.1-20	BWR only—not used				
3.5.1-21	BWR only—not used				

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-22	Prestressed containment: tendons and anchorage components	Loss of material due to corrosion	ISI (IWL)	No	Not applicable. This item applies to prestressed concrete containments. Davis-Besse Containment is a PWR steel containment
Safety-Related and Other Structures; and Component Supports					
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. Cracking and loss of material are managed by the Structures Monitoring Program for the affected concrete structural components. Further evaluation is documented in Section 3.5.2.2.2.1.
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	Consistent with NUREG-1801. Cracking and loss of material are managed by the Structures Monitoring Program for the affected concrete structural components. Further evaluation is documented in Section 3.5.2.2.2.1.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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. Loss of material is managed by the Structures Monitoring Program for the affected steel structural components. Protective coatings are not relied upon to manage the effects of aging. Davis-Besse has provided responses to the NRC regarding Generic Letter 2004-02. Containment coating condition assessment inspections are performed each refueling outage to identify and correct degraded coating materials under the current licensing basis. Containment coatings are subject to ongoing oversight that addresses their current status, which will continue to address their status over the period of extended operation. Further evaluation is documented in Section 3.5.2.2.2.1.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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. Cracking and loss of material are managed by the Structures Monitoring Program for the affected concrete structural components. Further evaluation is documented in Section 3.5.2.2.2.1. The condition of exposed above grade concrete structures are an indicator for inaccessible concrete and provides reasonable assurance that degradation of inaccessible structures will be detected before a loss of an intended function. Operating experience review has not identified significant loss of material and cracking due to freeze-thaw of below-grade structures concrete. Further evaluation is documented in Section 3.5.2.2.2.2.1.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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	Not applicable. Concrete aging is addressed by Item Number 3.5.1-26. In addition, the Structures Monitoring Program is credited for aging management of these effects and mechanisms for the affected concrete structural components, in accordance with the current NRC position, even though the AMR did not identify aging effects requiring management. Further evaluation is documented in Section 3.5.2.2.2.1 and Section 3.5.2.2.2.2.2.
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	Not applicable. The Structures Monitoring Program is used to monitor cracks and distortion. Further evaluation is documented in Section 3.5.2.2.2.1 and Section 3.5.2.2.2.2.3.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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	Not applicable. The concrete foundations at Davis-Besse are not constructed of porous concrete below-grade and are not subject to flowing water, thereby precluding these aging effects and mechanisms. Davis-Besse does not employ a de-watering system for any of the site structures. Further evaluation is documented in Section 3.5.2.2.2.1 and Section 3.5.2.2.2.2.3.
3.5.1-30	Group 4: Radial beam seats in BWR drywell; RPV support shoes for PWR with nozzle supports; Steam generator supports	Lock-up due to wear	ISI (IWF) or Structures monitoring Program	Yes, if not within the scope of ISI or structures monitoring program	Not applicable. Aging degradation of supports designed with sliding connections is addressed in Item Number 3.5.1-53 and is managed by the Inservice Inspection (ISI) Program – IWF and the Structures Monitoring Program. Further evaluation is documented in Section 3.5.2.2.2.1.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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 environment is aggressive.	Yes, plant-specific, if environment is aggressive	Consistent with NUREG-1801. Davis-Besse's area groundwater is aggressive and operating experience has shown that structural elements have experienced degradation. Although there is no evidence that the aggressive groundwater has contributed to structural degradation, a plant-specific provision in the Structures Monitoring Program will be implemented to monitor below-grade inaccessible concrete components before and during the period of extended operation. Further evaluation is documented in Section 3.5.2.2.2.2.4.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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, and 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	Consistent with NUREG-1801. Change in material properties is managed by the Structures Monitoring Program for the affected concrete structural components. Further evaluation is documented in Section 3.5.2.2.2.2.5.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-33	Groups 1-5: concrete	Reduction of strength and modulus of concrete due to elevated temperature	A plant-specific aging management program is to be evaluated.	Yes, plant-specific if temperature limits are exceeded	Not applicable. Group 1, 3, and 5 concrete structures and concrete components are not exposed to temperatures that exceed the limits associated with aging degradation due to elevated temperature. For the Group 4 structures, one area in the upper regions of the Containment internal structures has maximum temperatures exceeding 200°F. Plant-specific calculations have addressed this localized temperature during plant operation. Further evaluation is documented in Section 3.5.2.2.2.3.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-34	Group 6: Concrete; all	Increase in porosity and permeability, cracking, loss of material due to aggressive chemical attack; cracking, loss of bond, loss of material due to corrosion of embedded steel	Inspection of Water-Control Structures or FERC/US Army Corps of Engineers dam inspections and maintenance programs and for inaccessible concrete, an 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 environment is aggressive.	Yes, plant-specific if environment is aggressive	Consistent with NUREG-1801. Cracking and loss of material are managed by the Water Control Structures Inspection for the affected concrete structural components. Further evaluation is documented in Section 3.5.2.2.2.4.1.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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 or FERC/US Army Corps of Engineers dam inspections and maintenance programs. 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	Consistent with NUREG-1801. Cracking and loss of material are managed by the Water Control Structures Inspection for the affected concrete structural components. Further evaluation is documented in Section 3.5.2.2.2.4.2.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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 or FERC/US Army Corps of Engineers dam inspections and maintenance programs. 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	Not applicable. Concrete aging is addressed by Item Numbers 3.5.1-34, 3.5.1-35 and 3.5.1-37. In addition, the Water Control Structures Inspection is credited for aging management of these effects and mechanisms for the affected concrete structural components, in accordance with the current NRC position, even though the AMR did not identify aging effects requiring management. Further evaluation is documented in Section 3.5.2.2.2.4.3.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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 or FERC/US Army Corps of Engineers dam inspections and maintenance programs. 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	Consistent with NUREG-1801. Change in material properties is managed by the Water Control Structures Inspection for the affected concrete structural components. Further evaluation is documented in Section 3.5.2.2.2.4.3.
3.5.1-38	Groups 7, 8: Tank liners	Cracking due to stress corrosion cracking; loss of material due to pitting and crevice corrosion	A plant-specific aging management program is to be evaluated	Yes, plant specific	Not applicable. No tanks with stainless steel liners are included in the structural reviews for aging management. Tanks subject to aging management review are evaluated with the respective mechanical systems. Further evaluation is documented in Section 3.5.2.2.2.5.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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. Loss of material for Groups B2-B5 supports is managed by the Structures Monitoring Program. Further evaluation is documented in Section 3.5.2.2.2.6.
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	Not applicable. The Structures Monitoring Program is credited for aging management of this effect and mechanisms for the affected concrete structural components, in accordance with the current NRC position, even though the AMR did not identify aging effects requiring management. Further evaluation is documented in Section 3.5.2.2.2.6.
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	Not applicable. Davis-Besse has not identified non-metallic vibration isolator elements.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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	Not applicable. A fatigue analysis does not exist in the current licensing basis for the applicable supports. Therefore, no TLAA evaluation is necessary as specified in NUREG-1801. Further evaluation is documented in Section 3.5.2.2.2.7.
3.5.1-43	Groups 1-3, 5, 6: all masonry block walls	Cracking due to restraint shrinkage, creep, and aggressive environment	Masonry Wall Program	No	Consistent with NUREG-1801. Cracking of masonry block walls is managed by the Masonry Wall Inspection. Masonry block walls with a fire barrier intended function are also managed by the Fire Protection Program. The Structures Monitoring Program encompasses and implements the Masonry Wall Inspection.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	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. Cracking and change in material properties for Groups 1-3, 5, 6 elastomeric components are managed by the Structures Monitoring Program, not just group 6. Seals with a fire barrier intended function are managed by the Fire Protection Program. See Item Number 3.3.1-61. NUREG-1801 lists loss of sealing as the aging effect for elastomers. Loss of sealing is not considered an aging effect, but rather a consequence of elastomer degradation. This effect may be caused by cracking and/or change in material properties for elastomeric material.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-45	Group 6: exterior above and below grade concrete foundation; interior slab	Loss of material due to abrasion, cavitation	Inspection of Water-Control Structures or FERC/US Army Corps of Engineers dam inspections and maintenance	No	Not applicable. Concrete aging is addressed by Item Numbers 3.5.1-34, 3.5.1-35 and 3.5.1-37. In addition, the Water Control Structures Inspection is credited for aging management of these effects and mechanisms for the affected concrete structural components, in accordance with the current NRC position, even though the AMR did not identify aging effects requiring management.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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 in accordance with technical specifications and leakage from the leak chase channels.	No	Consistent with NUREG-1801. Cracking due to SCC is not an applicable effect for this item, because, to be susceptible to SCC, stainless steel must be subjected to both high temperature (> 140°F) and an aggressive chemical environment. The stainless steel liner temperature is maintained < 140°F. Loss of material is managed by the PWR Water Chemistry Program. Spent fuel pool water level is maintained in accordance with an existing Technical Specification commitment. The Leak Chase Monitoring Program detects leakage from the leak chase channels during normal operation and refueling.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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 or FERC/US Army Corps of Engineers dam inspections and maintenance. If protective coatings are relied upon to manage aging, protective coating monitoring and maintenance provisions should be included.	No	Consistent with NUREG-1801. Loss of material is managed by the Water Control Structures Inspection.
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 or FERC/US Army Corps of Engineers dam inspections and maintenance programs	No	Consistent with NUREG-1801. Loss of material and loss of form are managed by the Water Control Structures Inspection for the affected concrete and earthen structural components.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-49	Support members; welds; bolted connections; support anchorage to building structure	Loss of material/ general, pitting, and crevice corrosion	Water Chemistry and ISI(IWF)	No	Consistent with NUREG-1801, BWR row with the corresponding PWR programs assigned. Loss of material of structural components exposed to treated water is managed by the Structures Monitoring Program and the PWR Water Chemistry Program. Components are the stainless steel supports in the spent fuel pool which are not within the scope of the Inservice Inspection (ISI) Program – IWF.
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. Loss of material of the listed structural components is managed by the Structures Monitoring Program.
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	Consistent with NUREG-1801. Cracking of the listed structural components is managed by the Bolting Integrity Program. Loss of material is addressed in Item Number 3.5.1-53.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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	Not applicable. Davis-Besse did not identify sliding support surfaces for Groups B2 and B4. Groups B2 and B4 support aging is addressed by Item Numbers 3.5.1-39 and 3.5.1-50.
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. Loss of material of the listed structural components is managed by the Inservice Inspection (ISI) Program – IWF.
3.5.1-54	Groups 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	Not applicable. Davis-Besse addressed aging of these component types in Item Number 3.5.1-53. Aging degradations on Groups B1.1, B1.2, and B1.3 constant and variable load spring hangers; guides; stops are managed by the Inservice Inspection (ISI) Program – IWF. The inspection criteria for supports within the program effectively envelope misalignment and accumulation of debris.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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. Loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion Program.
3.5.1-56	Groups B1.1, B1.2, and B1.3: Sliding surfaces	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. Loss of mechanical function of Groups B1.1, B1.2, and B1.3 supports designed with sliding surfaces are managed by the Inservice Inspection (ISI) Program – IWF.
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	Not applicable. Davis-Besse has not identified non-metallic vibration isolator elements for Groups B1.1, B1.2, and B1.3 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 AEM or AMP	Consistent with NUREG-1801.

Table 3.5.1 Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-59	Stainless steel support members; welds; bolted connections; support anchorage to building structure	None	None	NA - No AEM or AMP	Consistent with NUREG-1801.

Table 3.5.2-1 Aging Management Review Results - Containment									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
1	Containment Emergency Sump Recirculation Valve Enclosure Bellows	EN, SPB, SSR	Stainless Steel	Air-indoor	None	ISI Program-IWE 10 CFR Part 50, Appendix J	N/A	N/A	I 0501 0502
2	Containment Emergency Sump Recirculation Valve Enclosures	EN, SPB, SSR	Stainless Steel	Air-indoor	None	ISI Program-IWE 10 CFR Part 50, Appendix J	N/A	N/A	I 0501 0502
3	Containment Normal Sump Liners	SNS	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
4	Containment Normal Sump Liners	SNS	Stainless Steel	Raw water	Loss of material	Structures Monitoring	N/A	N/A	J 0503
5	Containment Vessel	EN, FLB, HELB, SHD, SPB, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	ISI Program-IWE 10 CFR Part 50, Appendix J	II.A2-9	3.5.1-06	A
6	Containment Vessel	EN, FLB, HELB, SHD, SPB, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-14	3.5.1-55	C 0504
7	Containment Vessel	EN, FLB, HELB, SHD, SPB, SRE, SSR	Carbon Steel	Air-indoor	Cumulative fatigue damage/ fatigue	TLAA	II.A3-4	3.5.1-09	C 0513

Table 3.5.2-1 Aging Management Review Results - Containment									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
8	Containment Vessel Emergency Sump (including sump liner, antivortexing gratings, perforated plates, and trash racks)	DF, SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
9	Cranes, including Bridge, Trolley, Rails, and Girders	SNS, SSR	Carbon Steel	Air-indoor	Loss of material	Cranes and Hoists Inspection	VII.B-3	3.3.1-73	A
10	Cranes, including Bridge, Trolley, Rails, and Girders	SNS, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
11	Emergency Air Lock (including flange gaskets and closure mechanisms)	EN, SPB, SSR	Carbon Steel/ Elastomer	Air-indoor	Loss of material	ISI Program-IWE 10 CFR Part 50, Appendix J Plant Technical Specification	II.A3-6 II.A3-7 II.A3-5 II.A3-6 II.A3-7 II.A3-5	3.5.1-18 3.5.1-16 3.5.1-17 3.5.1-18 3.5.1-16 3.5.1-17	A A A A A A 0505
12	Emergency Air Lock (including flange gaskets and closure mechanisms)	EN, SPB, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
13	Equipment Hatch (including flange gaskets and closure mechanisms)	EN, SPB, SSR	Carbon Steel/ Elastomer	Air-indoor	Loss of material	ISI Program-IWE 10 CFR Part 50, Appendix J	II.A3-6 II.A3-7	3.5.1-18 3.5.1-16	A A
14	Equipment Hatch (including flange gaskets and closure mechanisms)	EN, SPB, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
15	Floor Decking	SNS	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	C
16	Floor Decking	SNS	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
17	LOCA Restraint Rings	SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	C
18	LOCA Restraint Rings	SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
19	LOCA Restraint Ring Cooling Fins	SSR	Stainless Steel	Air-indoor	None	None	III.B1.2-7	3.5.1-59	C
20	Neutron Streaming Shield Panels	SHD, SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	C 0506
21	Neutron Streaming Shield Panels	SHD, SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504 0506

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
22	Nuclear Instrumentation Shielding	SHD, SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	C 0510
23	Nuclear Instrumentation Shielding	SHD, SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504 0510
24	Nuclear Instrumentation Support	SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	C
25	Nuclear Instrumentation Support	SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
26	Nuclear Instrumentation Support	SSR	Aluminum	Air-indoor	None	None	III.B4-4	3.5.1-58	C
27	Nuclear Instrumentation Support	SSR	Aluminum	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
28	Penetration Bellows	EN, SPB, SSR	Stainless Steel	Air-indoor	None	ISI Program-IWE 10 CFR Part 50, Appendix J	N/A	N/A	I 0501
29	Penetrations (Mechanical and Electrical, containment boundary)	EN, SPB, SSR	Carbon Steel/ Elastomer	Air-indoor	Loss of material	ISI Program-IWE 10 CFR Part 50, Appendix J	II.A3-1	3.5.1-18	A 0507

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
30	Penetrations (Mechanical and Electrical, containment boundary)	EN, SPB, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.2-11	3.5.1-55	C 0504 0507
31	Penetrations (Mechanical and Electrical, containment boundary)	EN, SPB, SSR	Stainless Steel	Air-indoor	None	ISI Program-IWE 10 CFR Part 50, Appendix J	N/A	N/A	I 0501
32	Permanent Reactor Cavity Seal Plate	FLB, SSR	Stainless Steel	Air-indoor	None	None	III.B1.2-7	3.5.1-59	C
33	Permanent Reactor Cavity Seal Plate	FLB, SSR	Stainless Steel	Air-indoor	Cumulative fatigue damage/ fatigue	TLAA	II.A3-4	3.5.1-09	C 0514
34	Personnel Air Lock (including gaskets, hatch locks, hinges and closure mechanisms)	EN, SPB, SSR	Carbon Steel / Elastomer	Air-indoor	Loss of material	ISI Program-IWE 10 CFR Part 50, Appendix J Plant Technical Specifications	II.A3-6 II.A3-7 II.A3-5 II.A3-6 II.A3-7 II.A3-5	3.5.1-18 3.5.1-16 3.5.1-17 3.5.1-18 3.5.1-16 3.5.1-17	A A A A A A 0505

Table 3.5.2-1 Aging Management Review Results - Containment									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
35	Personnel Air Lock (including gaskets, hatch locks, hinges and closure mechanisms)	EN, SPB, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
36	Pressurizer Supports	SSR	Carbon Steel	Air-indoor	Loss of material	ISI Program-IWF	III.B1.1-13	3.5.1-53	A
37	Pressurizer Supports	SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-14	3.5.1-55	A 0504
38	Reactor Closure Head and CRD Service Structure	SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	A
39	Reactor Closure Head and CRD Service Structure	SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
40	Reactor Coolant Pressure Boundary Thermal Insulation	SNS	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
41	Reactor Head Storage Stand	SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	A
42	Reactor Head Storage Stand	SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
43	Reactor Head Storage Stand	SNS	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
44	Reactor Shield Wall Liner	SHD, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	A
45	Reactor Shield Wall Liner	SHD, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
46	Reactor Vessel Supports	SSR	Carbon Steel	Air-indoor	Loss of material	ISI Program-IWF	III.B1.1-13	3.5.1-53	A
47	Reactor Vessel Supports	SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-14	3.5.1-55	A 0504
48	Reactor Vessel Thermal Insulation	EN, SNS	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
49	Refueling Canal Fuel Storage Rack	SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
50	Refueling Canal Liner	FLB, SSR	Stainless Steel	Air-indoor	None	Structures Monitoring Boric Acid Corrosion	N/A	N/A	I 0501 0508
51	Station Vent Stack Supports	SNS	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
52	Steam Generator Supports	SSR	Carbon Steel	Air-indoor	Loss of material	ISI Program-IWF	III.B1.1-13	3.5.1-53	A
53	Steam Generator Supports	SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-14	3.5.1-55	A 0504
54	Structural Steel: Beams, Columns, Plates, and Trusses	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A4-5	3.5.1-25	A

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
55	Structural Steel: Beams, Columns, Plates, and Trusses	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
56	Trash Rack Gates	SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
57	Trisodium Phosphate Baskets	SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
58	Containment Normal Sump	SNS	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
59	Containment Vessel Emergency Sump	DF, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
60	Foundations	EN, EXP, FLB, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A1-4	3.5.1-31	A
61	Foundations	EN, EXP, FLB, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A1-5	3.5.1-31	A
62	Foundations	EN, EXP, FLB, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A1-7	3.5.1-32	A 0509
63	Incore Tunnel	DF, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501

Table 3.5.2-1 Aging Management Review Results - Containment									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
64	Primary Shield Wall	EN, MB, SHD, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
65	Reactor Cavity Missile Shield	EN, MB, SHD, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
66	Refueling Canal	SHD, SSR	Concrete	Air-indoor	Loss of material	Structures Monitoring Boric Acid Corrosion	III.A5-9	3.5.1-23	A 0508
67	Refueling Canal	SHD, SSR	Concrete	Air-indoor	Loss of material Change in material properties	Structures Monitoring Boric Acid Corrosion	III.A5-10	3.5.1-24	A 0508
68	Refueling Canal	SHD, SSR	Concrete	Air-indoor	Change in material properties	Structures Monitoring Boric Acid Corrosion	III.A5-7	3.5.1-32	A 0508 0509
69	Reinforced Concrete: Walls, floors, and ceilings	EN, FLB, HELB, MB, PW, SHD, SNS, SPB, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
70	Secondary Shield Wall	EN, HELB, MB, SHD, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
71	Shield Building Emergency Air Lock Enclosure	EN, MB, SSR	Concrete	Air-indoor	None	Structures Monitoring 10 CFR Part 50, Appendix J	N/A	N/A	I 0501 0511

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
72	Shield Building Emergency Air Lock Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-9	3.5.1-23	A 0511
73	Shield Building Emergency Air Lock Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-10	3.5.1-24	A 0511
74	Shield Building Emergency Air Lock Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-6	3.5.1-26	A 0511
75	Shield Building Emergency Air Lock Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-7	3.5.1-32	A 0509 0511
76	Shield Building Dome	EN, MB, SPB, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring 10 CFR Part 50, Appendix J	N/A	N/A	I 0501 0511
77	Shield Building Dome	EN, MB, SPB, SRE, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-9	3.5.1-23	A 0511

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
78	Shield Building Dome	EN, MB, SPB, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-10	3.5.1-24	A 0511
79	Shield Building Dome	EN, MB, SPB, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-6	3.5.1-26	A 0511
80	Shield Building Dome	EN, MB, SPB, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-7	3.5.1-32	A 0509 0511
81	Shield Building Walls (above grade)	EN, FB, MB, SHD, SPB, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring 10 CFR Part 50, Appendix J Fire Protection	N/A	N/A	I 0501 0511 0512
82	Shield Building Walls (above grade)	EN, FB, MB, SHD, SPB, SRE, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring 10 CFR Part 50, Appendix J Fire Protection	III.A1-9	3.5.1-23	A 0511 0512

Table 3.5.2-1 Aging Management Review Results - Containment									
Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
83	Shield Building Walls (above grade)	EN, FB, MB, SHD, SPB, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J Fire Protection	III.A1-10	3.5.1-24	A 0511 0512
84	Shield Building Walls (above grade)	EN, FB, MB, SHD, SPB, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring 10 CFR Part 50, Appendix J Fire Protection	III.A1-6	3.5.1-26	A 0511 0512
85	Shield Building Walls (above grade)	EN, FB, MB, SHD, SPB, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J Fire Protection	III.A1-7	3.5.1-32	A 0509 0511 0512
86	Shield Building Walls (below grade)	EN, FB, SPB, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring 10 CFR Part 50, Appendix J Fire Protection	N/A	N/A	I 0501 0511 0512
87	Shield Building Walls (below grade)	EN, SPB, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-4	3.5.1-31	A 0511

Table 3.5.2-1 Aging Management Review Results - Containment

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
88	Shield Building Walls (below grade)	EN, SPB, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-5	3.5.1-31	A 0511
89	Shield Building Walls (below grade)	EN, SPB, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring 10 CFR Part 50, Appendix J	III.A1-7	3.5.1-32	A 0509 0511
90	Lubrite® sliding supports	SSR	Lubrite®	Air-indoor	Loss of Mechanical Function	ISI Program-IWF	III.B1.1-5 III.B1.2-3	3.5.1-56	A

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
1	Battery Rack	SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
2	Blowoff Roof Vents	EN, PR, SSR	Aluminum	Air-indoor	None	None	III.B4-4	3.5.1-58	C
3	Blowoff Roof Vents	EN, PR, SSR	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
4	Blowout Panels	PR, SSR	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
5	Blowout Panels	PR, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B4-6	3.5.1-55	C 0504
6	Blowout Panels	PR, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
7	Cask Pit Liner	FLB, SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
8	Cask Pit Liner	FLB, SSR	Stainless Steel	Treated borated water	Loss of material	PWR Water Chemistry Leak Chase Monitoring	VII.A2-1	3.3.1-91	C 0521
9	Control Room Ceiling	SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
10	Cranes, including Bridge, Trolley, Rails, and Girders	SNS, SSR	Carbon Steel	Air-indoor	Loss of material	Cranes and Hoists Inspection	VII.B-3	3.3.1-73	A

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
11	Floor Decking	SNS	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
12	Floor Decking	SNS	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B4-6	3.5.1-55	C 0504
13	Fuel Transfer Pit Liner	FLB, SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
14	Fuel Transfer Pit Liner	FLB, SSR	Stainless Steel	Treated borated water	Loss of material	PWR Water Chemistry Leak Chase Monitoring	VII.A2-1	3.3.1-91	C 0521
15	Fuel Transfer Pit Struts	SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
16	Fuel Transfer Tubes	SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
17	Louvered Penthouses	EN, SSR	Aluminum	Air-indoor	None	None	III.B4-4	3.5.1-58	C
18	Louvered Penthouses	EN, SSR	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
19	Masonry Block Wall Bracings and Frames	SNS, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
20	Masonry Block Wall Bracings and Frames	SNS, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
21	New Fuel Storage Racks	SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
22	Roof Decking	SNS	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
23	Roof Decking	SNS	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
24	Shield Panels	SHD, SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A 0515
25	Shield Panels	SHD, SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504 0515
26	Spent Fuel Pool Bulkhead Gates	SSR	Stainless Steel	Air-indoor	None	None	III.B5-5	3.5.1-59	C
27	Spent Fuel Pool Bulkhead Gates	SSR	Stainless Steel	Treated borated water	Loss of material	PWR Water Chemistry	VII.A2-1	3.3.1-91	C
28	Spent Fuel Pool Liner	FLB, SSR	Stainless Steel	Treated borated water	Loss of material	PWR Water Chemistry Spent Fuel Pool water level monitoring per Tech Spec Leak Chase Monitoring	III.A5-13	3.5.1-46	A 0516 0521
29	Spent Fuel Storage Racks	SSR	Stainless Steel	Treated borated water	Loss of material	PWR Water Chemistry	VII.A2-1	3.3.1-91	C
30	Station Vent Stack	RP, SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
31	Station Vent Stack	RP, SNS	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
32	Structural Steel: Beams, Columns, Plates, and Trusses	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
33	Structural Steel: Beams, Columns, Plates, and Trusses	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
34	Auxiliary Building Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
35	Auxiliary Building Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
36	Auxiliary Building Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
37	Auxiliary Building Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
38	Auxiliary Building Exterior Walls (below grade)	EN, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-2			Aging Management Review Results - Auxiliary Building						
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
39	Auxiliary Building Exterior Walls (below grade)	EN, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
40	Auxiliary Building Exterior Walls (below grade)	EN, FLB, SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
41	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
42	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
43	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
44	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
45	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
46	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
47	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
48	Auxiliary Feedpump Turbine Exhaust	EN, MB, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
49	Cask Pit	SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
50	Foundations	EN, EXP, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
51	Foundations	EN, EXP, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
52	Foundations	EN, EXP, FLB, SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
53	Fuel Transfer Pit	SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
54	Masonry Block Walls	EN, FB, FLB, SHD, SNS, SRE, SSR	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection Fire Protection	III.A3-11	3.5.1-43	A 0515 0517

Table 3.5.2-2	Aging Management Review Results - Auxiliary Building								
Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
55	Missile Shield Walls	MB, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
56	Missile Shield Walls	MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
57	Missile Shield Walls	MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
58	Missile Shield Walls	MB, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
59	Missile Shield Walls	MB, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
60	Missile Shield Walls	MB, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
61	Missile Shield Walls	MB, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
62	New Fuel Storage Pit	EN, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
63	Pipe Tunnel	EN, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
64	Pipe Tunnel	EN, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
65	Pipe Tunnel	EN, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
66	Pipe Tunnel	EN, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
67	Reinforced Concrete: Walls, floors, and ceilings	EN, FB, FLB, HELB, MB, PW, SHD, SNS, SPB, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring Fire Protection	N/A	N/A	I 0501 0512 0515
68	Roof Penthouses	EN, MB, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
69	Roof Penthouses	EN, MB, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
70	Roof Penthouses	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
71	Roof Penthouses	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
72	Roof Penthouses	EN, MB, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
73	Roof Slabs	EN, MB, SNS, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501 0518
74	Spent Fuel Pool	SHD, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
75	Sump	SNS	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
76	Sump	SNS	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0519
77	Sump	SNS	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0519
78	Sump	SNS	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0519
79	Spent Fuel Rack Neutron Absorbers	ABN, SSR	Boral®	Treated borated water	Loss of material	Boral® Monitoring PWR Water Chemistry	VII.A2-5	3.3.1-13	J 0520

Table 3.5.2-2 Aging Management Review Results - Auxiliary Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-3 Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
1	Battery Rack	SRE	Carbon Steel	Air-indoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
2	Cranes, including Bridge, Trolley, Rails, and Girders	SNS	Carbon Steel	Air-outdoor	Loss of material	Cranes and Hoists Inspection	VII.B-3	3.3.1-37	A 0529
3	Louvered Penthouse	EN, SSR	Galvanized Steel	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
4	Metal Siding	SNS, SRE	Carbon Steel	Air-indoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
5	Metal Siding	SNS, SRE	Carbon Steel	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
6	Roof Decking	SNS, SRE	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	C
7	Roof Decking	SNS, SRE	Galvanized Steel	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
8	Sheet Pilings (includes Support Braces and Rock Anchors)	FLB, SNS, SSR	Carbon Steel	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
9	Sheet Pilings (includes Support Braces and Rock Anchors)	FLB, SNS, SSR	Carbon Steel	Water-flowing	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522 0528

Table 3.5.2-3 Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
10	Structural Steel: Beams, Columns, Plates, and Trusses	SNS, SRE	Carbon Steel	Air-indoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
11	Trash Rack Guides	SNS	Galvanized Steel	Water-flowing	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
12	Trash Racks	SNS	Carbon steel	Water-flowing	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
13	Traveling Screen Casing and Associated Framing	SNS	Carbon Steel	Air-indoor	Loss of material	Water Control Structures Inspection	III.A6-11	3.5.1-47	B 0522
14	Fan Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522
15	Fan Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0525
16	Fan Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522
17	Fan Enclosure	EN, MB, SSR	Concrete	Air-outdoor	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509

Table 3.5.2-3 Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
18	Forebay Retaining Walls	FLB, SSR	Concrete	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522
19	Forebay Retaining Walls	FLB, SSR	Concrete	Soil	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522 0524
20	Forebay Retaining Walls	FLB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0525
21	Forebay Retaining Walls	FLB, SSR	Concrete	Soil	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522
22	Forebay Retaining Walls	FLB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522
23	Forebay Retaining Walls	FLB, SSR	Concrete	Water-flowing	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522 0523
24	Forebay Retaining Walls	FLB, SSR	Concrete	Air-outdoor	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509

Table 3.5.2-3 Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
25	Forebay Retaining Walls	FLB, SSR	Concrete	Soil	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509
26	Foundations	EN, EXP, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522 0524
27	Foundations	EN, EXP, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522
28	Foundations	EN, EXP, FLB, SNS, SRE, SSR	Concrete	Soil	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509
29	Intake Structure Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522
30	Intake Structure Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0525
31	Intake Structure Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522

Table 3.5.2-3	Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure								
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
32	Intake Structure Exterior Walls (above grade)	EN, FLB, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509
33	Intake Structure Exterior Walls (below grade)	EN, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522 0524
34	Intake Structure Exterior Walls (below grade)	EN, FLB, SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522
35	Intake Structure Exterior Walls (below grade)	EN, FLB, SNS, SRE, SSR	Concrete	Soil	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509
36	Louvered Penthouse	EN, MB, SSR	Concrete	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522
37	Louvered Penthouse	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0525
38	Louvered Penthouse	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522
39	Louvered Penthouse	EN, MB, SSR	Concrete	Air-outdoor	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509

Table 3.5.2-3 Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
40	Masonry Block Walls	EN, FB, FLB, SRE, SSR	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection Fire Protection	III.A6-10	3.5.1-43	A 0517
41	Pump Intake Cells	HS, SRE, SSR	Concrete	Water-flowing	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522 0523
42	Reinforced Concrete: Walls, floors, and ceilings	EN, FB, FLB, MB, SNS, SRE, SSR	Concrete	Air-indoor	None	Water Control Structures Inspection Fire Protection	N/A	N/A	I 0501 0522 0512
43	Roof Slabs	EN, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522
44	Roof Slabs	EN, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0525
45	Roof Slabs	EN, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522
46	Roof Slabs	EN, MB, SNS, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509

Table 3.5.2-3 Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
47	Service Water Discharge Pipe Sleeve	EN, SSR	Concrete	Soil	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0526
48	Service Water Discharge Pipe Sleeve	EN, SSR	Concrete	Soil	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509 0526
49	Service Water Discharge Structure	EN, SSR	Concrete	Soil	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0526
50	Service Water Discharge Structure	EN, SSR	Concrete	Soil	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522 0526
51	Service Water Discharge Structure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material	Water Control Structures Inspection	III.A6-1	3.5.1-34	B 0522 0526
52	Service Water Discharge Structure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0525

Table 3.5.2-3 Aging Management Review Results – Intake Structure, Forebay, and Service Water Discharge Structure

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
53	Service Water Discharge Structure	EN, MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Water Control Structures Inspection	III.A6-5	3.5.1-35	B 0522
54	Service Water Discharge Structure	EN, MB, SSR	Concrete	Air-outdoor	Change in material properties	Water Control Structures Inspection	III.A6-6	3.5.1-37	B 0522 0509
55	Sump	SNS	Concrete	Air-indoor	None	Water Control Structures Inspection	N/A	N/A	I 0501 0522
56	Sump	SNS	Concrete	Raw Water	Loss of material Change in material properties	Water Control Structures Inspection	III.A6-3	3.5.1-34	B 0522 0527
57	Forebay (including riprap)	HS, SRE, SSR	Earthen	Air-outdoor	Loss of material Loss of Form	Water Control Structures Inspection	N/A	N/A	G
58	Forebay (including riprap)	HS, SRE, SSR	Earthen	Water-flowing	Loss of material Loss of Form	Water Control Structures Inspection	III.A6-9	3.5.1-48	B 0522

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-4 Aging Management Review Results – Borated Water Storage Tank Level Transmitter Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
1	Metal Roof	EN, SNS	Aluminized Steel (Aluminum)	Air-indoor	None	None	III.B4-4	3.5.1-58	C
2	Metal Roof	EN, SNS	Aluminized Steel (Aluminum)	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
3	Metal Roof	EN, SNS	Aluminized Steel (Aluminum)	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
4	Metal Siding	EN, SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
5	Metal Siding	EN, SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
6	Metal Siding	EN, SNS	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
7	Metal Siding	EN, SNS	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
8	Structural Steel: Beams, Columns, Plates, and Trusses	SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
9	Structural Steel: Beams, Columns, Plates, and Trusses	SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504

Table 3.5.2-4 Aging Management Review Results – Borated Water Storage Tank Level Transmitter Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
10	Foundation Piers	SNS	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
11	Foundation Piers	SNS	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
12	Foundation Piers	SNS	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-5 Aging Management Review Results – Miscellaneous Diesel Generator Building									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
1	Structural Steel: Beams, Columns, Plates, and Trusses	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
2	Exterior Walls (above grade)	SRE	Concrete Blocks	Air-outdoor	Cracking	Masonry Wall Inspection	III.A3-11	3.5.1-43	A
3	Foundations	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
4	Foundations	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
5	Foundations	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
6	Masonry Block Walls	FB, SRE	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection Fire Protection	III.A3-11	3.5.1-43	A 0517
7	Reinforced Concrete: Walls, floors, and ceilings	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
8	Roof	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A

Table 3.5.2-5 Aging Management Review Results – Miscellaneous Diesel Generator Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
9	Roof	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
10	Roof	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
11	Roof	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-6 Aging Management Review Results – Office Building (Condensate Storage Tanks)									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
1	Structural Steel: Beams, Columns, Plates, and Trusses	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
2	Wall Panel Support Frames	SRE	Aluminum	Air-indoor	None	None	III.B4-4	3.5.1-58	C
3	Wall Panel Support Frames	SRE	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
4	Condensate Storage Tanks Foundation	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
5	Exterior Walls (above grade)	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
6	Exterior Walls (above grade)	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
7	Exterior Walls (above grade)	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
8	Exterior Walls (above grade)	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A

Table 3.5.2-6 Aging Management Review Results – Office Building (Condensate Storage Tanks)

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
9	Foundations (including caissons)	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
10	Foundations (including caissons)	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
11	Foundations (including caissons)	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
12	Masonry Block Walls	FB, SRE	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection Fire Protection	III.A3-11	3.5.1-43	A 0517
13	Reinforced Concrete: Walls and floors	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
14	Reinforced Concrete: Ceilings	FB, SRE	Concrete	Air-indoor	None	Structures Monitoring Fire Protection	N/A	N/A	I 0501
15	Window Wall Panels	SRE	Porcelain	Air-indoor	None	None	N/A	N/A	I 0549
16	Window Wall Panels	SRE	Porcelain	Air-outdoor	None	None	N/A	N/A	I 0549

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-7 Aging Management Review Results – Personnel Shop Facility Passageway (Missile Shield Area)

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
1	Metal Floor Deck	SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
2	Metal Roof Decking	SNS	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
3	Metal Siding	SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
4	Metal Siding	SNS	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
5	Structural Steel: Beams, Columns, Plates, and Trusses	SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
6	Exterior Walls (above grade)	MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
7	Exterior Walls (above grade)	MB, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
8	Exterior Walls (above grade)	MB, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A

Table 3.5.2-7 Aging Management Review Results – Personnel Shop Facility Passageway (Missile Shield Area)

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
9	Exterior Walls (above grade)	MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
10	Foundations	SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
11	Foundations	SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
12	Foundations	SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
13	Roof	MB, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501 0518
14	Reinforced Concrete: Walls, floors, and ceilings	MB, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-8 Aging Management Review Results – Service Water Pipe Tunnel and Valve Rooms

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
1	Foundations	SNS, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
2	Foundations	SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
3	Foundations	SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
4	Reinforced Concrete: Walls, floors, and ceilings	EN, FB, FLB, MB, SNS, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring Fire Protection	N/A	N/A	I 0501 0512
5	Reinforced Concrete: Walls, floors, and ceilings	EN, FLB, MB, SNS, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
6	Reinforced Concrete: Walls, floors, and ceilings	EN, FLB, MB, SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A

Table 3.5.2-8 Aging Management Review Results – Service Water Pipe Tunnel and Valve Rooms

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
7	Reinforced Concrete: Walls, floors, and ceilings	EN, FLB, MB, SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
8	Sumps	SNS	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
9	Sumps	SNS	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0530
10	Sumps	SNS	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0530
11	Sumps	SNS	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0530

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-9 Aging Management Review Results – Station Blackout Diesel Generator Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
1	Battery Rack	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
2	Metal Roof	SRE	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
3	Metal Roof	SRE	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
4	Metal Siding	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
5	Metal Siding	SRE	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
6	Structural Steel: Beams, Columns, Plates, and Trusses	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
7	Foundations	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
8	Foundations	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
9	Foundations	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509

Table 3.5.2-9 Aging Management Review Results – Station Blackout Diesel Generator Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
10	Masonry Block Walls	SRE	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection	III.A3-11	3.5.1-43	A
11	Radiator Skid Foundation	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
12	Radiator Skid Foundation	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
13	Radiator Skid Foundation	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
14	Radiator Skid Foundation	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
15	Radiator Skid Foundation	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
16	Radiator Skid Foundation	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
17	Radiator Skid Foundation	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509

Table 3.5.2-9 Aging Management Review Results – Station Blackout Diesel Generator Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
18	Reinforced Concrete: Floors and ceilings	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
19	Sumps	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
20	Sumps	SRE	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0530
21	Sumps	SRE	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0530
22	Sumps	SRE	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0530
23	Transformer Foundation	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
24	Transformer Foundation	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
25	Transformer Foundation	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A

Table 3.5.2-9 Aging Management Review Results – Station Blackout Diesel Generator Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
26	Transformer Foundation	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
27	Transformer Foundation	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
28	Transformer Foundation	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
29	Transformer Foundation	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-10 Aging Management Review Results – Turbine Building									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
1	Metal Roof Decking	EN, SNS, SRE	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
2	Metal Siding	EN, SNS, SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
3	Metal Siding	EN, SNS, SRE	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
4	Structural Steel: Beams, Columns, Plates, and Trusses	SNS, SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
5	Foundations	EN, EXP, FLB, SNS, SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
6	Foundations	EN, EXP, FLB, SNS, SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
7	Foundations	EN, EXP, FLB, SNS, SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
8	Masonry Block Walls	FB, SRE	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection Fire Protection	III.A3-11	3.5.1-43	A 0517

Table 3.5.2-10 Aging Management Review Results – Turbine Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
9	Reinforced Concrete: Walls, floors, and ceilings	EN, FB, SNS, SRE	Concrete	Air-indoor	None	Structures Monitoring Fire Protection	N/A	N/A	I 0501 0512
10	Sumps	SNS	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
11	Sumps	SNS	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0530
12	Sumps	SNS	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0530
13	Sumps	SNS	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0530
14	Turbine Generator Pedestal	SNS	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-11 Aging Management Review Results – Water Treatment Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
1	Metal Roof Decking	SRE	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
2	Metal Siding	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
3	Metal Siding	SRE	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
4	Structural Steel: Beams, Columns, Plates, and Trusses	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
5	Foundations	EXP, SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
6	Foundations	EXP, SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
7	Foundations	EXP, SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
8	Masonry Block Walls	FB, SRE	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection Fire Protection	III.A3-11	3.5.1-43	A 0517

Table 3.5.2-11 Aging Management Review Results – Water Treatment Building

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
9	Reinforced Concrete: Walls, floors, and ceilings	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
10	Sumps	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
11	Sumps	SRE	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0530
12	Sumps	SRE	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0530
13	Sumps	SRE	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0530

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-12 Aging Management Review Results – Yard Structures									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
1	BWST Pipe Trench Cover Plates	EN, SNS	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	C
2	BWST Pipe Trench Cover Plates	EN, SNS	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
3	BWST Pipe Trench Cover Plates	EN, SNS	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
4	BWST Pipe Trench Cover Plates	EN, SNS	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
5	Cable Trench Cover Plates	SRE	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	C
6	Cable Trench Cover Plates	SRE	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
7	EDG Fuel Oil Storage Tank Hold Down Restraints	SSR	Carbon Steel	Concrete	None	None	VII.J-21	3.3.1-96	C
8	EDG Fuel Oil Storage Tank Hold Down Restraints	SSR	Carbon Steel	Structural backfill	None	Structures Monitoring	N/A	N/A	H 0531
9	Fire Hydrant Hose Houses	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
10	Fire Hydrant Hose Houses	SRE	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
11	Manhole Covers and Frames	EN, SNS, SRE	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
12	Metal Roof Decking (Nitrogen Storage Building)	SNS	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C 0536
13	SBO Component Support Structures	SRE	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
14	SBO Component Support Structures	SRE	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
15	Structural Steel: Beams, Columns, Plates, and Trusses (BWST trench cover support)	SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
16	Structural Steel: Beams, Columns, Plates, and Trusses (BWST trench cover support)	SNS	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
17	Structural Steel: Beams, Columns, Plates, and Trusses (Diesel Oil Pump House)	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
18	Structural Steel: Beams, Columns, Plates, and Trusses (Nitrogen Storage Building)	SNS	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A 0536
19	Structural Steel: Beams, Columns, Plates, and Trusses (Relay House)	SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.A3-12	3.5.1-25	A
20	Wave Protection Dike Corrugated Pipe Casings	EN, SNS	Galvanized Steel	Structural backfill	Loss of material	Structures Monitoring	N/A	N/A	H 0532
21	Wave Protection Dike Piles	SNS	Carbon Steel	Structural backfill	Loss of material	Structures Monitoring	N/A	N/A	H 0532
22	BWST Foundation	EN, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
23	BWST Foundation	EN, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
24	BWST Foundation	EN, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
25	BWST Foundation	EN, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
26	BWST Foundation	EN, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
27	BWST Foundation	EN, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
28	BWST Foundation	EN, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
29	BWST Pipe Trench	EN, SNS, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
30	BWST Pipe Trench	EN, SNS, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
31	BWST Pipe Trench	EN, SNS, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
32	BWST Pipe Trench	EN, SNS, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
33	BWST Pipe Trench	EN, SNS, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
34	BWST Pipe Trench	EN, SNS, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
35	BWST Pipe Trench	EN, SNS, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
36	BWST Pipe Trench	EN, SNS, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
37	BWST Pipe Trench Hatch Covers	EN, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
38	BWST Pipe Trench Hatch Covers	EN, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
39	BWST Pipe Trench Hatch Covers	EN, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
40	BWST Pipe Trench Hatch Covers	EN, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
41	BWST Pipe Trench Hatch Covers	EN, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
42	Cable Trench Top Slabs	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
43	Cable Trench Top Slabs	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
44	Cable Trench Top Slabs	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
45	Cable Trench Top Slabs	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
46	Cable Trench Top Slabs	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
47	Cable Trenches	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501

Table 3.5.2-12 Aging Management Review Results – Yard Structures									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
48	Cable Trenches	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
49	Cable Trenches	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
50	Cable Trenches	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
51	Diesel Oil Pump House Foundation	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
52	Diesel Oil Pump House Foundation	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
53	Diesel Oil Pump House Foundation	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
54	Diesel Oil Storage Tank Foundation	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
55	Diesel Oil Storage Tank Foundation	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
56	Diesel Oil Storage Tank Foundation	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
57	Diesel Oil Storage Tank Foundation	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
58	Diesel Oil Storage Tank Foundation	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
59	Diesel Oil Storage Tank Foundation	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
60	Diesel Oil Storage Tank Foundation	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
61	Diesel Oil Storage Tank Retaining Area and Dike	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
62	Diesel Oil Storage Tank Retaining Area and Dike	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
63	Diesel Oil Storage Tank Retaining Area and Dike	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
64	Diesel Oil Storage Tank Retaining Area and Dike	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
65	Diesel Oil Storage Tank Retaining Area and Dike	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
66	Diesel Oil Storage Tank Retaining Area and Dike	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
67	Diesel Oil Storage Tank Retaining Area and Dike	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
68	Duct Banks	EN, SNS, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
69	Duct Banks	EN, SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
70	Duct Banks	EN, SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
71	EDG Fuel Oil Storage Tanks Foundation (tank manhole)	SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
72	EDG Fuel Oil Storage Tanks Foundation (tank manhole)	SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
73	EDG Fuel Oil Storage Tanks Foundation (tank manhole)	SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
74	EDG Fuel Oil Storage Tanks Foundation (tank manhole)	SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
75	EDG Fuel Oil Storage Tanks Foundation	SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
76	EDG Fuel Oil Storage Tanks Foundation	SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
77	EDG Fuel Oil Storage Tanks Foundation	SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
78	Fire Hydrant Hose House Foundations	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
79	Fire Hydrant Hose House Foundations	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
80	Fire Hydrant Hose House Foundations	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
81	Fire Hydrant Hose House Foundations	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
82	Fire Hydrant Hose House Foundations	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
83	Fire Hydrant Hose House Foundations	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
84	Fire Hydrant Hose House Foundations	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
85	Fire Water Piping Thrust Blocks	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	C
86	Fire Water Piping Thrust Blocks	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	C
87	Fire Water Piping Thrust Blocks	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	C 0509
88	Fire Walls (transformers)	FB, SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring Fire Protection	III.A3-6	3.5.1-26	A 0533
89	Fire Walls (transformers)	FB, SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring Fire Protection	III.A3-7	3.5.1-32	A 0509 0533
90	Fire Walls (transformers)	FB, SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring Fire Protection	III.A3-9	3.5.1-23	A 0533

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
91	Fire Walls (transformers)	FB, SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring Fire Protection	III.A3-10	3.5.1-24	A 0533
92	Fire Water Storage Tank Foundation	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
93	Fire Water Storage Tank Foundation	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
94	Fire Water Storage Tank Foundation	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
95	Fire Water Storage Tank Foundation	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
96	Fire Water Storage Tank Foundation	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
97	Fire Water Storage Tank Foundation	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
98	Fire Water Storage Tank Foundation	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
99	Manhole Missile Shields	MB, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
100	Manhole Missile Shields	MB, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
101	Manhole Missile Shields	MB, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
102	Manhole Missile Shields	MB, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
103	Manhole Missile Shields	MB, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
104	Manholes	EN, SNS, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
105	Manholes	EN, SNS, SRE, SSR	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
106	Manholes	EN, SNS, SRE, SSR	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
107	Manholes	EN, SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
108	Masonry Block Walls (Relay House)	SRE	Concrete Blocks	Air-indoor	Cracking	Masonry Wall Inspection	III.A3-11	3.5.1-43	A
109	Nitrogen Storage Building Foundation	SNS	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
110	Nitrogen Storage Building Foundation	SNS	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
111	Nitrogen Storage Building Foundation	SNS	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
112	Nitrogen Storage Building Foundation	SNS	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
113	Nitrogen Storage Building Foundation	SNS	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
114	Nitrogen Storage Building Foundation	SNS	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
115	Nitrogen Storage Building Foundation	SNS	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
116	Precast Panels (Relay House)	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
117	Precast Panels (Relay House)	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
118	Precast Panels (Relay House)	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
119	Precast Panels (Relay House)	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
120	Reinforced Concrete: Walls, Floors, and Ceilings (Diesel Oil Pump House)	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
121	Reinforced Concrete: Walls, Floors, and Ceilings (Diesel Oil Pump House)	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
122	Reinforced Concrete: Walls, Floors, and Ceilings (Diesel Oil Pump House)	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
123	Reinforced Concrete: Walls, Floors, and Ceilings (Diesel Oil Pump House)	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
124	Reinforced Concrete: Walls, Floors, and Ceilings (Diesel Oil Pump House)	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
125	Reinforced Concrete: Walls, Floors, and Ceilings (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A 0536
126	Reinforced Concrete: Walls, Floors, and Ceilings (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
127	Reinforced Concrete: Walls, Floors, and Ceilings (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
128	Reinforced Concrete: Walls, Floors, and Ceilings (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
129	Reinforced Concrete: Walls, Floors, and Ceilings (Relay House)	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
130	Relay House Foundation	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
131	Relay House Foundation	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
132	Relay House Foundation	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
133	Roof (Diesel Oil Pump House)	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501 0518
134	Roof (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
135	Roof (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
136	Roof (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
137	Roof (Nitrogen Storage Building)	MB, SNS	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
138	Roof (Relay House)	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501 0518

Table 3.5.2-12 Aging Management Review Results – Yard Structures									
Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
139	SBO Component Foundations	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
140	SBO Component Foundations	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
141	SBO Component Foundations	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
142	SBO Component Foundations	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
143	SBO Component Foundations	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
144	SBO Component Foundations	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
145	SBO Component Foundations	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
146	Sumps (Diesel Oil Pump House)	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
147	Sumps (Diesel Oil Storage Tank Retaining Area)	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
148	Sumps (Diesel Oil Pump House and Diesel Oil Storage Tank Retaining Area)	SRE	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0534
149	Sumps (Diesel Oil Pump House and Diesel Oil Storage Tank Retaining Area)	SRE	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0534
150	Sumps (Diesel Oil Pump House and Diesel Oil Storage Tank Retaining Area)	SRE	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0534
151	Sumps (Manholes)	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A 0535
152	Sumps (Manholes)	SRE	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0534

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
153	Sumps (Manholes)	SRE	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0534
154	Sumps (Manholes)	SRE	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0534
155	Sumps (Relay House)	SRE	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
156	Sumps (Relay House)	SRE	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0534
157	Sumps (Relay House)	SRE	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0534
158	Sumps (Relay House)	SRE	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0534
159	Sumps (Transformer Foundations)	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A 0535
160	Sumps (Transformer Foundations)	SRE	Concrete	Raw water	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A 0534

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
161	Sumps (Transformer Foundations)	SRE	Concrete	Raw water	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0534
162	Sumps (Transformer Foundations)	SRE	Concrete	Raw water	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A 0534
163	Transformer Foundations	SRE	Concrete	Soil	Loss of material	Structures Monitoring	III.A3-4	3.5.1-31	A
164	Transformer Foundations	SRE	Concrete	Soil	Loss of material Change in material properties	Structures Monitoring	III.A3-5	3.5.1-31	A
165	Transformer Foundations	SRE	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
166	Transformer Foundations	SRE	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
167	Transformer Foundations	SRE	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
168	Transformer Foundations	SRE	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A

Table 3.5.2-12 Aging Management Review Results – Yard Structures

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
169	Transformer Foundations	SRE	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A
170	Wave Protection Dikes (including riprap)	FLB, SNS	Earthen	Air-outdoor	Loss of form	Structures Monitoring	N/A	N/A	G
171	EDG Fuel Oil Storage Tanks Backfill	EN, MB, SSR	Earthen	Air-outdoor	Loss of form	Structures Monitoring	N/A	N/A	G

1 Refer to Table 2.0-1 for intended function descriptions.

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
Steel and Other Metals									
1	Anchorage / Embedments	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A
2	Anchorage / Embedments	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-14 III.B1.2-11 III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504
3	Anchorage / Embedments	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-8 III.B1.2-6 III.B1.3-6 III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504
4	Anchorage / Embedments	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8 III.B3-5 III.B4-8 III.B5-5	3.5.1-59	A

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
5	Anchorage / Embedments	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A
6	Anchorage / Embedments	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B1.2-11 III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504
7	Anchorage / Embedments	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A
8	Anchorage / Embedments	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B1.2-6 III.B1.3-6 III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504
9	Anchorage / Embedments	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A
10	Cable Tray and Conduit Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
11	Cable Tray and Conduit Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
12	Cable Tray and Conduit Supports	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
13	Cable Tray and Conduit Supports	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8	3.5.1-59	A
14	Cable Tray and Conduit Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
15	Cable Tray and Conduit Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
16	Cable Tray and Conduit Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
17	Cable Tray and Conduit Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504
18	Cable Tray and Conduit Supports	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
19	Cable Trays and Conduits	EN, SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	C
20	Cable Trays and Conduits	EN, SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
21	Cable Trays and Conduits	EN, SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504
22	Cable Trays and Conduits	EN, SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B3-5	3.5.1-59	A
23	Cable Trays and Conduits	EN, SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	C

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
24	Cable Trays and Conduits	EN, SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
25	Cable Trays and Conduits	EN, SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C
26	Cable Trays and Conduits	EN, SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504
27	Cable Trays and Conduits	EN, SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
28	Cable Trays and Conduits	EN, SNS, SRE, SSR	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
29	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Carbon Steel	Air-indoor	Loss of material	ISI Program-IWF	III.B1.1-13 III.B1.2-10 III.B1.3-10	3.5.1-53	A
30	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-14 III.B1.2-11 III.B1.3-11	3.5.1-55	A 0504
31	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-8 III.B1.2-6 III.B1.3-6	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
32	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B1.1-9 III.B1.2-7 III.B1.3-7	3.5.1-59	A
33	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	ISI Program-IWF	III.B1.2-10 III.B1.3-10	3.5.1-53	A
34	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B1.2-11 III.B1.3-11	3.5.1-55	A 0504
35	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	ISI Program-IWF	III.B1.2-10 III.B1.3-10	3.5.1-53	A
36	Component and Piping Supports (ASME Class 1, 2, and 3)	SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B1.2-6 III.B1.3-6	3.5.1-55	A 0504
37	Damper Framing (in-wall)	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
38	Damper Framing (in-wall)	SNS, SRE, SSR	Galvanized Steel	Air-indoor	None	None	III.B2-5	3.5.1-58	C

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
39	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B3-7	3.5.1-39	C
40	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B3-8	3.5.1-55	C 0504
41	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B3-4	3.5.1-55	C 0504
42	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B3-5	3.5.1-59	C
43	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B3-7	3.5.1-39	C
44	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B3-8	3.5.1-55	C 0504
45	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C
46	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B3-4	3.5.1-55	C 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
47	Electrical and Instrument Panels & Enclosures	EN, SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C
48	Electrical Cable Bus Ducts	EN, SRE, SSR	Aluminum	Air-indoor	None	None	III.B3-2	3.5.1-58	C
49	Electrical Cable Bus Ducts	EN, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	VI.A-13	3.6.1-9	A
50	Electrical Cable Bus Ducts	EN, SRE, SSR	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
51	Electrical Cable Bus Ducts	EN, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	VI.A-13	3.6.1-9	A
52	Equipment Component Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A
53	Equipment Component Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504
54	Equipment Component Supports	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
55	Equipment Component Supports	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8 III.B3-5 III.B4-8 III.B5-5	3.5.1-59	A
56	Equipment Component Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A
57	Equipment Component Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504
58	Equipment Component Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A
59	Equipment Component Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504
60	Equipment Component Supports	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A
61	Flood, Pressure, and Specialty Doors	FLB, MB, SPB, SHD, SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B4-10	3.5.1-39	C

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
62	Flood, Pressure, and Specialty Doors	FLB, MB, SPB, SHD, SNS, SRE, SSR	Galvanized Steel	Air-indoor	None	None	III.B4-5	3.5.1-58	C
63	Flood, Pressure, and Specialty Doors	FLB, MB, SPB, SHD, SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-10	3.5.1-39	C
64	Flood, Pressure, and Specialty Doors	FLB, MB, SPB, SHD, SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
65	HELB Barriers (includes pipe restraints, whip restraints, and jet/missile impingement shields/plate barriers)	HELB, PW, SNS, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B5-7	3.5.1-39	C

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
66	HELB Barriers (includes pipe restraints, whip restraints, and jet/missile impingement shields/plate barriers)	HELB, PW, SNS, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
67	HELB Barriers (includes pipe restraints, whip restraints, and jet/missile impingement shields/plate barriers)	HELB, PW, SNS, SSR	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	C
68	HELB Barriers (includes pipe restraints, whip restraints, and jet/missile impingement shields/plate barriers)	HELB, PW, SNS, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
69	HVAC Duct Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
70	HVAC Duct Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
71	HVAC Duct Supports	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504
72	HVAC Duct Supports	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8	3.5.1-59	A
73	Instrument Line Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
74	Instrument Line Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
75	Instrument Line Supports	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504
76	Instrument Line Supports	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8	3.5.1-59	A
77	Instrument Line Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
78	Instrument Line Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
79	Instrument Line Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
80	Instrument Line Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
81	Instrument Line Supports	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
82	Instrument Racks and Frames	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B3-7	3.5.1-39	C
83	Instrument Racks and Frames	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B3-8	3.5.1-55	A 0504
84	Instrument Racks and Frames	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B3-7	3.5.1-39	C
85	Missile Barriers	MB, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B5-7	3.5.1-39	C
86	Missile Barriers	MB, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
87	Missile Barriers	MB, SSR	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	C
88	Missile Barriers	MB, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
89	Missile Barriers	MB, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B5-7	3.5.1-39	C
90	Missile Barriers	MB, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
91	Monorails, Hoists and Miscellaneous Cranes	SNS	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B5-7	3.5.1-39	A
92	Penetrations (Mechanical and Electrical)	EN, FB, FLB, SPB, SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring Fire Protection	III.B2-10	3.5.1-39	C 0547
93	Pipe Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10 III.B4-10	3.5.1-39	A
94	Pipe Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B4-11	3.5.1-55	A 0504
95	Pipe Supports	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B4-6	3.5.1-55	A 0504
96	Pipe Supports	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8 III.B4-8	3.5.1-59	A
97	Pipe Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10 III.B4-10	3.5.1-39	A
98	Pipe Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B4-11	3.5.1-55	A 0504
99	Pipe Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
100	Pipe Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B4-6	3.5.1-55	A 0504
101	Pipe Supports	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A
102	Pipe Supports	SNS, SRE, SSR	Stainless Steel	Treated water	Loss of material	Structures Monitoring PWR Water Chemistry	III.B1.1-11	3.5.1-49	J 0545
103	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Aluminum	Air-indoor	None	None	III.B5-2	3.5.1-58	C
104	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Aluminum	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
105	Stairs, Ladders, Platforms, and Gratings	FLB, SNS, SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B5-7	3.5.1-39	C 0548
106	Stairs, Ladders, Platforms, and Gratings	FLB, SNS, SRE	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504 0548
107	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	C

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
108	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
109	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B4-7	3.5.1-50	C
110	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Aluminum	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
111	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B5-7	3.5.1-39	C
112	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B5-8	3.5.1-55	C 0504
113	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C
114	Stairs, Ladders, Platforms, and Gratings	SNS, SRE	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	C 0504
115	Tube Track Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
116	Tube Track Supports	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
117	Tube Track Supports	SNS, SRE, SSR	Galvanized Steel	Air-indoor	None	None	III.B5-3	3.5.1-58	A
118	Tube Track Supports	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504
119	Tube Track Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
120	Tube Track Supports	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	A 0504
121	Tube Track Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	A
122	Tube Track Supports	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6	3.5.1-55	A 0504
123	Tube Tracks	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	C
124	Tube Tracks	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	C
125	Tube Tracks	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	C
126	Tube Tracks	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11	3.5.1-55	C 0504
127	Vents and Louvers	SNS, SRE, SSR	Aluminum	Air-indoor	None	None	III.B2-4	3.5.1-58	C

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
128	Vents and Louvers	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	C
129	Vents and Louvers	SNS, SRE, SSR	Galvanized Steel	Air-indoor	None	None	III.B2-5	3.5.1-58	C
130	Vents and Louvers	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8	3.5.1-59	C
131	Vents and Louvers	SNS, SRE, SSR	Aluminum	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C
132	Vents and Louvers	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	C
133	Vents and Louvers	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C
134	Vents and Louvers	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7	3.5.1-50	C
135	Vibration Isolators	SNS, SRE	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10	3.5.1-39	A
Threaded Fasteners									
136	Anchor Bolts	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
137	Anchor Bolts	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504
138	Anchor Bolts	SNS, SRE, SSR	Carbon Steel	Air-indoor	Cracking	Bolting Integrity	III.B1.1-3	3.5.1-51	C 0537 0544
139	Anchor Bolts	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504
140	Anchor Bolts	SNS, SRE, SSR	Stainless Steel	Air-indoor	Cracking	Bolting Integrity	III.B1.1-3	3.5.1-51	C 0537 0544
141	Anchor Bolts	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A
142	Anchor Bolts	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504
143	Anchor Bolts	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
144	Anchor Bolts	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504
145	Anchor Bolts	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A
146	Anchor Bolts	SNS, SRE, SSR	Stainless Steel	Air-outdoor	Cracking	Bolting Integrity	III.B1.1-3	3.5.1-51	C 0537 0544
147	Anchor Bolts (ASME Class 1, 2, and 3 Supports Bolting)	SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring ISI Program-IWF	III.B1.1-13 III.B1.2-10 III.B1.3-10	3.5.1-53	A
148	Anchor Bolts (ASME Class 1, 2, and 3 Supports Bolting)	SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-14 III.B1.2-11	3.5.1-55	A 0504
149	Anchor Bolts (ASME Class 1, 2, and 3 Supports Bolting)	SRE, SSR	Carbon Steel	Air-indoor	Cracking	Bolting Integrity	III.B1.1-3	3.5.1-51	A
150	Anchor Bolts (ASME Class 1, 2, and 3 Supports Bolting)	SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B1.1-8 III.B1.2-6	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
151	Anchor Bolts (ASME Class 1, 2, and 3 Supports Bolting)	SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B1.1-9	3.5.1-59	A
152	Anchor Bolts (ASME Class 1, 2, and 3 Supports Bolting)	SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring ISI Program-IWF	III.B1.2-10 III.B1.3-10	3.5.1-53	A
153	Anchor Bolts (ASME Class 1, 2, and 3 Supports Bolting)	SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring ISI Program-IWF	III.B1.2-10 III.B1.3-10	3.5.1-53	A
154	Blowout Panel Release Fasteners	PR, SSR	Aluminum	Air-indoor	None	Structures Monitoring	III.B4-4	3.5.1-58	C
155	Blowout Panel Release Fasteners	PR, SSR	Aluminum	Air-indoor	Loss of material	Boric Acid Corrosion	III.B5-4	3.5.1-55	A 0504
156	Expansion Anchors	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A
157	Expansion Anchors	SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
158	Expansion Anchors	SNS, SRE, SSR	Carbon Steel	Air-indoor	Cracking	Bolting Integrity	III.B1.1-3	3.5.1-51	C 0544
159	Expansion Anchors	SNS, SRE, SSR	Galvanized Steel	Air-indoor	None	None	III.B2-5 III.B3-3 III.B4-5 III.B5-3	3.5.1-58	A
160	Expansion Anchors	SNS, SRE, SSR	Galvanized Steel	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504
161	Expansion Anchors	SNS, SRE, SSR	Stainless Steel	Air-indoor	None	None	III.B2-8 III.B3-5 III.B4-8 III.B5-5	3.5.1-59	A
162	Expansion Anchors	SNS, SRE, SSR	Stainless Steel	Air-indoor	Cracking	Bolting Integrity	III.B1.1-3	3.5.1-51	C 0544
163	Expansion Anchors	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-10 III.B3-7 III.B4-10 III.B5-7	3.5.1-39	A
164	Expansion Anchors	SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-11 III.B3-8 III.B4-11 III.B5-8	3.5.1-55	A 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
165	Expansion Anchors	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	A
166	Expansion Anchors	SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	A 0504
Concrete									
167	Equipment Pads	SNS, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
168	Equipment Pads	SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
169	Equipment Pads	SNS, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
170	Equipment Pads	SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
171	Equipment Pads	SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
172	Equipment Pads	SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
173	Flood Curbs	FLB, SNS	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
174	Hatches & Hatch Plugs	EN, FB, FLB, MB, SPB, SHD, SNS, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring Fire Protection	N/A	N/A	I 0501 0547
175	Hatches & Hatch Plugs	EN, FB, FLB, MB, SPB, SHD, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring Fire Protection	III.A3-6	3.5.1-26	A 0547
176	Hatches & Hatch Plugs	EN, FB, FLB, MB, SPB, SHD, SNS, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring Fire Protection	III.A3-7	3.5.1-32	A 0509 0547
177	Hatches & Hatch Plugs	EN, FB, FLB, MB, SPB, SHD, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring Fire Protection	III.A3-9	3.5.1-23	A 0547

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
178	Hatches & Hatch Plugs	EN, FB, FLB, MB, SPB, SHD, SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring Fire Protection	III.A3-10	3.5.1-24	A 0547
179	Support Pedestals	SNS, SRE, SSR	Concrete	Air-indoor	None	Structures Monitoring	N/A	N/A	I 0501
180	Support Pedestals	SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Cracking	Structures Monitoring	III.A3-6	3.5.1-26	A
181	Support Pedestals	SNS, SRE, SSR	Concrete	Air-outdoor	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
182	Support Pedestals	SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material	Structures Monitoring	III.A3-9	3.5.1-23	A
183	Support Pedestals	SNS, SRE, SSR	Concrete	Air-outdoor	Loss of material Change in material properties	Structures Monitoring	III.A3-10	3.5.1-24	A

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
184	Support Pedestals	SNS, SRE, SSR	Concrete	Soil	Change in material properties	Structures Monitoring	III.A3-7	3.5.1-32	A 0509
				Elastomers					
185	Compressible Joints and Seals	EXP, FLB, SNS, SSR	Elastomer	Air-indoor	Cracking Change in material properties	Structures Monitoring	III.A6-12	3.5.1-44	C 0538 0539
186	Compressible Joints and Seals	EXP, FLB, SNS, SSR	Elastomer	Air-outdoor	Cracking Change in material properties	Structures Monitoring	III.A6-12	3.5.1-44	C 0538 0540
187	Expansion Boots	EXP, FLB, SNS, SRE, SSR	Elastomer	Air-outdoor	Cracking Change in material properties	Structures Monitoring	III.A6-12	3.5.1-44	C 0538 0540
188	Expansion Boots	EXP, FLB, SNS, SRE, SSR	Elastomer	Soil	None	Structures Monitoring	N/A	N/A	J 0543

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities									
Row No.	**Component / Commodity**	**Intended Function[1]**	**Material**	**Environment**	**Aging Effect Requiring Management**	**Aging Management Program**	**NUREG-1801, Volume 2 Item**	**Table 1 Item**	**Notes**
189	Flexible Conduit Fittings	EN, SNS, SRE, SSR	Elastomer	Air-indoor	Cracking Change in material properties	Structures Monitoring	III.A6-12	3.5.1-44	C 0538 0539
190	Flexible Conduit Fittings	EN, SNS, SRE, SSR	Elastomer	Air-outdoor	Cracking Change in material properties	Structures Monitoring	III.A6-12	3.5.1-44	C 0538 0540
191	Roof Membrane	EN, FLB, SNS, SRE, SSR	Elastomer / Built-up Roofing	Air-outdoor	Cracking Change in material properties	Structures Monitoring	III.A6-12	3.5.1-44	C 0538 0540
192	Waterproofing Membrane	FLB, SNS, SSR	Elastomer	Soil	None	Structures Monitoring	N/A	N/A	J 0543
193	Waterstops	FLB, SNS, SSR	Elastomer	Air-indoor (within walls, floors, or foundations)	None	Structures Monitoring	N/A	N/A	J 0543
194	Waterstops	FLB, SNS, SSR	Elastomer	Soil	None	Structures Monitoring	N/A	N/A	J 0543

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
Fire Barriers									
195	Fire Doors	FB, SNS, SRE, SSR	Carbon Steel	Air-indoor	Loss of material	Fire Protection	VII.G-3	3.3.1-63	B 0541 0546
						Structures Monitoring	III.B4-10	3.5.1-39	C
196	Fire Doors	FB, SNS, SRE, SSR	Galvanized Steel	Air-indoor	None	Fire Protection	N/A	N/A	I 0501
						Structures Monitoring	N/A	N/A	I 0501
197	Fire Doors	FB, SNS, SRE, SSR	Carbon Steel	Air-outdoor	Loss of material	Fire Protection	VII.G-4	3.3.1-63	B 0541 0546
						Structures Monitoring	III.B4-10	3.5.1-39	C
198	Fire Doors	FB, SNS, SRE, SSR	Galvanized Steel	Air-outdoor	Loss of material	Fire Protection	VII.G-4	3.3.1-63	B 0541 0546
						Structures Monitoring	III.B4-7	3.5.1-50	C
199	Fire Stops	FB, FLB, SPB, SNS, SRE, SSR	Silicone Elastomer	Air-indoor	Cracking/ Delamination/ Separation	Fire Protection	VII.G-1	3.3.1-61	B 0541 0542

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
200	Fire Stops	FB, FLB, SPB, SNS, SRE, SSR	Silicone Elastomer	Air-outdoor	Cracking/ Delamination/ Separation	Fire Protection	VII.G-1	3.3.1-61	B 0541 0542
201	Fire Stops	FB, FLB, SPB, SNS, SRE, SSR	Silicone Elastomer	Air-indoor	Change in material properties	Fire Protection	VII.G-1	3.3.1-61	B 0541 0542
202	Fireproofing	FB, SNS, SRE, SSR	Isolatek Mandoseal/ Monokote	Air-indoor	Loss of material Cracking/ Delamination	Fire Protection	N/A	N/A	J
203	Fire Wraps	FB, SNS, SRE, SSR	Ceramic fiber/ 3M Interam	Air-indoor	Loss of material Cracking/ Delamination	Fire Protection	N/A	N/A	J
Miscellaneous Materials									
204	Containment Penetration Insulation	SNS	Fiberglass	Air-indoor	None	None	N/A	N/A	J

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
205	Piping and Mechanical Equipment Insulation	SNS	Aluminum jacketing	Air-indoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	J 0504
206	Piping and Mechanical Equipment Insulation	SNS	Calcium Silicate	Air-indoor	None	None	N/A	N/A	J
207	Piping and Mechanical Equipment Insulation	SNS	Fiberglass	Air-indoor	None	None	N/A	N/A	J
208	Piping and Mechanical Equipment Insulation	SNS	Stainless Steel Mirror insulation	Air-indoor	None	None	N/A	N/A	J
209	Piping and Mechanical Equipment Insulation	SNS	Aluminum jacketing	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	J
210	Piping and Mechanical Equipment Insulation	SNS	Aluminum jacketing	Air-outdoor	Loss of material	Boric Acid Corrosion	III.B2-6 III.B3-4 III.B4-6 III.B5-4	3.5.1-55	J 0504

Table 3.5.2-13 Aging Management Review Results – Bulk Commodities

Row No.	Component / Commodity	Intended Function[1]	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
211	Piping and Mechanical Equipment Insulation	SNS	Calcium Silicate	Air-outdoor	None	None	N/A	N/A	J
212	Piping and Mechanical Equipment Insulation	SNS	Fiberglass	Air-outdoor	None	None	N/A	N/A	J
213	Piping and Mechanical Equipment Insulation	SNS	Stainless Steel Mirror insulation	Air-outdoor	Loss of material	Structures Monitoring	III.B2-7 III.B4-7	3.5.1-50	J

Generic 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 item 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:	
0501	No applicable aging effects have been identified for the component type. However, the identified AMP or AMPs will be used to confirm the absence of significant aging effects for the period of extended operation.
0502	The containment emergency sump recirculation valve enclosures and bellows are extensions of the containment pressure boundary and provide an essentially leak tight barrier. They are locally leak tested similar to containment penetration bellows that serve as containment pressure boundaries.
0503	The containment normal sump is assumed to have a raw water environment for license renewal evaluation because system leakages can be from various sources and may contain contaminants. It is assumed that the waste liquid collected in the stainless steel lined sump can be aggressive. Therefore, loss of material is an aging effect requiring management for the sump. The material and environment combination is not evaluated in NUREG-1801 civil chapters II or III.
0504	Aging mechanism applies to the areas that contain borated systems.
0505	Elastomeric seals, gaskets, or o-rings are sub-parts of the host component and their leak tightness is monitored by the 10 CFR Part 50, Appendix J Program. Plant Technical Specifications ensure that access airlocks maintain leak tightness in the closed position.
0506	Neutron shielding material is used for radiation shielding only and is not relied upon as a structural element. Neutron shielding material is enclosed within steel covering. Therefore, neutron shielding material does not require aging effects evaluation. Aging effects evaluation is performed on the outer steel panels.
0507	The process line penetrations are of welded steel construction without gaskets, or sealing compounds. Electrical penetration assembly internal o-rings are sub-components of each electrical penetration and are included in this commodity group. Insulation for hot penetrations is addressed in bulk commodities.
0508	The refueling canal has experienced leakage through the refueling canal liner. The repair of the refueling canal leakage is processed by the Corrective Action Program. The identified AMP will be used to confirm the absence of significant aging effects for the period of extended operation.
0509	The NUREG-1801 item for leaching of calcium hydroxide does not list exposed to soil, air-indoor or air-outdoor environments. Water leakages through concrete (above and below grade) have been observed at the plant from operating experience. The environment is considered a match since the degradation initiation mechanism is the same. The identified AMP is used to manage aging effects for the period of extended operation.
0510	Lead is used for radiation shielding only and is not relied upon as a structural element. Lead shielding material is enclosed within steel covering. Therefore, lead does not require aging effects evaluation. Aging effects evaluation is performed on the outer steel covering.

Plant-Specific Notes:	
0511	In addition to aging management by the Structures Monitoring Program, the Shield Building concrete is also managed by the 10 CFR Part 50, Appendix J Program's Containment Vessel and Shield Building Visual Inspection.
0512	Concrete walls, floors, and ceilings with a fire barrier (FB) intended function receive additional inspection as part of the Fire Protection Program.
0513	The TLAAs excluding the Containment Vessel from fatigue analysis per Section N415-1 of the ASME Code will remain valid through the period of extended operation (See Section 4.6.1).
0514	The effects of fatigue on the intended functions of the Permanent Reactor Cavity Seal Plate seal membrane will be managed for the period of extended operation by the Fatigue Monitoring Program.
0515	Lead is used for radiation shielding only and is not relied upon as a structural element. Lead is protected within steel panels, masonry walls, or concrete plugs. As such, aging management is performed on the covering material. Radiation shielding panels have lead bricks or lead panels protected with steel plates. Lead bricks are sandwiched within reinforced masonry walls. Temporary lead blankets are hung on steel supports. Lead plates are installed between concrete hatch plugs. Lead shot, covered with steel panels, is used to fill trenches containing radioactive piping.
0516	The PWR Water Chemistry Program manages loss of material due to crevice and pitting corrosion. Cracking due to SCC is not applicable. Spent fuel pool water level monitoring is per Technical Specifications. The Leak Chase Monitoring Program detects leakage from the leak chase channels.
0517	Masonry Walls are inspected by the Masonry Wall Inspection implemented as part of the Structures Monitoring Program. Masonry walls with a fire barrier (FB) intended function receive additional inspection as part of the Fire Protection Program.
0518	The roof has built-up roofing. Therefore, the environment for this concrete roof slab is air-indoor for the underside of the slab. The roof membrane is evaluated and addressed in bulk commodities.
0519	The Auxiliary Building sump is assumed to have a raw water environment for license renewal evaluation because system leakages can be from various sources and may contain contaminants. It is assumed that the waste liquid collected in the sump can be aggressive. Therefore, loss of material and change in material properties are aging effects requiring management for the sump. The NUREG-1801 items for aggressive chemical attack, corrosion of embedded steel and steel reinforcement, and leaching of calcium hydroxide do not list a raw water environment. The environment is considered a match since the degradation initiation mechanism is the same. The identified AMP is used to manage aging effects for the period of extended operation.
0520	The listed AMP is a plant-specific program for this item. Davis-Besse plant-specific AMR concluded Boral® does not require aging management for the period of extended operation for its neutron absorbing function. However, because of recent industry experience, a new Boral® Monitoring Program will be instituted at Davis-Besse for the period of extended operation. Aging management for loss of material of its aluminum constituent is required.

Plant-Specific Notes:	
0521	The Leak Chase Monitoring Program detects leakage from the leak chase channels during normal operation and refueling.
0522	Davis-Besse is not committed to Regulatory Guide 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants, Revision 1. However, the Water Control Structures Inspection as implemented by the Structures Monitoring Program will be enhanced to include applicable inspection elements delineated in Regulatory Guide 1.127, Revision 1 per NUREG-1801 Chapter XI.S7.
0523	The NUREG-1801 item for freeze-thaw does not list exposed to raw water environment. Freeze-thaw can be possible near the water line. This environment is both exposed to air-outdoor and exposed to raw water; therefore environment is considered a match. The identified AMP is used to manage aging effects for the period of extended operation.
0524	The NUREG-1801 item for corrosion of embedded steel and steel reinforcement does not list exposed to soil environment. Concrete components below grade are exposed to an aggressive groundwater environment. The environment is considered a match since the degradation initiation mechanism is the same. The identified AMP is used to manage aging effects for the period of extended operation.
0525	The NUREG-1801 item for aggressive chemicals does not list exposed to air-outdoor environment. Concrete components in an air-outdoor environment are exposed to an aggressive rainwater environment. Their external surfaces may be wetted for a period of time due to moderate precipitation and snowfall. The environment is considered a match since the degradation initiation mechanism is the same. The identified AMP is used to manage aging effects for the period of extended operation.
0526	The service water discharge pipe sleeve and the buried portion of the Service Water Discharge Structure do not contain steel reinforcement therefore the corrosion of embedded steel and steel reinforcement aging mechanism is not applicable. The Service Water Discharge Structure end-wall, slab, and spillway do contain steel reinforcement and are exposed to air-outdoor on the top sides, soil on the bottom sides. The corrosion of embedded steel and steel reinforcement aging mechanism is applicable.
0527	The Intake Structure sump is assumed to have a raw water environment for license renewal evaluation because system leakages can be from various sources and may contain contaminants. It is assumed that the waste liquid collected in the sump can be aggressive. Therefore, loss of material and change in material properties due to aggressive chemicals are aging effects requiring management for the sump. The identified AMP is used to manage aging effects for the period of extended operation.
0528	Rock anchors provide rock stability in the vicinity of sheet piling anchors. Rock anchors are grouted into bedrock. The Structural Tools does not list a concrete or grouted environment for steel components. Steel embedded in concrete does not require aging management. This conclusion is consistent with NUREG-1801 item VII.J-21 and the Mechanical Tools.
0529	The Intake Structure gantry crane is located in an air-outdoor environment. The NUREG-1801 item for crane VII.B-3 only listed an air-indoor (uncontrolled) environment. The identified AMP is used to manage aging effects for the period of extended operation.

Plant-Specific Notes:	
0530	Building sumps are assumed to have a raw water environment for license renewal evaluation because system leakages can be from various sources and may contain contaminants. It is assumed that the waste liquid collected in the sump can be aggressive. Therefore, loss of material and change in material properties are aging effects requiring management for the sump. The NUREG-1801 items for aggressive chemical, corrosion of embedded steel and steel reinforcement, and leaching of calcium hydroxide do not list a raw water environment. The environment is considered a match since the degradation initiation mechanism is the same. The identified AMP is used to manage aging effects for the period of extended operation.
0531	NUREG-1801 does not list a structural backfill environment for steel components. No aging effects requiring management were identified for the EDG Fuel Oil Storage Tank hold down wire rope in a structural backfill environment. However, the identified AMP will be used to confirm the absence of significant aging effects for the period of extended operation. The structural backfill is above grade and the elevation location of the wire rope is above the site's groundwater elevation.
0532	The Wave Protection Dike corrugated pipe casings and Wave Protection Dike piles buried in the wave protection dikes can be exposed to groundwater since the corrugated pipe casings are located below site groundwater elevation. Since these buried steel components can be in direct contact with groundwater, a raw water environment is conservatively used for aging evaluation.
0533	Walls with a fire barrier (FB) intended function receive additional inspection as part of the Fire Protection Program.
0534	Structure sumps are assumed to have a raw water environment for license renewal evaluation because system leakages can be from various sources and may contain contaminants. It is assumed that the waste liquid collected in the sump can be aggressive. Therefore, loss of material and change in material properties are aging effects requiring management for the sump. The NUREG-1801 items for aggressive chemical attack, corrosion of embedded steel and steel reinforcement, and leaching of calcium hydroxide do not list a raw water environment. The environment is considered a match since the degradation initiation mechanism is the same. The identified AMP is used to manage aging effects for the period of extended operation.
0535	Structure sumps are assumed to have a raw water environment for license renewal evaluation because system leakages can be from various sources and may contain contaminants. Loss of material and cracking due to freeze-thaw are aging effects requiring management for concrete components exposed to raw water because Yard Structure sumps are located within outdoor structures. This is conservative since the transformer sumps and manhole sumps are located below grade elevation.
0536	The north and west sides of the Nitrogen Storage Building do not have reinforced concrete walls. Instead, they have a chain link fence. Therefore an air-outdoor environment was assigned inside the building.
0537	Applicable to low-alloy high strength bolts with yield strength (Sy) greater than 150 ksi, Low-alloy Quenched and Tempered (LAQT), and high-nickel managing steel bolting with high tensile stresses in a corrosive environment.

Plant-Specific Notes:	
0538	The NUREG-1801 item lists loss of sealing as an aging effect for elastomer. Loss of sealing is not considered as an aging effect but rather as a consequence of elastomer degradation. This effect may be caused by cracking or change in material properties for elastomeric material. Note C is used since the NUREG-1801 item is intended for Group 6 – water-control structures' components; the line item covers all in-scope structures.
0539	Ionizing radiation is an applicable aging mechanism for elastomers inside Containment and portions of the Auxiliary Building where the radiation exceeds the threshold. The ionizing radiation mechanism does not apply to elastomers located in mild radiation areas.
0540	Cracking and change in material properties due to ultra-violet radiation and ozone are applicable aging effects for rubber only.
0541	The Fire Protection Program does not contain any exceptions which are applicable to structural components.
0542	The NUREG-1801 item lists aging effects as increased hardness, shrinkage, and loss of strength. The applicable aging effects identified are cracking/ delamination and change in material properties. The aging effect is a match since increased hardness, shrinkage and loss of strength are consequences of a change in material properties. Gamma irradiation mechanism does not apply to elastomeric fire stops located in mild radiation areas.
0543	No applicable aging effects have been identified for the component type. However, Davis-Besse operating experience indicates groundwater in-leakage. Therefore, elastomer seals below grade and waterstops require aging management when accessible.
0544	Aging effect applies to expansion anchors because of the use of moly-sulfide based lubricant. Aging effect applies to non-Class 1 anchor bolts because of Davis-Besse operating experience with boric acid wastage which is a corrosive environment.
0545	Components are the stainless steel supports in the spent fuel pool which are not within the scope of ISI-IWF. The identified AMPs will be used to manage the aging effects for the period of extended operation.
0546	The aging mechanism loss of material due to wear is not an aging effect for fire doors since wear of the hardware, appurtenances and closure mechanisms is a consequence of frequent or rough usage. The aging mechanism loss of material due to general corrosion was not specified in the corresponding NUREG-1801 item as an aging effect requiring management. Generic Note "A" was used to align to the NUREG-1801 item since the material, environment, aging effect, and program (MEAP) match. The identified AMP will be used to manage loss of material due to general corrosion and will confirm the absence of significant wear of fire doors for the period of extended operation. The Fire Protection Program inspects for excessive wear of latches, strike plates, hinges, sills, and closing devices, and proper clearances (gaps) between the door, frame, and threshold.
0547	Components with a fire barrier (FB) intended function receive additional inspection as part of the Fire Protection Program.
0548	There are sections of checkered plate flooring installed over the heater bay grating in the Turbine Building at elevation 603'-0". This flooring is installed to protect the auxiliary feedwater pumps from flooding.

Plant-Specific Notes:	
0549	Porcelain window wall panels are an architectural feature that serve a shelter intended function for the Condensate Storage Tank room. A review of the site-specific operating experience and industry operating experience has not identified any aging effects that can affect or challenge the intended function provided by porcelain window wall panels.

3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROL SYSTEMS

3.6.1 INTRODUCTION

Section 3.6 provides the results of the aging management reviews (AMRs) for those components and commodities identified in Section 2.5, Scoping and Screening Results – Electrical and Instrumentation and Control Systems, subject to AMR. The components and commodity groups subject to AMR are:

- Non-Environmentally Qualified Insulated Cables and Connections (Section 2.5.5.1)
- Switchyard Bus and Connections (Section 2.5.5.2)
- Transmission Conductors and Connections (Section 2.5.5.3)
- High-Voltage Insulators (Section 2.5.5.4)

Table 3.6.1, Summary of Aging Management Programs for Electrical and I&C Components Evaluated in Chapter VI of NUREG-1801, provides the summary of the programs evaluated in NUREG-1801 that are applicable to component and commodity groups in this section. Text addressing summary items requiring further evaluation is provided in Section 3.6.2.2.

3.6.2 RESULTS

The following table summarizes the results of the AMR for the components and commodity groups in the Electrical and Instrumentation and Control Systems area:

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups

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

The materials from which specific components and commodity groups are fabricated, the environments to which they are exposed, the aging effects requiring management, and the aging management programs (AMPs) used to manage these aging effects are provided for each component and commodity group in the following sections.

3.6.2.1.1 Non-Environmentally Qualified Insulated Cables and Connections

The Non-Environmentally Qualified Insulated Cables and Connections commodity group is subdivided for AMR into the following categories:

- Non-Environmentally Qualified Insulated Cables and Connections
- Non-Environmentally Qualified Electrical and I&C Penetration Assemblies (electrical insulation)
- Non-Environmentally Qualified Sensitive, High-Voltage, Low-Level Signal Instrument Cables and Connections
- Non-Environmentally Qualified Medium-Voltage Power Cables
- Cable Connections (Metallic Parts)

Materials

The materials of construction for the Non-Environmentally Qualified Insulated Cables and Connections are:

- Various metals
- Various organic polymers

Environments

The Non-Environmentally Qualified Insulated Cables and Connections are exposed to the following environments:

- Adverse localized environments
- Air – indoor uncontrolled
- Air – outdoor
- Air with borated water leakage

Aging Effects Requiring Management

The aging effects requiring management for the Non-Environmentally Qualified Cables and Connections exposed to adverse localized environments are the following:

- Increased connection resistance
- Reduced insulation resistance

Aging Management Programs

The following aging management programs manage the aging effects for the Non-Environmentally Qualified Cables and Connections components:

- 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 Environmental Qualification Requirements Used in Instrumentation Circuits Program
- Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection
- Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program
- Boric Acid Corrosion Program (for the metallic cable connections exposed to air with borated water leakage)

3.6.2.1.2 Switchyard Bus and Connections

The Switchyard Bus and Connections commodity group is evaluated for aging management as follows:

Materials

The materials of construction for the Switchyard Bus and Connections are:

- Aluminum
- Galvanized steel
- Stainless steel

Environments

The Switchyard Bus and Connections are exposed to the following environment:

- Air - outdoor

Aging Effects Requiring Management

There are no aging effects identified as requiring management for the Switchyard Bus and Connections components (See Section 3.6.2.2.3).

Aging Management Programs

There are no aging effects identified as requiring management; therefore, no aging management programs are required for the Switchyard Bus and Connections components.

3.6.2.1.3 Transmission Conductors and Connections

The Transmission Conductors and Connections commodity group is evaluated for aging management as follows:

Materials

Transmission conductors are aluminum conductor aluminum reinforced (ACAR). The materials of construction for the Transmission Conductor and Connection components are:

- Aluminum
- Galvanized steel
- Stainless steel

Environments

The Transmission Conductor and Connection components are exposed to the following environment:

- Air - outdoor

Aging Effects Requiring Management

There are no aging effects identified as requiring management for the Transmission Conductor and Connection components (See Section 3.6.2.2.3).

Aging Management Programs

There are no aging effects identified as requiring management; therefore, no aging management programs are required for the Transmission Conductors and Connections components.

3.6.2.1.4 High-Voltage Insulators

The High-Voltage Insulators commodity group is evaluated for aging management as follows:

Materials

The materials of construction for the High-Voltage Insulators are:

- Cement
- Galvanized metal
- Malleable iron
- Porcelain

Environments

The High-Voltage Insulators are exposed to the following environment:

- Air - outdoor

Aging Effects Requiring Management

There are no aging effects identified as requiring management for the High-Voltage Insulator components (See Section 3.6.2.2.2 and Section 3.6.2.2.3).

Aging Management Programs

There are no aging effects identified as requiring management; therefore, no aging management programs are required for the High-Voltage Insulator components.

3.6.2.2 Aging Management Review Results for Which Further Evaluation is Recommended by NUREG-1801

For the electrical and instrumentation and control (I&C) components, the items that require further evaluation are addressed in the following sections.

3.6.2.2.1 Electrical Equipment Subject to Environmental Qualification

Environmental qualification (EQ) is a time-limited aging analysis as defined in 10 CFR 54.3. Time-limited aging analyses are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of the environmental qualification time-limited aging analysis is addressed separately in Section 4.

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

Degradation of insulator quality due to presence of any salt deposits and surface contamination could occur in high voltage insulators.

The high-voltage insulators evaluated for license renewal at Davis-Besse include those used to support and insulate high-voltage electrical components (i.e., transmission conductors and connections, and switchyard bus). The in-scope power pathway involves the transmission conductors and connections associated with Startup Transformers 01 and 02, and the in-scope transmission conductors and connections located in the 345-kV switchyard adjacent to the plant.

Various airborne contaminants such as dust and industrial effluents can contaminate the insulator surfaces. The rural location of Davis-Besse on the shore of Lake Erie provides for minimal contamination from industrial effluents, and the city of Toledo is more than 20 miles away. The regular rainfall at the site is sufficient to wash the insulators. There have been no incidents of insulator contamination causing flashover or other insulator failures at Davis-Besse.

Loss of material due to mechanical wear is an aging effect for certain strain insulators if they are subject to significant movement. Such movement of the insulators can be caused by wind blowing the supported transmission conductor, causing it to sway from side to side. If this swinging motion occurs frequently enough, it could cause wear on the metallic contact points of the insulator string and between an insulator and the supporting hardware. Although this aging mechanism is possible, industry experience has shown that transmission conductors do not normally swing unless subjected to a substantial wind, and they stop swinging shortly after the wind subsides. Wind loading that can result in conductor sway is considered in the transmission system design. In addition, the sections of transmission conductor that are within the license renewal evaluation boundary at Davis-Besse are relatively short (from Startup Transformers 01 and 02 into the plant switchyard in lengths of about 200 feet, and then in increments of about 70 feet within the switchyard itself). Therefore, loss of material due to mechanical wear is not an aging effect requiring management for the high voltage insulators at Davis-Besse.

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

At Davis-Besse, there are relatively short lengths of switchyard bus in scope for license renewal, located in the plant switchyard. This bus is fabricated of 4-inch and 5-inch aluminum tube. The switchyard bus is connected to flexible connections that do not normally vibrate and are supported by insulators and ultimately by static structural components such as concrete footings and structural steel.

The aluminum bus will form a thin surface layer of oxidation but the conductor properties are not degraded by this thin surface oxidation layer. The galvanized and aluminum bolted connections are exposed to the same service conditions (in the plant switchyard) and do not experience any aging effects, except for minor oxidation of the exterior surfaces, which does not impact their ability to perform their intended function.

For the transmission conductors and connections and the switchyard bus and connections, subject to aging management review, there are no aging effects identified that require aging management.

Wind-induced abrasion and fatigue are not aging effects applicable to the in-scope transmission conductors. Industry experience has shown that transmission conductors do not normally swing unless subjected to substantial winds and they stop swinging after a short period once the wind subsides. Because the transmission conductors are not normally moving, the loss of material due to wind-induced abrasion and fatigue is not an aging effect requiring management. In addition, wind loading that can result in conductor sway is considered in the transmission system design.

Loss of conductor strength due to corrosion of the transmission conductor is not identified as an aging effect due to the ample design margin and a minimal corrosion process at Davis-Besse. Connection resistance is not identified as a stressor based on the use of good bolting practices and review of the site operating experience.

In the industry, transmission conductors are generally aluminum conductor steel reinforced (ACSR). The transmission conductor at Davis-Besse is ACAR.

Aluminum is more corrosion-resistant than steel. Aluminum quickly forms an oxide layer which protects the material underneath and this layer will re-form if damaged (in the absence of environmental stress). Aluminum is lighter than steel and provides a much higher strength-to-weight ratio. The ACAR conductor therefore is more resistant to corrosion and to loss of conductor strength than the ACSR conductor.

Corrosion in ACSR conductors is a very slow-acting mechanism and the corrosion rates depend on the air quality, which is affected by suspended particles chemistry, sulfur dioxide concentration in the air, precipitation, fog, air chemistry, and general meteorological conditions. For ACSR conductors, degradation begins as a loss of zinc from the galvanized steel core wires. Air quality in rural areas generally contains low concentrations of suspended particles and sulfur dioxide, which keeps the corrosion rate to a minimum. Davis-Besse is located in a rural area with no other industries in the immediate area.

As described in the EPRI Electrical Handbook, testing performed by Ontario Hydroelectric showed a 30% loss of composite conductor strength of an 80-year-old ACSR conductor due to corrosion. The Ontario Hydroelectric test report is available from the Institute of Electrical and Electronic Engineers (IEEE). The report is documented in two parts, which present the test methods and results on both field and laboratory tests on samples of ACSR conductors from Ontario Hydroelectric's older transmission lines. The field testing involved detection of steel core galvanizing loss via the use of an overhead line conductor corrosion detector. The laboratory tests involved examination of fatigue, tensile strength, torsional ductility, and electrical performance. The report also addressed metallurgical data and analysis of potential environmental contributors.

The Davis-Besse transmission conductors for the 345-kV offsite power recovery path are 1024.5 MCM ACAR, Type T-2614, Bare Cable, 24/13, overhead transmission conductors. The Ontario Hydroelectric test did not include this specific conductor type, but these types are bounded because of the conductor size, configuration, and support strand material. The Ontario Hydroelectric example discussed in the EPRI Electrical Handbook uses a 4/0 ACSR conductor while the Davis-Besse ACAR conductor has 13 aluminum-alloyed conductors wrapped by a 24-strand aluminum wire (24/13). The rated strength of the ACAR configuration is 23,100 lbs. The Davis-Besse conductors have aluminum reinforcing strands, so the Ontario Hydroelectric ACSR transmission conductors would bound the corrosion evaluation for the ACAR conductors at Davis-Besse. The aluminum conductors in the Ontario Hydroelectric test showed very little evidence of corrosion.

The National Electrical Safety Code (NESC) requires that tension on installed conductors be a maximum of 60% of the ultimate conductor strength. The NESC also sets the maximum tension to which a conductor must be designed to withstand under heavy load conditions, which includes the consideration of ice, wind, and temperature. The NESC requirements and the guidance of the EPRI Electrical Handbook and the Ontario Hydroelectric study were applied to evaluate the in-scope Davis-Besse transmission conductors.

The ultimate strength and NESC heavy load strength of the Davis-Besse ACAR conductors are 23,100 lbs. and 13,860 lbs., respectively. The margin between the NESC heavy load and the ultimate strength is 9,240 lbs. The Ontario Hydroelectric study showed a 30% loss of composite conductor strength in the 80 year-old sample. In the case of the Davis-Besse ACAR conductor, a 30% reduction in strength would reduce the ultimate strength from 23,100 lbs. to 16,170 lbs., which still exceeds the NESC heavy load limit of 13,860 lbs. by 2,310 lbs. Therefore, the Davis-Besse ACAR transmission conductors will have an ample margin regarding conductor strength through the period of extended operation.

The bolted connections of the transmission conductors are associated with the field connections of transmission conductor to high-voltage insulators and to switchyard bus. The bolting hardware is chosen to be compatible with the transmission conductor. Stainless steel Belleville washers are specified for use with the transmission conductors. These methods of assembly are consistent with EPRI 1003471.

3.6.2.2.4 Quality Assurance for Aging Management of Nonsafety Related Components

See Appendix B, Section B.1.3, for a discussion of FENOC quality assurance procedures and administrative controls for aging management programs.

3.6.2.3 Time-Limited Aging Analyses

The time-limited aging analyses identified below are associated with the electrical and I&C components. The section of the application that contains the time-limited aging analysis review results is indicated in parentheses.

- Analyses for Environmental Qualification of components with a qualified life of 40 years or greater (Section 4.4, Environmental Qualification of Electrical Equipment)

3.6.3 CONCLUSIONS

The electrical and I&C components and commodities subject to aging management review have been identified in accordance with 10 CFR 54.21. The aging management programs selected to manage the effects of aging for the electrical components and commodities are identified in the following tables and Section 3.6.2.1. A description of the aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the electrical and I&C components and commodities will be managed so 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 Programs for Electrical and I&C Components Evaluated in Chapter VI of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.6.1-01	Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Degradation due to various aging mechanisms	Environmental Qualification of Electrical Components	Yes, TLAA	Evaluation of the EQ time-limited aging analyses (TLAAs) is documented in Section 4.4. Further evaluation is documented in Section 3.6.2.2.1.
3.6.1-02	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.
3.6.1-03	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.
3.6.1-04	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.

Table 3.6.1 Summary of Aging Management Programs for Electrical and I&C Components Evaluated in Chapter VI of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.6.1-05	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. Davis-Besse manages the aging effect with the Boric Acid Corrosion Program.
3.6.1-06	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	Not applicable for Davis-Besse. A review of Davis-Besse documents indicated that fuse holders utilizing metallic clamps are either part of an active electrical panel or are located in circuits that perform no license renewal intended function. Therefore, fuse holders with metallic clamps at Davis-Besse are not subject to aging management review.
3.6.1-07	Metal-enclosed bus – Bus/connections	Loosening of bolted connections due to thermal cycling and ohmic heating	Metal-Enclosed Bus	No	Not applicable to Davis-Besse. There is no metal-enclosed bus within the license renewal evaluation boundary at Davis-Besse.

Table 3.6.1 Summary of Aging Management Programs for Electrical and I&C Components Evaluated in Chapter VI of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.6.1-08	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	Not applicable to Davis-Besse. There is no metal-enclosed bus within the license renewal evaluation boundary at Davis-Besse.
3.6.1-09	Metal-enclosed bus – Enclosure assemblies	Loss of material due to general corrosion	Structures Monitoring Program	No	Not applicable to Davis-Besse. There is no metal-enclosed bus within the license renewal evaluation boundary at Davis-Besse.
3.6.1-10	Metal-enclosed bus – Enclosure Assemblies	Hardening and loss of strength due to elastomer degradation	Structures Monitoring Program	No	Not applicable to Davis-Besse. There is no metal-enclosed bus within the license renewal evaluation boundary at Davis-Besse.

Table 3.6.1 Summary of Aging Management Programs for Electrical and I&C Components Evaluated in Chapter VI of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.6.1-11	High-Voltage Insulators	Degradation of insulation quality due to the 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	Degradation of insulator quality due to the deposition of contaminants on the insulator surface is not an applicable aging effect at Davis-Besse. Further evaluation is documented in Section 3.6.2.2.2.
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 pre-load	A plant-specific aging management program is to be evaluated	Yes, plant-specific	No aging effects are identified as requiring aging management. Further evaluation is documented in Section 3.6.2.2.3.

Table 3.6.1 Summary of Aging Management Programs for Electrical and I&C Components Evaluated in Chapter VI of NUREG-1801

Item Number	Component/Commodity	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
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, with exceptions. See Appendix B Section B.2.11 for details regarding this AMP.
3.6.1-14	Fuse Holders (Not Part of a Larger Assembly) – Insulation Material	None	None	N/A – No AEM or AMP	Consistent with NUREG-1801.

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
1	Cable Connections (metallic parts)	Conduct electricity	Various Metals (used for electrical contact)	Air-indoor uncontrolled and Air-outdoor	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 EQ Requirements Inspection	VI.A-1	3.6.1-13	B

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups									
Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
2	Non-Environment-ally Qualified Insulated Cables and Connections	Conduct electricity	Various Organic Polymers	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/ degradation of organics (thermal/ thermoxidative) radiolysis and photolysis (UV-sensitive materials only) of organics; radiation-induced oxidation, and moisture intrusion	Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements	VI.A-2	3.6.1-02	A

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
3	Non-Environment-ally Qualified Sensitive, High-Voltage, Low-Level Signal Instrument Cables and Connections	Conduct electricity	Various Organic Polymers	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/ degradation of organics (thermal/ thermoxidative) radiolysis and photolysis (UV-sensitive materials only) of organics; radiation-induced oxidation, and moisture intrusion	Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits	VI.A-3	3.6.1-03	A

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
4	Non-Environment-ally Qualified Medium-Voltage Power Cables	Conduct electricity	Various Organic Polymers	Adverse localized environment caused by exposure to moisture and voltage	Localized damage and breakdown of insulation leading to electrical failure / moisture intrusion, water trees	Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 EQ Requirements	VI.A-4	3.6.1-04	A
5	Cable Connections (metallic parts)	Conduct electricity	Various Metals	Air with borated water leakage	Corrosion of connector surfaces / intrusion of borated water	Boric Acid Corrosion	VI.A-5	3.6.1-05	A

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
6	Fuse Holders (insulation)	Insulation (and support)	Various Organic Polymers	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/ degradation of organics (thermal/ thermoxidative) radiolysis and photolysis (UV-sensitive materials only) of organics; radiation-induced oxidation, and moisture intrusion	Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements	VI.A-6	3.6.1-02	A

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
7	Fuse Holders (insulation)	Insulation (and support)	Various Organic Polymers	Air-indoor uncontrolled	None	None	VI.A-7	3.6.1-14	A
8	High-Voltage Insulators	Insulation (and support)	Porcelain, Malleable Iron, Galvanized Metal, Cement	Air-outdoor	None	None	VI.A-9	3.6.1-11	I 0601
9	High-Voltage Insulators	Insulation (and support)	Porcelain, Malleable Iron, Galvanized Metal, Cement	Air-outdoor	None	None	VI.A-10	3.6.1-11	I 0602
10	Switchyard Bus and Connections	Conduct electricity	Aluminum, Galvanized Steel	Air-outdoor	None	None	VI.A-15	3.6.1-12	I 0603
11	Transmission Conductors and Connections	Conduct electricity	Aluminum, Galvanized Steel, Stainless Steel	Air-outdoor	None	None	VI.A-16	3.6.1-12	I 0604

Table 3.6.2-1 Aging Management Review Results - Electrical Component Commodity Groups									
Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801, Volume 2 Item	Table 1 Item	Notes
12	Electrical Equipment Subject to 10 CFR 50.49 EQ Requirements	Conduct electricity	Various organic polymers and metallic materials	Adverse localized environment caused by heat, radiation, oxygen, moisture, or voltage	Various degradations / various mechanisms	TLAA	VI.B-1	3.6.1-01	A

Generic 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 item 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:	
0601	Degradation of insulator quality due to the deposition of contaminants on the insulator surface is not an applicable aging effect for Davis-Besse. See Section 3.6.2.2.2 for evaluation.
0602	Loss of material due to wear is not an applicable aging effect for the in-scope high-voltage insulators at Davis-Besse. See Section 3.6.2.2.2 for evaluation.
0603	For the switchyard bus and connections, no aging effects are identified that require aging management - refer to Section 3.6.2.2.3 for evaluation. An aging management program is not required for the switchyard bus and connections that are within the scope of license renewal.
0604	The transmission conductors within the license renewal scope are those that connect Start-up transformers 01 and 02 to circuits in the plant switchyard. These segments of transmission conductor and associated connections do not exhibit significant aging mechanisms or effects. An aging management program is not required for the segment of transmission conductor that is within the scope of license renewal. See Section 3.6.2.2.3 for details.

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4.0 TIME-LIMITED AGING ANALYSES

The License Renewal Rule, 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," governs the issuance of renewed operating licenses for nuclear power plants and includes requirements for the performance of an integrated plant assessment and for the review of time-limited aging analyses (TLAAs). The results of the integrated plant assessment and TLAA evaluations form the technical bases upon which the License Renewal Application for Davis-Besse Nuclear Power Station, Unit 1 (Davis-Besse) is built.

10 CFR 54.21(c) requires a list of TLAAs as part of the application for a renewed license. 10 CFR 54.21(c)(2) requires a list of current exemptions to 10 CFR Part 50 based on time-limited aging analyses as part of the application for a renewed license.

§54.21 Contents of application -- technical information.

(c) An evaluation of time-limited aging analyses.

1. *A list of time-limited aging analyses, as defined in §54.3, must be provided. The applicant shall demonstrate that -*
   - *(i) The analyses remain valid for the period of extended operation;*
   - *(ii) The analyses have been projected to the end of the period of extended operation; or*
   - *(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.*
2. *A list must be provided of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in §54.3. The applicant shall provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.*

This section (Section 4) describes the TLAAs and Exemptions applicable to Davis-Besse in accordance with 10 CFR 54.

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4.1 TIME-LIMITED AGING ANALYSES AND EXEMPTIONS

4.1.1 TIME-LIMITED AGING ANALYSES

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 as 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.

The major emphasis in the License Renewal Rule (10 CFR 54) is that the current licensing basis (CLB) must be maintained during the period of extended operation. By definition, TLAAs are contained or incorporated by reference in the CLB. Therefore, the documentation that describes the CLB at Davis-Besse was searched to identify TLAAs.

The CLB documentation searched to identify potential TLAAs includes the following:

- Updated Safety Analysis Report (USAR)
- Fire Hazards Analysis Report (incorporated by reference in the USAR)
- Quality Assurance program
- In-Service Inspection program
- In-Service Testing program
- Operating License (including Technical Specifications)
- Exemptions and Inspection Relief Requests
- Docketed Licensing Correspondence
- Design Calculations and Reports (incorporated in the CLB, e.g., by reference)

Industry documents that list generic TLAAs were also consulted to ensure the completeness of the plant-specific evaluations. These documents include:

- NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Revision 1
- NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule," Revision 6
- EPRI Report TR-105090, "Guidelines to Implement the License Renewal Technical Requirements of 10 CFR 54 for Integrated Plant Assessments and Time-Limited Aging Analyses" [Reference 4.8-9]
- License renewal applications for Babcock & Wilcox (B&W) pressurized water reactor (PWR) designs and those other PWR designs which utilize B&W reactor vessels, and the associated safety evaluation reports
- Recent license renewal applications for PWRs

Table 4.1-1 provides a summary listing of the Davis-Besse TLAAs along with reference to the section where each TLAA is reviewed.

Table 4.1-2 provides a summary of the results of a review of potential TLAAs identified in NUREG-1800 Tables 4.1-2 and 4.1-3, and identifies the section where each TLAA is reviewed, if applicable.

4.1.2 EXEMPTIONS

In order to identify exemptions in effect for Davis-Besse, a keyword search was conducted of the following documents:

- USAR
- Fire Hazards Analysis Report
- Operating License (including Technical Specifications)
- Initial Davis-Besse Safety Evaluation Report (NUREG-1036, including Supplement 1)
- Docketed Licensing Correspondence

This review involved a search to identify exemptions that were granted pursuant to 10 CFR 50.12, as well as those related to 10 CFR 50 Appendix R. The search criteria utilized key words and phrases, including: "50.12," "deviation," "exception," "exempt," "exemption," and "relief request". As a result of the review, there were no exemptions identified as granted pursuant to 10 CFR 50.12 and in effect that are based on a TLAA.

Table 4.1-1 Time-Limited Aging Analyses

Results of TLAA Evaluation by Category	54.21(c)(1) Paragraph	LRA Section
Reactor Vessel Neutron Embrittlement		4.2
Neutron Fluence	Not a TLAA	4.2.1
Upper-Shelf Energy	(ii)	4.2.2
Pressurized Thermal Shock	(ii)	4.2.3
Pressure-Temperature Limits	(iii)	4.2.4
Low Temperature Overpressure Protection Limits	(iii)	4.2.5
Intergranular Separation (Underclad Cracking) – reactor vessel shell	(ii)	4.2.6
Intergranular Separation (Underclad Cracking) – reactor vessel head	Not a TLAA	4.2.6
Reduction in Fracture Toughness of Reactor Vessel Internals	(iii)	4.2.7
Metal Fatigue		4.3
Class 1 Fatigue		4.3.2
Reactor Vessel	(iii)	4.3.2.2.1
Reactor Vessel internals – low cycle fatigue	(iii)	4.3.2.2.2.1
Reactor Vessel internals – flow induced vibration	(i)	4.3.2.2.2.2
Incore Instrumentation Nozzles and Surveillance Capsule Holder Tubes – flow induced vibration	(ii)	4.3.2.2.2.3
Control rod drive housings	(iii)	4.3.2.2.3
Reactor coolant pump casings	(iii)	4.3.2.2.4
Pressurizer	(iii)	4.3.2.2.5
Once Through Steam Generators (OTSGs)	(iii)	4.3.2.2.6.1
OTSGs tube sleeves	(i)	4.3.2.2.6.2
OTSGs AFW modification	(iii)	4.3.2.2.6.3
OTSGs tubes and tube stabilizers – flow induced vibration	(ii)	4.3.2.2.6.4
Class 1 piping	(iii)	4.3.2.3.1
Class 1 valves	Not a TLAA	4.3.2.3.2
High Energy Line Break Postulations	(iii)	4.3.2.3.3

Table 4.1-1 Time-Limited Aging Analyses (continued)

Results of TLAA Evaluation by Category	54.21(c)(1) Paragraph	LRA Section
Non-class 1 Fatigue		4.3.3
Non-class 1 Piping and In-Line Components	(i)	4.3.3.1
Non-class 1 Major Components	Not a TLAA	4.3.3.2
Effects of reactor water environment on fatigue	(iii)	4.3.4
Environmental Qualification of Electrical Equipment	(iii)	4.4
Concrete Containment Tendon Prestress	Not a TLAA	4.5
Containment Fatigue		4.6
Containment Vessel	(i)	4.6.1
Containment Penetrations	Not a TLAA	4.6.2
Permanent Reactor Cavity Seal Plate (also known as Permanent Canal Seal Plate (PCSP))	(iii)	4.6.3
Other Plant-Specific Time-Limited Aging Analyses		4.7
Leak-Before-Break	(iii)	4.7.1
Metal Corrosion Allowance for Pressurizer Instrument Nozzles	(ii)	4.7.2
Reactor Vessel Thermal Shock due to Borated Water Storage Tank water injection	(ii)	4.7.3
High Pressure Injection / Makeup Nozzle Thermal Sleeves – life prediction	(iii)	4.7.4
RCS Loop 1 Cold Leg drain line weld overlay repair	(iii)	4.7.5.1
OTSG 1-2 flaw evaluations	(iii)	4.7.5.2

Table 4.1-2 Review of Generic TLAAs Listed in NUREG-1800

NUREG-1800 Generic TLAAs	Applicable to Davis-Besse (Y/N?)	LRA Section
NUREG-1800, Table 4.1-2		
Reactor vessel neutron embrittlement	Yes	4.2
Concrete containment tendon prestress	No – Davis-Besse does not have pre-stressed containment tendons	4.5
Metal fatigue	Yes	4.3
Environmental qualification of electrical equipment	Yes	4.4
Metal corrosion allowance	Yes	4.7.2
Inservice flaw growth analyses that demonstrate structure stability for 40 years	Yes	4.7.5
Inservice local metal containment corrosion analyses	No – No TLAA identified	--
High-energy line-break postulation based on fatigue cumulative usage factor	Yes	4.3.2.3.3
NUREG-1800, Table 4.1-3		
Intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic stainless steel cladding	Yes	4.2.6
Low-temperature overpressure protection (LTOP) analyses	Yes	4.2.5
Fatigue analysis for the main steam supply lines to the turbine-driven auxiliary feedwater pump turbines	Yes	4.3.3.1
Fatigue analysis of the reactor coolant pump flywheels	No – No TLAA identified	--
Fatigue analysis of the polar crane	No – No TLAA identified	--
Flow-induced vibration endurance limit for the reactor vessel internals	Yes	4.3.2.2.2
Transient cycle count assumptions for the reactor vessel internals	Yes	4.3.2.2.2
Ductility reduction of fracture toughness for the reactor vessel internals	Yes	4.2.7
Leak-Before-Break	Yes	4.7.1
Fatigue analysis for the containment liner plate	No – Davis-Besse does not have a containment liner plate	--
Containment penetration pressurization cycles	No – No TLAA identified	4.6
Reactor vessel circumferential weld inspection relief (BWR)	No – Davis-Besse is a PWR.	--

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4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

Neutron embrittlement is the term used to describe changes in mechanical properties of reactor vessel materials that result from exposure to fast neutron flux, energy greater than 1.0 million electron volts (E > 1.0 MeV), within the vicinity of the reactor core, called the beltline region. The most pronounced material change is a reduction in fracture toughness. As fracture toughness decreases with cumulative fast neutron exposure, the material's resistance to crack propagation decreases. The rate of neutron exposure is neutron flux ($n/cm^2/sec$) and the cumulative neutron exposure over time is neutron fluence (n/cm^2).

Fracture toughness can be expressed in terms of the reference temperature for nil-ductility transition (RT_{NDT}). RT_{NDT} is the temperature above which the material behaves in a ductile manner and below which the material behaves in a brittle manner. As fluence increases, RT_{NDT} increases. This means higher temperatures are required for the material to continue to act in a ductile manner. Determining the projected reduction in fracture toughness as a function of fluence affects the following analyses used to support the operation of Davis-Besse:

- Neutron Fluence
- Upper-Shelf Energy
- Pressurized Thermal Shock
- Pressure-Temperature Limits
- Low-Temperature Overpressure Protection Limits
- Intergranular Separation (Underclad Cracking)
- Reduction in Fracture Toughness of Reactor Vessel Internals

These analyses include time dependent parameters that must be investigated with respect to the fracture toughness of Davis-Besse reactor vessel materials. USAR Table 5.2-15 gives the properties of reactor vessel materials, including identification of the beltline materials.

10 CFR 50.60 requires that fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary be satisfied in accordance with 10 CFR 50, Appendix G and Appendix H. 10 CFR 50, Appendix G specifies upper-shelf energy and pressure-temperature limits that account for neutron irradiation effects for the life of the plant. 10 CFR 50, Appendix H requires a reactor vessel material surveillance program; the Reactor Vessel Surveillance Program is discussed in Appendix B.

The following sections address reactor vessel embrittlement analyses, and related topics, for extended operation of the plant. The data differs somewhat from the information currently in the NRC's Reactor Vessel Integrity Database (RVID2). This later data have either been previously submitted to the NRC, or are submitted herein, as described in the subsections below.

4.2.1 NEUTRON FLUENCE

4.2.1.1 Effective Full Power Years (EFPY) Projection

End-of-life fluence is based on a projected value of EFPY over the licensed life of the plant. For the current term of operation, end-of-life for Davis-Besse is 40 years and reactor vessel embrittlement calculations are based on fluence projections at 32 EFPY. The Davis-Besse operating license was issued in April 22, 1977 and the plant lifetime capacity factor through April 2006 is 0.622. The plant capacity factor between 2006 and 2008 is ~0.90. Assuming a plant capacity factor of 98.5% beyond 2008, Davis-Besse is expected to conservatively accrue approximately 50.3 EFPY by April 22, 2037. Therefore, projection of fluence based on 52 EFPY at 60 years is conservative for the period of extended operation for Davis-Besse. In 1977 Davis-Besse was licensed for a maximum thermal power of 2772 MWt. In 2008 the maximum thermal power was increased to 2817 MWt through a measurement uncertainty recapture power uprate. However, calculation of EFPY is independent of plant maximum thermal power.

4.2.1.2 Fluence Projection

The fluence analysis methodology from BAW-2241P-A [Reference 4.8-6] was used to calculate the fast neutron fluence (E > 1.0 MeV) of the reactor vessel welds and forgings of interest. The fast neutron fluence at each location was calculated in accordance with the requirements of U.S. Nuclear Regulatory Guide 1.190.

Fluence results were calculated for Cycles 13-14 irradiation using a computer model that extends from below the core to the vessel mating surface. The sum of the end of cycle (EOC) 12 and Cycles 13-14 fluence results in the EOC 14 cumulative fluence. This data was benchmarked against cavity dosimetry data for Cycles 13-14. To extrapolate the fluence values to end of life, Cycle 15 design information was utilized to develop flux projections at each location. These Cycle 15 flux values were used to extrapolate the EOC 14 fluence to 52 EFPY assuming 100% power at 2,817 MWt and a partial low leakage core design whereby High Thermal Performance fuel assemblies (a total of 12) were introduced on the periphery.

A summary of all inner surface fluence values over 1E+17 n/cm^2 at 52 EFPY for the Davis-Besse reactor vessel is shown in Table 4.2-1.

4.2.1.3 Beltline Evaluation

10 CFR 50.61 defines the reactor vessel beltline as the region of the reactor vessel (shell materials including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most controlling material with regard to radiation damage. 10 CFR 50, Appendix G, Section II.F identifies the beltline as the regions of the reactor vessel "that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most controlling material with regard to radiation damage."

The Davis-Besse beltline for the first 40 years of operation includes the nozzle belt forging (ADB 203), the nozzle belt forging to upper shell forging circumferential weld (WF-232/233), the upper shell forging (AKJ 233), the upper shell forging to lower shell forging circumferential weld (WF-182-1), and the lower shell forging (BCC 241).

For the period of extended operation, the beltline will include all items with 52 EFPY surface fluence greater than 1.0E+17 n/cm^2, as shown in Table 4.2-1. Upper-shelf energy (USE), reference temperature for pressurized thermal shock (RT_{PTS}) and adjusted reference temperature (ART) values are provided in Table 4.2-2, Table 4.2-3, and Table 4.2-4. The limiting weld with regard to USE, ART, and RT_{PTS} is the upper shell to lower shell weld, WF-182-1, as was the case at 40 years. The limiting forging with regard to ART and RT_{PTS} is lower shell forging BCC 241 as was the case at 40 years. Both of these materials are included in the Reactor Vessel Surveillance Vessel Program and no additional materials are required for irradiation and testing.

Disposition:	Not a TLAA	Neutron fluence is an assumption used in various neutron embrittlement TLAAs evaluated below.

Table 4.2-1 Fluence Values at 52 EFPY

Reactor Vessel Location		Material ID. / (Heat Number)	52 EFPY Peak Fluence (inside wetted surface unless otherwise noted) (n/cm^2) (E>1MeV)
Forgings - Top to Bottom of Reactor Vessel			
1	Reactor Vessel Closure Flange Forging	NA	2.57E+16[1]
2	Reactor Vessel Inlet Nozzle Forgings	BSS 270 / (A13315)	1.17E+17
3	Reactor Vessel Outlet Nozzle Forgings	ATS 239 / (2V1520)	2.30E+17
4	Nozzle Belt Forging	ADB 203 / (123Y317)	2.29E+18
5	Upper Shell Forging	AKJ 233 / (123X244)	1.69E+19
6	Lower Shell Forging	BCC 241 / (5P4086)	1.70E+19
7	Dutchman Forging	122Y384VA1 / (122Y384VA1)	2.33E+17
8	Lower Head	NA	3.86E+16[2]
Welds - Top to Bottom of Reactor Vessel			
9	Reactor Vessel Flange to Nozzle Belt Forging Circumferential Weld	NA	2.57E+16[1]
10	Nozzle Belt Forging to Bottom of Reactor Vessel Inlet Nozzle Forging	WF-233 / WF-232 (T29744 / 8T3914)	1.17E+17
11	Nozzle Belt Forging to Bottom of Reactor Vessel Outlet Nozzle Forging	WF-233 (T29744)	2.30E+17
12	Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (Inside 9%), Outside 91% is WF-233	WF-232 (9%) / (8T3914) WF-233 (91%) / (T29744)	2.29E+18
13	Upper Shell Forging to Lower Shell Forging Circumferential Weld	WF-182-1 / (821T44)	1.69E+19
14	Lower Shell Forging to Dutchman Forging Circumferential Weld (inside 12%), Outside 88% is WF-233	WF-232 (12%) / (8T3914) WF-233 (88%) / (T29744)	2.33E+17
15	Dutchman Forging to Lower Head Circumferential Weld	WF-182 / (821T44)	3.86E16[2]

[1] Peak fluence is located at the outer diameter of the reactor vessel at this location. Location is conservatively chosen as nozzle belt forging (NBF) to top of inlet nozzle forging weld.

[2] Peak fluence is located at the outer diameter of the reactor vessel at this location. Location is conservatively chosen as the dutchman to lower head weld.

4.2.2 UPPER-SHELF ENERGY

4.2.2.1 Background

10 CFR 50 Appendix G requires the USE of the reactor vessel beltline materials to be no less than 50 ft-lb at all times during plant operation, including the effects of neutron radiation. If USE cannot be shown to remain above this limit, then an equivalent margin analysis must be performed to show that the margins of safety against fracture are equivalent to those required by Appendix G of ASME Section XI.

Initial (unirradiated) USE values for the Davis-Besse reactor vessel base metal are recorded in USAR Table 5.2-15. As no initial USE is available for the beltline welds (Linde80 welds), operation for 32 EFPY was justified based on an equivalent margins analysis (fracture mechanics analysis) [References 4.8-2 and 4.8-3].

USE was re-evaluated for the measurement uncertainty recapture power uprate [Reference 4.8-3]. An equivalent margin analysis was performed for the controlling weld, WF-182-1. The equivalent margin analysis demonstrated that the controlling reactor vessel beltline weld satisfies the acceptance criteria of ASME Section XI, Appendix K. An equivalent margin analysis was not required for the reactor vessel beltline forging materials since all applicable materials were predicted to have upper-shelf Charpy energy levels in excess of 50 ft-lb at 32 EFPY.

4.2.2.2 USE Projections

For license renewal, the initial USE values are projected to 52 EFPY using Regulatory Guide 1.99, Revision 2, Position 1.2. Position 2.2, use of surveillance data, was also used for weld WF-182-1 and lower shell forging BCC 241. Note that since there is only one capsule that has been tested that includes upper shell forging (AKJ 233), there is insufficient data to conduct surveillance data credibility assessments relative to Regulatory Guide 1.99, Revision 2 for forging AKJ 233. Fluence is from Table 4.2-1. All locations are above 50 ft-lb with the exception of weld WF-182-1. The predicted USE is conservatively calculated based on a ¼ T fluence of 1.0E+18 n/cm^2 (the lowest fluence in Regulatory Guide 1.99, Revision 2, Figure 2), for the RV inlet nozzle forging and attachment weld, RV outlet nozzle forging and attachment weld, and dutchman forging and weld that connects the lower shell forging to the dutchman forging. The results are presented in Table 4.2-2.

4.2.2.3 Equivalent Margins Analyses

The limiting Davis-Besse reactor vessel beltline weld WF-182-1 is the only 60-year (52 EFPY) beltline location with a projected Charpy impact energy level below 50 ft-lbs. The fracture mechanics evaluation of weld WF-182-1 at Davis-Besse was extended from 40-years (32 EFPY) to 60-years (52 EFPY) based on the projected 52 EFPY neutron fluence values. The analysis demonstrates that the limiting reactor vessel beltline weld at Davis-Besse satisfies the ASME Code requirements of Appendix K for ductile flaw extensions and tensile stability using projected upper-shelf Charpy impact energy levels for the weld material at 52 EFPY.

The 52 EFPY fracture mechanics analysis addresses ASME Levels A, B, C, and D Service Loadings and is performed using the procedures and acceptance criteria in Appendix K to Section XI of the ASME Code. Levels C and D Service Loadings are evaluated using the one-dimensional, finite element, thermal and stress models and linear elastic fracture mechanics methodology of the PCRIT computer code to determine stress intensity factors for a worst case pressurized thermal shock transient.

In order to extend the 32 EFPY analysis to 52 EFPY, the calculations that are time dependent were identified and updated accordingly. It was confirmed that the analytical methodology and applied loadings have not changed. Key points of the analysis are summarized below.

Initial RT_{NDT} was revised from +2 °F to -80.2 °F and margin from +56 °F to +59 °F (Revised initial RT_{NDT} and margins for weld WF-182-1 were obtained from BAW-2308, Revision 1-A)[1]. All other mechanical properties are unchanged. The ASME transition region fracture toughness curve K_{Ic}, used to define the beginning of the upper-shelf toughness region, is indexed by the initial RT_{NDT} of the weld material. The existing transition region fracture toughness curve evaluation is conservative for 52 EFPY since the initial RT_{NDT} has decreased.

Projected inside surface fluence at 52 EFPY has increased, affecting the J-integral resistance of the material. Fluence at the crack tip is determined using the attenuation equation from Regulatory Guide 1.99, Revision 2.

The Hot Leg Large Break Loss of Coolant Accident (LOCA) is the limiting transient at 32 EFPY and 52 EFPY since it most closely approaches the KJc limit of the weld. In the upper-shelf toughness range, the KI curve is closest to the lower bound KJc curve at 5.60 minutes into the transient. This time is selected as the critical time in the transient at which to perform the flaw evaluation for Levels C and D Service Loadings.

Summary of Results for Level A, B, C and D Service Loadings at 52 EFPY

Evidence that the ASME Code, Section XI, Appendix K acceptance criteria have been satisfied for Levels A and B Service Loadings is provided by the following:

---

[1] FENOC submitted a request (FENOC Letter L-09-225 [Reference 4.8-16]) for exemption to use an alternate method, as described in approved-topical report BAW-2308, Revision 1-A, for determining initial RT_{NDT} values of the Linde 80 weld materials present in the beltline region of the Davis-Besse reactor pressure vessel.

(1) With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J_1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. ($J_{0.1}$). The ratio $J_{0.1}/J_1$ = 3.69 which is significantly greater than the required value of 1.0.

(2) With factors of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect.

Evidence that the ASME Code, Section XI, Appendix K acceptance criteria have been satisfied for Levels C and D Service Loadings is provided by the following:

(1) With a factor of safety of 1.0 on loading, the applied J-integral (J_1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. ($J_{0.1}$). The ratio $J_{0.1}/J_1$ = 2.16, which is significantly greater than the required value of 1.0.

(2) With a factor of safety of 1.0 on loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slopes of both the lower bound and mean J-R curves at the points of intersection.

(3) Flaw growth is stable at much less than 75% of the vessel wall thickness. It has also been shown that the remaining ligament is sufficient to preclude tensile instability by a large margin.

The limiting reactor vessel beltline weld at Davis-Besse satisfies the ASME Code requirements of Appendix K for ductile flaw extensions and tensile stability using projected upper-shelf Charpy impact energy levels for the weld material at 32 EFPY and 52 EFPY.

Disposition: 10 CFR 54.21(c)(1)(ii) Reactor vessel USE and equivalent margin analyses have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Table 4.2-2 USE Values at 52 EFPY for Davis-Besse Reactor Vessel Beltline Materials
(RG 1.99 Position 1.2, Unless Otherwise Noted)

Item	Material Type	Material ID.	USE @ 52 EFPY at ¼T, ft-lbs	¼T Neutron Fluence, n/cm², E>1MeV	Unirradiated USE, ft-lbs	% Drop in USE @ EOL 1/4T	Cu, %
Reactor Vessel Forgings							
Reactor Vessel Inlet Nozzle Forgings	A508-2	BSS 270	51.2	1.00E+18[1]	61.5	16.8	0.20
Reactor Vessel Outlet Nozzle Forgings	A508-2	ATS 239	64.6	1.00E+18[1]	75.5	14.5	0.16
Nozzle Belt Forging	A508-2	ADB 203	123.2	1.34E+18	134	8.1	0.04
Upper Shell Forging	A508-2	AKJ 233	125.3	9.87E+18	144	13	0.04
Lower Shell Forging	A508-2	BCC 241	105	9.93E+18	118	11	0.02
			95.6[2]	9.93E+18	118	19.0[2]	0.02
Dutchman Forging	A508-2	122Y384VA1	96.4	1.00E+18[1]	109	11.6	0.11
Reactor Vessel Welds							
Nozzle Belt Forging to Bottom of Reactor Vessel Inlet Nozzle Forging	Linde 80	WF-233 / 232	55.8	1.00E+18[1]	70	20.3	0.21
Nozzle Belt Forging to Bottom of Reactor Vessel Outlet Nozzle Forging	Linde 80	WF-233	55.8	1.00E+18[1]	70	20.3	0.21
Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (inner 9%)	Linde 80	WF-232	NA[3]	NA[3]	70	NA[3]	0.18
Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (outer 91%)	Linde 80	WF-233	54.8	1.34E+18	70	21.7	0.21
Upper Shell Forging to Lower Shell Forging Circumferential Weld	Linde 80	WF-182-1	43.5 43.4[2]	9.87E+18 9.87E+18	70 70	37.9 38.0[2]	0.24 0.24
Lower Shell Forging to Dutchman Forging Circumferential Weld (inner 12%)	Linde 80	WF-232	NA[3]	NA[3]	70	NA[3]	0.18
Lower Shell Forging to Dutchman Forging Circumferential Weld (outer 88%)	Linde 80	WF-233	55.8	1.00E+18[1]	70	20.3	0.21

1 In accordance with Regulatory Guide 1.99, Revision 2, Figure 2, the lowest 1/4T fluence is 1E+18 n/cm². The predicted USE is conservatively calculated based on a fluence of 1E+18 n/cm² for this material, which is higher than the projected peak fluence at 52 EFPY for this location (see Table 4.2-4).
2 Regulatory Guide 1.99 Position 2.2, Use of Surveillance data
3 Location does not extend to 1/4T

4.2.3 PRESSURIZED THERMAL SHOCK

10 CFR 50.61(a)(2) defines pressurized thermal shock (PTS) as an event or transient in a pressurized water reactor causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. 10 CFR 50.61(b)(2) defines screening criteria for embrittlement of reactor vessel materials in pressurized water reactors, and required actions if the screening criteria are exceeded. The screening criteria are based on the RT_{PTS}. The screening criterion for circumferential welds is 300°F maximum, and the screening criterion for forgings is 270°F maximum. If the projected RT_{PTS} values remain below the applicable screening temperature, no corrective action is required.

For license renewal, a 52 EFPY RT_{PTS} evaluation was performed for the reactor vessel beltline materials. In accordance with 10 CFR 50.61, RT_{PTS} values were calculated by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT} plus a margin to cover uncertainties, as prescribed by Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials". The predicted radiation induced ΔRT_{NDT} was calculated using the 52 EFPY neutron fluence at the clad-low alloy steel interface. Table 4.2-3 includes 52 EFPY RT_{PTS} values for all 60-year beltline materials using Position 1.1. Surveillance data was not used since there are not two credible sets of RT_{PTS} surveillance data for any Davis-Besse location. Initial RT_{NDT} and margins for welds WF-182-1 and WF-232 (Nozzle Belt Forging to Upper Shell Forging Circumferential Weld) were obtained from BAW-2308, Revision 1-A [Reference 4.8-14][2]. Using Regulatory Guide 1.99 Revision 2, Table 1, the Chemistry Factor for weld WF-232 is 157.3. However, when initial RT_{NDT} values from BAW-2308, Revision 1-A are used, the Chemistry Factor cannot be less than 167.0. Thus the Chemistry Factor shown in Table 4.2-3 for weld WF-232 is 167.0.

All RT_{PTS} values are below the screening criteria at 60 years. The beltline weld WF-182-1 is the limiting material relative to RT_{PTS}.

Disposition: 10 CFR 54.21(c)(1)(ii) Reactor vessel RT_{PTS} TLAAs have been projected to the end of the period of extended operation.

[2] FENOC submitted a request (FENOC Letter L-09-225 [Reference 4.8-16]) for exemption to use an alternate method, as described in approved-topical report BAW-2308, Revision 1-A, for determining initial RT_{NDT} values of the Linde 80 weld materials present in the beltline region of the Davis-Besse reactor pressure vessel.

Table 4.2-3 RT_{PTS} Values for 52 EFPY for Davis-Besse Reactor Vessel Beltline Materials
(RG 1.99 Position 1.1)

Item	Material ID.	RT_{PTS} / Acceptance Criterion, °F	Fluence at clad-low alloy steel interface, n/cm^2, E>1MeV	RT_{NDT}(u), °F	ΔRT_{NDT}, °F	Fluence Factor	Chemistry Factor	Margin, °F	Copper (wt%)	Nickel (wt%)
Reactor Vessel Forgings										
Reactor Vessel Inlet Nozzle Forging	BSS 270	86.2 / 270	1.14E+17	3	18.5	0.120	154.5	64.7	0.20	0.71
Reactor Vessel Outlet Nozzle Forging	ATS 239	91.7 / 270	2.24E+17	3	22.7	0.184	123.0	66.0	0.16	0.80
Nozzle Belt Forging	ADB 203	81.2 / 270	2.27E+18	50	15.6	0.600	26.0	15.6	0.04	0.68
Upper Shell Forging	AKJ 233	79.4 / 270	1.68E+19	20	29.7	1.143	26.0	29.7	0.04	0.77
Lower Shell Forging	BCC 241	95.7 / 270	1.68E+19	50	22.9	1.143	20.0	22.9	0.02	0.81
Dutchman Forging	122Y384VA1	80.8 / 270	2.28E+17	3	14.2	0.186	76.1	63.6	0.11	0.74
Reactor Vessel Welds										
Nozzle Belt Forging to Bottom of Reactor Vessel Inlet Nozzle Forging	WF-233 / 232	60.1 / 270	1.14E+17	-5	20.6	0.120	172.3	44.5	0.21	0.65
Nozzle Belt Forging to Bottom of Reactor Vessel Outlet Nozzle Forging	WF-233	77.4 / 270	2.24E+17	-5	31.8	0.184	172.3	50.6	0.21	0.65
Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (ID 9%)	WF-232	118.3 / 300	2.27E+18	-47.6[1]	100.2	0.600	167[1]	65.7[1]	0.18	0.62
Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (OD 91%)	WF-233	NA[3]	NA[3]	-5	NA[3]	NA[3]	NA[3]	NA[3]	0.21	0.65
Upper Shell Forging to Lower Shell Forging Circumferential Weld	WF-182-1	182.2[2]/ 300	1.68E+19	-80.2[1]	203.4	1.143	178.0	59.0[1]	0.24	0.63
Lower Shell Forging to Dutchman Forging Circumferential Weld (ID 12%)	WF-232	73.4 / 300	2.28E+17	-5	29.3	0.186	157.3	49.1	0.18	0.62
Lower Shell Forging to Dutchman Forging Circumferential Weld (OD 88%)	WF-233	NA[3]	NA[3]	-5	NA[3]	NA[3]	NA[3]	NA[3]	0.21	0.65

1 - Value based on BAW-2308 Rev. 1A
2 - Limiting location
3 - Location does not extend to the clad base interface

4.2.4 PRESSURE-TEMPERATURE LIMITS

10 CFR 50 Appendix G requires the establishment of pressure and temperature (P-T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials. Appendix G mandates the use of the ASME Section III, Appendix G to determine the stresses and fracture toughness at locations within the reactor coolant pressure boundary.

One measure of the fracture toughness of a material is the reference temperature for nil-ductility transition (RT_{NDT}). RT_{NDT} will increase with cumulative exposure to neutron irradiation resulting in an ART. This ART is used in the development of P-T limit curves.

Table 4.2-4 includes 52 EFPY ART at the 1/4T and 3/4T locations for all 60-year beltline materials using Regulatory Guide 1.99, Revision 2, Position 1.1. Minimum cladding thickness is 0.125 inches and the vessel low alloy steel thickness for the upper shell and lower shell forgings is 8.44 inches, 8.563 inches for the nozzle belt forging, and 12.0 inches for the inlet and outlet nozzle forgings. Using these vessel wall depths and the neutron fluence at the inner wetted surface of the vessel, the ¼T and ¾T fluence values for the Davis-Besse reactor vessel materials are calculated in accordance with Equation 1 of Regulatory Guide 1.99 Revision 2. Fluence values at the 1/4T and 3/4T locations for the RV inlet and outlet nozzle and associated welds that connect the nozzles to the nozzle belt forging were obtained by adding the attenuation from both the inside and outside surface. Position 2.1 was not used since two sets of credible ART surveillance data were not available. Initial RT_{NDT} and margins for weld WF-182-1 and WF-233 are obtained from BAW-2308, Revision 1-A [Reference 4.8-14].[3]

The current P-T limits, generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99 Revision 2, are valid until 21 EFPY. A revised pressure and temperature limits report will be submitted to the NRC, in accordance with Technical Specification 5.6.4, before Davis-Besse operates beyond 21 EFPY, in accordance with the requirements of 10 CFR 50, Appendix G. The P-T limit curves, as contained in the pressure-temperature limit report and providing the information required by Technical Specification 5.6.4, will be updated as necessary through the period of extended operation as part of the Reactor Vessel Surveillance Program.

Disposition: 10 CFR 54.21(c)(1)(iii) Reactor vessel P-T limits will be managed, as part of the Reactor Vessel Surveillance Program for the period of extended operation.

[3] FENOC submitted a request (FENOC Letter L-09-225 [Reference 4.8-16]) for exemption to use an alternate method, as described in approved-topical report BAW-2308, Revision 1-A, for determining initial RT_{NDT} values of the Linde 80 weld materials present in the beltline region of the Davis-Besse reactor pressure vessel.

Table 4.2-4 ARTs at 52 EFPY for Davis-Besse Reactor Vessel Beltline Materials

(RG 1.99 Position 1.1)

Item	Material ID	ART, °F		Fluence n/cm^2 10E18		$RT_{NDT(u)}$, °F	ΔRT_{NDT}, °F		Fluence Factor		Chem. Factor	Margin, °F		Cu %	Ni %
		¼T	¾ T	¼T	¾T		¼T	¾T	¼T	¾T		¼T	¾T		
Reactor Vessel Forgings															
Reactor Vessel Inlet Nozzle Forgings SA-580 Class 2	BSS 270	78.6	76.1	0.064	0.0494	3	12.4	10.3	0.080	0.066	154.5	63.2	62.8	0.20	0.71
Reactor Vessel Outlet Nozzle Forgings SA-580 Class 2	ATS 239	82.0	76.3	0.119	0.0688	3	15.1	10.4	0.123	0.084	123.0	63.8	62.9	0.16	0.80
Nozzle Belt Forging SA-580 Class 2	ADB 203	74.8	64.8	1.33	0.476	50	12.4	7.4	0.476	0.285	26.0	12.4	7.4	0.04	0.68
Upper Shell Forging SA-580 Class 2	AKJ 233	71.8	57.3	9.89	3.59	20	25.9	18.6	0.997	0.717	26.0	25.9	18.6	0.04	0.77
Lower Shell Forging SA 580 Class 2	BCC 241	89.9	78.8	9.94	3.61	50	20.0	14.4	0.998	0.719	20.0	20.0	14.4	0.02	0.81
Dutchman Forging SA-580 Class 2	122Y384VA1	76.1	70.3	0.136	0.0495	3	10.3	5.1	0.135	0.067	76.1	62.8	62.2	0.11	0.74
Reactor Vessel Welds															
Nozzle Belt Forging to Bottom of Reactor Vessel Inlet Nozzle Forging	WF-233 / 232	50.6	47.5	0.064	0.0494	-5	13.8	11.4	0.080	0.066	172.3	41.7	41.0	0.21	0.65
Nozzle Belt Forging to Bottom of Reactor Vessel Outlet Nozzle Forging	WF-233	61.0	51.6	0.119	0.0688	-5	21.2	14.6	0.123	0.084	172.3	44.7	42.0	0.21	0.65
Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (ID 9%)	WF-232	NA[2]	NA[2]	NA[2]	NA[2]	-5	NA[2]	NA[2]	NA[2]	NA[2]	157.3	NA[2]	NA[2]	0.18	0.62
Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (OD 91%)	WF-233	100.4	67.8	1.34	0.487	-47.6[1]	82.3	49.7	0.478	0.288	172.3	65.7[1]	65.7[1]	0.21	0.65
Upper Shell Forging to Lower Shell Forging Circumferential Weld	WF-182-1	156.2	106.4	9.89	3.59	-80.2[1]	177.4	127.6	0.997	0.717	178.0	59.0[1]	59.0[1]	0.24	0.63
Lower Shell Forging to Dutchman Forging Circumferential Weld (ID 12%)	WF-232	NA[2]	NA[2]	NA[2]	NA[2]	-5	NA[2]	NA[2]	NA[2]	NA[2]	157.3	NA[2]	NA[2]	0.18	0.62
Lower Shell Forging to Dutchman Forging Circumferential Weld (OD 88%)	WF-233	63.9	47.5	0.136	0.0495	-5	23.2	11.5	0.135	0.067	172.3	45.7	41.0	0.21	0.65

1 - BAW-2308 Revision 1 A for initial RT_{NDT} and margin
2 - Location does not extend to ¼ T or ¾ T.

4.2.5 LOW-TEMPERATURE OVERPRESSURE PROTECTION LIMITS

Appendix G of ASME Section XI establishes procedures and limits for Reactor Coolant System (RCS) pressure under low temperature conditions to provide protection against non-ductile failure of the reactor vessel.

Low-temperature overpressure protection (LTOP) is provided in two ways at Davis-Besse.

1. Administrative controls are used to assure protection within the existing pressure-temperature limits when the pressurizer power-operated relief valve and the safety valves are not providing over-pressure protection,
2. A relief valve in the Decay Heat Removal System suction piping is placed into service when the RCS temperature is below 280°F.

The current technical specifications for LTOP are valid through 21 EFPY. These technical specifications used an improved methodology to calculate LTOP limits in accordance with generically approved topical report BAW-10046A [Reference 4.8-8].

Maintaining the LTOP limits in accordance with Appendix G of ASME Section XI, as required by Appendix G of 10 CFR 50, assures that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) LTOP limits will be managed, as part of the Reactor Vessel Surveillance Program, for the period of extended operation.

4.2.6 INTERGRANULAR SEPARATION (UNDERCLAD CRACKING)

Underclad cracking (UCC) refers to intergranular separation in the heat-affected zone of low-alloy steel under austenitic stainless steel cladding in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice, and clad by high-heat-input submerged arc processes. BAW-10013-A [Reference 4.8-7] contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to design fatigue cycles. The flaw growth analysis was performed for a 40 year cyclic loading, and an end-of-life assessment of radiation embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report concluded that the intergranular separations found in B&W vessels would not lead to vessel failure. This report was accepted by the Atomic Energy Commission.

In May 1973, the AEC issued Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." The guide states that intergranular separation "has been reported only in forgings and plate material of SA-508 Class 2 composition made to coarse grain practice when clad using high-deposition-rate welding processes identified as 'high-heat-input' processes such as the submerged-arc wide-strip and the submerged-arc 6-wire processes. Cracking was not observed in clad SA-508 Class 2 materials clad by 'low-heat-input' processes controlled to minimize heating of the base metal. Further, cracking was not observed in clad SA-533 Grade B Class 1 plate material, which is produced to fine grain practice. Characteristically, the cracking occurs only in the grain-coarsened region of the base-metal heat-affected zone at the weld bead overlap." The guide also notes that the maximum observed dimensions of these subsurface cracks is 0.5 inch x 0.165 inch.

The methodology used to evaluate intergranular separations in the Davis-Besse SA-508 Class 2 forgings is consistent with the methodology reported in the update of BAW-10013 included as Appendix C of BAW-2251A [Reference 4.8-15]. The Davis-Besse specific analysis was performed for 60-years using the current fracture toughness information, applied stress intensity factor solutions, and fatigue crack growth correlations for SA-508 Class 2 material.

The analysis was applied to two relevant regions of the RV: the beltline and the nozzle belt. Both axial and circumferential oriented flaws were considered in the evaluation; however, the detailed flaw evaluation was only performed for the bounding axially oriented flaws. All the significant normal and upset condition transients and emergency and faulted condition transients were evaluated in the analysis. The fatigue crack growth analysis considered all the normal and upset condition transients with associated 60-year projected cycles for the period of extended operation.

As provided in Confirmatory Action Letter, Number 3-10-001, FENOC has voluntarily committed to shutdown the Davis-Besse plant no later than October 1, 2011, and replace the RV closure head. Therefore, the current head (purchased from the Midland Plant and installed during the Cycle 13 refueling outage) is not considered in the underclad cracking evaluation. The replacement RV closure head/head flange, to be installed during the October 2011 outage, was fabricated using SA-508 Class 3 material, which is not susceptible to intergranular separations. Therefore, this replacement closure head/head flange is not considered in the underclad cracking evaluation.

An axially oriented, semi-elliptical surface flaw with an initial flaw size of 0.353-inch deep (approximately twice that which has been observed) and 2.12-inch long (approximately four times that which has been observed) with a 6:1 aspect ratio was conservatively assumed at each of the two regions. This is contrasted to the observed flaws which are subsurface with a maximum size of 0.165 inch deep by 0.5 inch long.

For an axially oriented flaw, the limiting location for satisfying the requirements of IWB-3612 is at the lower end of the nozzle belt forging where the thickness transitions from 8.438 to 12.0 inches. The maximum crack growth, considering normal/upset condition transients with associated 60-year projected cycles for the period of extended operation was determined to be 0.043 inches, which results in a final flaw depth of 0.396 inches. The maximum applied stress intensity factor for the normal and upset condition results in a fracture toughness margin of 3.67 which is greater than the acceptance criterion of $\sqrt{10}$ (3.16). The maximum applied stress intensity factor for the emergency and faulted conditions results in a fracture toughness margin of 1.43, which is greater than the acceptance criterion of $\sqrt{2}$ (1.41). Therefore, the postulated underclad cracks in the Davis-Besse reactor vessel are acceptable for continued safe operation through the period of extended operation.

Disposition:	10 CFR 54.21(c)(1)(ii)	For the reactor vessel shell, including the flange, UCC TLAA have been projected to the end of the period of extended operation.
Disposition:	Not a TLAA	The replacement reactor vessel head is not susceptible to UCC.

4.2.7 REDUCTION IN FRACTURE TOUGHNESS OF REACTOR VESSEL INTERNALS

Reduction of fracture toughness of (stainless steel) reactor vessel internals is an aging effect caused by exposure to neutron irradiation. Prolonged exposure to high-energy neutrons results in changes to the mechanical properties, such as an increase in tensile and yield strength, and decreases in ductility and fracture toughness. The extent of loss of fracture toughness is a function of the material, irradiation temperature, and neutron fluence. The reactor vessel internals components most susceptible to reduction in fracture toughness are those nearest to the reactor core.

USAR Appendix 4A describes the detailed stress analysis of the internals under accident conditions for the current term of operation. The analysis shows that although there is some internals deflection, the internals will not fail because the stresses are within established limits. The effect of irradiation on the mechanical properties and deformation limits for the reactor vessel internals was also evaluated for the current term of operation. That analysis concluded that the reactor internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits.

The impact of the measurement uncertainty recapture (MUR) power uprate on the structural integrity of the reactor vessel internals components was evaluated. It was concluded that the temperature changes due to the MUR power uprate are bounded by those used in the existing analyses, and the existing analyses remain valid. As part of MUR uprate, FENOC provided the following commitment:

> "As appropriate, FENOC commits to incorporate recommendations from EPRI's MRP inspection guidelines into the reactor vessel internals program at Davis-Besse Nuclear Power Station, Unit, No. 1."

The disposition of the fracture toughness of reactor vessel internals TLAA for the period of extended operation is to continue the committed PWR Reactor Vessel Internals Program.

Disposition: 10 CFR 54.21(c)(1)(iii) Integrity of the reactor vessel internals will be managed by the PWR Reactor Vessel Internals Program for the period of extended operation.

4.3 METAL FATIGUE

4.3.1 FATIGUE CYCLES

4.3.1.1 Design Transients

ASME Class 1 components are designed to withstand the effects of cyclic loads due to temperature and pressure changes in the reactor system. These cyclic loads are introduced by normal unit load transients, reactor trips, startup and shutdown operations, and earthquakes.

The 14 original design transients for the RCS are found in USAR Table 5.1-8. Over the life of the plant, additional transients have been identified, including analyzed transients for new components and non-RCS components. The design cycles that are significant contributors to fatigue usage are included in the Fatigue Monitoring Program and are provided in Table 4.3-1.

NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," required the re-evaluation of the cyclic fatigue of the Pressurizer Surge Line. Topical Report BAW-2127 and its Supplements [References 4.8-10 through 4.8-13] describe the results of the revised evaluation. As part of this evaluation (Supplement 3 to BAW-2127) the Davis-Besse heatup and cooldown transients were redefined. Other transients were modified to include thermal stratification and striping. The transients and numbers of design cycles are listed in Table 4.3-1.

4.3.1.2 Projected Cycles

The number of cycles accrued to February 2008 were compiled. These accrued cycles were linearly extrapolated to 60 years of operation to determine whether the incurred cycles would remain below the number of design cycles. The results are presented in Table 4.3-1.

Transients 9C, 9D, and 32 are the only transients affecting Class 1 components where the 60-year projected cycles exceed the design cycles, and are discussed in some detail below. Transient 31A affects the non-Class 1 permanent canal seal plate, and is discussed in Section 4.6.3.

<u>Transient 9 (A through D):</u>

Transient 9 originally counted rapid depressurizations of the RCS because of the temperature transients a rapid depressurization would impose on the high pressure injection (HPI)/makeup nozzles. It was recognized that HPI flow testing also caused temperature swings on the HPI nozzles, and cycles of flow testing were added to this event. Today this transient counts HPI flow tests individually for each of the four HPI/makeup nozzles.

Forty (40) cycles of HPI flow testing were analyzed to determine the effect of HPI flow testing on the cumulative usage factor (CUF) of the HPI nozzles. See Section 4.3.2.3.1 below for discussion of the HPI/makeup nozzle CUFs. The analysis of the HPI nozzles determined that the elbowlets in HPI nozzles 1-1 and 1-2 were limited to 13 cycles each, Transients 9A and 9B, respectively. Davis-Besse is currently monitoring these nozzles against a limit of 13 design cycles. Current cycles are at 9 and 8 for nozzles 1-1 and 1-2, respectively. Current test practices do not cycle these nozzles and projections are that the cycles will remain at the current levels for 40 years and for 60 years of operation.

HPI nozzles 2-1 and 2-2 are limited to 40 cycles; Transients 9C and 9D, respectively. Current test practices cycle these nozzles. The 60-year cycle projection for these nozzles exceeds the design cycle number of 40. Because these nozzles may be reanalyzed for other reasons such as the planned modification to replace the nozzle safe ends and thermal sleeves, Davis-Besse will manage fatigue of these nozzles for the period of extended operation rather than reanalyze for the possible additional cycles at this time. Davis-Besse has committed (see Appendix A) to replace the nozzle safe ends and thermal sleeves prior to the period of extended operation rather than reanalyze for the possible additional cycles.

Transient 32

Each remote welded plug installed in the once-through steam generators (OTSGs) is limited to 33 cycles of heatup and cooldown. The 60-year cycle projection for some of these plugs exceeds the design cycle number. Davis-Besse monitors these cycles with the Fatigue Monitoring Program and will ensure action (either a reanalysis of record or a plant modification) is taken before the design number of cycles is reached. Because these plugs may be reanalyzed for other reasons, Davis-Besse will manage fatigue of these plugs for the period of extended operation rather than reanalyze for the possible additional cycles at this time.

The Fatigue Monitoring Program monitors the cycles incurred and assures that action is taken prior to any analyzed numbers of events being exceeded. The Fatigue Monitoring Program has been reviewed for consistency with the USAR and the supporting fatigue analyses.

Table 4.3-1 60-Year Projected Cycles

Program Transient #		*Transient*	*Accrued Cycles To 2/19/2008*	*60-year Projection Cycles*	*Design Cycles*	*Notes*
1	A	Reactor Coolant System (RCS) Heatup (70 to 558.7° F) [USAR Transient # 1]	65	128	240	None
1	B	RCS Cooldown (558.7 to 140° F) [USAR Transient # 1]	64	128	240	As Davis-Besse was operating at the time of the latest cycle count, there is one more heatup than cooldown. To reflect complete cycles, the cooldown projection was raised to match the heatup projection.
2	A	RCS Heatup (532 to 582° F) [USAR Transient # 2]	104	205	1440	None
2	B	RCS Cooldown (582 to 532° F) [USAR Transient # 2]	48	94	1440	None
3		Power Change 8-100% [USAR Transient # 3]	NA	NA	1800	Transients 3 and 4 are not monitored. Davis-Besse is not a load following plant and therefore; transients 3 and 4 could not credibly approach the number of design cycles during the period of extended operation.
4		Power Change 100-8% [USAR Transient # 3]	NA	NA	1800	
5		10% Step Load Increase [USAR Transient # 4]	34	67	8000	None
6		10% Step Load Decrease [USAR Transient # 5]	71	140	8000	None
7	A	Step Load Reduction 100-8% from Turbine Trip [USAR Transient # 6]	4	8	160	None
7	B	Step Load Reduction 100-8% from Electrical Load Rejection [USAR Transient # 6]	2	4	150	None
8	A	Reactor Trip from Low Reactor Coolant Flow [USAR Transient # 7]	2	4	40	None
8	B	Reactor Trip from High Temperature, Pressure, or Power [USAR Transient # 7]	24	47	160	None
8	C	Reactor Trip from High Pressure due to loss of Feedwater [USAR Transient # 7]	13	26	88	None
8	D	Reactor Trip from other [USAR Transient # 7]	56	110	112	None

Table 4.3-1 60-Year Projected Cycles

Program Transient #		Transient	Accrued Cycles To 2/19/2008	60-year Projection Cycles	Design Cycles	Notes
9	A	Rapid RCS Depressurization 1-1 [USAR Transient # 8]	9	9	13	The projection rate of future cycles for Transients 9A - 9D is based on the five-year period from 1/25/2003 to 2/19/2008, to include only the current test methodology. Accrued cycles as of 1/25/2003 for Transients 9A, 9B, 9C and 9D were respectively 9, 8, 17, and 14. This current test methodology does not cycle nozzles 1-1 and 1-2. Therefore the 60-year projection for Transients 9A and 9B is equal to the cycles that occurred before 1/25/2003.
9	B	Rapid RCS Depressurization 1-2 [USAR Transient # 8]	8	8	13	
9	C	Rapid RCS Depressurization 2-1 [USAR Transient # 8]	21	44	40	Transients 9C and 9D, high pressure injection nozzle cycles, are projected to exceed the number of design cycles prior to the end of the period of extended operation. Davis-Besse manages fatigue of these nozzles using the Fatigue Monitoring Program.
9	D	Rapid RCS Depressurization 2-2 [USAR Transient # 8]	19	48	40	
10		Loss of Reactor Coolant Pump without Reactor Trip [USAR Transient # 9]	5	10	20	None
11		Control Rod Withdrawal [USAR Transient # 10]	0	40	40	Transient 11 has not occurred; therefore the mathematical projection is zero. The number of 60-year projected cycles has been set to the number of design cycles to allow for future occurrence.
12	A	Hydro-test – RCS [USAR Transient # 12]	2	4	15	None
12	B	Hydro-test – Secondary	2	4	25	None
13		Steady State Power Variations	NA	NA	Infinite	Steady state power variations are not counted because they are not fatigue significant and the design cycle number is infinite.
14		Control Rod Drop	9	18	40	None
15		Loss of Offsite Power	3	6	40	None
16		Steam Line Failure	0	NA	1	Steam line failure is not considered in fatigue evaluations. Therefore, projected cycles are not provided.
17	A	Steam Generator Boiling Dry from Loss of Feedwater	3	6	20	None
17	B	Steam Generator Boiling Dry from Stuck Turbine Bypass Valve	1	NA	10	Transient 17B is an emergency conditions and is not considered in fatigue evaluations; therefore, it is not necessary to project cycles.
18		Feedwater Temperature Variation (Loss of Feedwater Heater)	0	40	40	Transient 18 has not occurred; therefore the mathematical projection is zero. The number of 60-year projected cycles has been set to the number of design cycles to allow for future occurrence.

Table 4.3-1 60-Year Projected Cycles

Program Transient #		*Transient*	*Accrued Cycles To 2/19/2008*	*60-year Projection Cycles*	*Design Cycles*	*Notes*
19		Feed and Bleed	NA	NA	4000	Feed and bleed is not counted as it is not a fatigue significant event.
20	A	Miscellaneous - Makeup Flow #1	NA	NA	30000	Miscellaneous makeup flow and pressurizer spray flow are not counted as they are not fatigue significant events.
20	B	Miscellaneous - Makeup Flow #2	NA	NA	$4X10^6$	
20	C	Miscellaneous - Pressurizer Spray	NA	NA	20000	
21		Loss of Coolant Accident (LOCA) [USAR Transient # 11]	0	NA	1	Transients 21 is a faulted condition and is not considered in fatigue evaluations. Therefore, projected cycles are not provided.
22	A	Test Transients – High Pressure Injection System [USAR Transient # 12]	NA	NA	40	Transient 22A is not applicable to Davis-Besse. High pressure injection pumps recirculate back to the Borated Water Storage Tank during the High Pressure Injection System Test and therefore, no inventory is added to the Reactor Coolant System.
22	B	Test Transients - Core Flood 1-1 [USAR Transient # 12]	13	26	240	None
22	C	Test Transients - Core Flood 1-2 [USAR Transient # 12]	13	26	240	None
23	A	OTSG - Fill Secondary	NA	NA	240	OTSG fill, flush, and chemical cleaning are not counted as they are not fatigue significant events.
23	B	OTSG - Fill Primary	NA	NA	240	
23	C	OTSG - Flush	NA	NA	40	
23	D	OTSG –Chemical Cleaning	NA	NA	20	
24		Hot Functional Testing	1	1	1	There will be no further Hot Functional Tests; therefore Transient 24 projection is zero additional cycles.
25	A	Pressurizer Heaters	NA	NA	5000	Pressurizer heater cycles are not counted as they are not fatigue events.
25	B	Pressurizer Heaters	NA	NA	20000	
26	A	Pressurizer Code Safeties	0	30	30	Transient 26A has not occurred; therefore the mathematical projection is zero. The number of 60-year projected cycles has been set to the number of design cycles to allow for future occurrence.
26	B	Pressurizer Electromatic Relief >=400° F	49	96	270	None
26	C	Pressurizer Electromatic Relief < 400° F	21	25	25	No cycles have been accrued for Transient 26C in the last 20 years due to plant modifications to keep the loop seal continuously drained and prevent this transient from occurring. Therefore, the number of 60-year projected cycles is set to the number of design cycles.

Table 4.3-1 60-Year Projected Cycles

Program Transient #	Transient	Accrued Cycles To 2/19/2008	60-year Projection Cycles	Design Cycles	Notes
27	Generator Abnormal Frequency	0	NA	1	Generator abnormal frequency is not considered in fatigue evaluations. Therefore, projected cycles are not provided.
28	Maximum Probable Earthquake [USAR Transient # 13]	0	NA	650	Transients 28 is a faulted condition and is not considered in fatigue evaluations. Therefore, projected cycles are not provided.
29	Pressurizer Spray Nozzle and Spray Line Delta Temperature >300° F	5	10	25	None.
30 A	Auxiliary Feedwater Bolted Nozzle 1-1	196.5	387	875	A Reactor Coolant System heatup and cooldown is one transient cycle and bolting/unbolting of the nozzles is one transient cycle for Transients 30A and 30B.
30 B	Auxiliary Feedwater Bolted Nozzle 1-2	224.5	442	875	
31 A	Permanent Canal Seal Plate (Heatup/Cooldown)	7.5	51	50	The permanent canal seal plate was installed on 1/25/2003. Transient 31A is counted from that date. A Reactor Coolant System heatup and cooldown is one transient cycle. Transient 31A is projected to exceed the number of design cycles prior to the end of the period of extended operation. Davis-Besse manages fatigue of this plate using the Fatigue Monitoring Program.
31 B	Permanent Canal Seal Plate (Operating Basis Earthquake)	0	NA	50	Transients 31B is a faulted condition and is not considered in fatigue evaluations. Therefore, projected cycles are not provided.
32	OTSG Welded Plug (limiting plug is Remote Welded Plug 2A 79-68) (Heatup/Cooldown)	17.5	64	33	The limiting plug (remote welded plug 2A) was installed on 5/23/2003. A Reactor Coolant System heatup and cooldown is one transient cycle. Transient 32 is projected to exceed the number of design cycles prior to the end of the period of extended operation. Davis-Besse manages fatigue of these plugs using the Fatigue Monitoring Program.

4.3.2 CLASS 1 FATIGUE

4.3.2.1 Class 1 Background

The specific codes and standards to which systems, structures, and components were designed are listed in USAR Table 3.2-2. The primary code governing design and construction of the Class 1 systems and components is the ASME Boiler and Pressure Vessel Code. The ASME Code requires evaluation of transient thermal and mechanical load cycles and determination of fatigue usage for Class 1 components.

4.3.2.2 Class 1 Vessels, Pumps, and Major Components

The Class 1 components are those components within the scope of the "Class 1" aging management review (see Section 3.1). The Class 1 components evaluated for license renewal include the reactor vessel, the control rod drives, the reactor coolant pumps, the pressurizer, and the steam generators.

Cumulative usage factors for the Class 1 components are calculated based on normal and upset design transient definitions contained in the component design specifications. The design transients used to generate cumulative usage factors for Class 1 components are discussed in Section 4.3.1 above. In accordance with Davis-Besse Technical Specification 5.5.5, the Allowable Operating Transient Cycles Program (Fatigue Monitoring Program) provides controls to track the USAR Section 5 cyclic and transient occurrences to ensure that components are maintained within the design limits.

Fatigue of Class 1 components is managed by the Fatigue Monitoring Program. This program tracks the occurrence of plant transients that affect fatigue. The number of design cycles originally considered in the design fatigue analyses is not a design limit. The design limit for fatigue is the ASME Code allowable CUF of 1.0. The fatigue usage for a component is normally the result of several different thermal transients, coupled with mechanical loads. Exceeding the design number of cycles for one or more transients does not necessarily imply that fatigue usage will exceed the allowable limit.

4.3.2.2.1 Reactor Vessel

The reactor is designed as a Class A vessel in accordance with the ASME Code, Section III, 1968 Edition through Summer 1968 Addenda.

The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange. The spherically dished reactor closure head (upper head) is welded to a ring flange which is bolted to the vessel ring flange with large-diameter studs. Structural components on the reactor vessel nozzles support the vessel. All internal surfaces of the vessel are clad with stainless steel weld deposit.

The vessel has two outlet nozzles through which reactor coolant flows to the steam generators, and four inlet nozzles through which reactor coolant re-enters the reactor vessel. Smaller nozzles between the reactor coolant nozzles serve as inlets for decay heat cooling and emergency cooling water injection (core flooding and low-pressure injection engineered safety functions).

The bottom head is penetrated by the incore instrumentation nozzles. The closure head is penetrated by flanged nozzles that provide for attaching the control rod drive mechanisms.

A stress analysis of the entire vessel was conducted under both steady-state and transient operations. The result is a complete evaluation of both primary and secondary stresses and the fatigue life of the entire vessel. The reactor vessel was analyzed for fatigue by the original equipment manufacturer.

Design CUFs for the limiting reactor vessel assembly locations were calculated to be less than 1.0 based on the design transients. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the analyzed numbers of transients are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the reactor vessel will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.2.2.2 Reactor Vessel Internals

Reactor vessel internal components include the plenum assembly and the core support assembly. The core support assembly comprises the core support shield, core barrel, lower grid, flow distributor, incore instrument guide tubes, thermal shield, and surveillance specimen holder tubes.

The reactor vessel internals are designed to support the core and to maintain alignment between the fuel assemblies and the control rod drives. The internals also direct the flow of reactor coolant, provide gamma and neutron shielding, provide guides for incore instrumentation between the reactor vessel lower head and the fuel assemblies, support the surveillance specimen assemblies in the annulus between the thermal shield and the reactor vessel wall, and support the internal vent valves.

4.3.2.2.2.1 Low Cycle Fatigue

The core support components are designed to meet the stress requirements of the ASME Section III during normal operation and transients. USAR Appendix 4A contains a detailed stress analysis of the internals under accident conditions. USAR Table 4.2-5 shows that stresses are within established limits, and that deflections would not prevent control rod assembly insertion.

Although the reactor vessel internals are designed to meet the stress requirements of ASME Section III, they are not code components. Consequently, a fatigue analysis of the reactor vessel internals was not performed as part of the original design. The stresses for faulted conditions were analyzed, but fatigue for normal and upset conditions was not analyzed.

Davis-Besse has replaced the majority of the Alloy A-286 bolts for the reactor vessel internals with Alloy X-750 HTH bolts. The replacement bolts were designed to ASME Section III, and fatigue analyses were performed for the replacement bolts. Davis-Besse has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts. All cumulative usage factors calculated for the reactor vessel internals bolts are based on the nuclear steam supply system design transients identified in Table 4.3-1, and are less than 1.0. Therefore, the effects of fatigue will be adequately managed for the period of extended operation by the Fatigue Monitoring Program.

Disposition: 10 CFR 54.21(c)(1)(iii) The low-cycle fatigue analysis TLAA for the reactor vessel internals will be managed by the Fatigue Monitoring Program for the period of extended operation.

4.3.2.2.2.2 Reactor Vessel Internals Flow Induced Vibration

The classic endurance limit approach to design of components subject to flow induced vibration is based on the observation that a fatigue curve becomes approximately asymptotic to a given value of stress (the endurance limit) for large numbers of cycles. A component can be designed for infinite life by maintaining the actual peak stresses below the endurance limit. Unfortunately, actual data, especially for austenitic stainless steel, has not been collected to the endurance limit.

For the Davis-Besse reactor vessel internals, the ASME Code fatigue curve was extended to 1E+12 cycles (the upper bound on the number of cycles for a 40-year design life). The resulting stress value of 20,400 psi was reduced to 18,000 psi as the endurance limit. For 60-years of operation, it follows that 1.5E+12 would bound the expected loading cycles. The extrapolated fatigue curve at 1.5E+12 cycles is approximately 20,200 psi, still above the 18,000 psi that was used as the endurance limit. Therefore the 18,000 psi endurance limit used for the flow induced vibration analyses of the reactor vessel internals remains valid for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(i) The endurance limit for flow induced vibration of the reactor vessel internals remains valid to the end of the period of extended operation.

4.3.2.2.2.3 Incore Instrumentation Nozzles, Surveillance Capsule Holder Tubes

The incore instrument nozzles were analyzed for fatigue due to flow induced vibration. The resulting CUF is 0.59. An additional 20 years of operation would result in a CUF of no more than 0.885 (1.5 x 0.59), which remains below the limit of 1.0. This CUF has been satisfactorily projected for the period of extended operation.

The re-designed surveillance capsule holder tubes were analyzed for fatigue due to flow induced vibration. The resulting CUF is 0.00042. An additional 20 years of operation would result in a CUF of no more than 0.00063 (1.5 x 0.00042), which remains below the limit of 1.0. This CUF has been satisfactorily projected for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(ii) The CUFs for flow induced vibration of select reactor vessel internals have been projected to the end of the period of extended operation.

4.3.2.2.3 Control Rod Drive Housings Fatigue

The control rod drive mechanism is an electro-mechanical device that includes a pressure vessel (housing).

The control rod drive housings are designed to ASME Section III, 1968 Edition through Summer 1970 Addenda. The control rod drive housings were analyzed for fatigue by the original equipment manufacturer. The cumulative usage factors calculated for the various control rod drive locations are based on the nuclear steam supply system design transients identified in Table 4.3-1, and are all less than 1.0. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the control rod drive housings will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.2.2.4 Reactor Coolant Pump Casings Fatigue

The reactor coolant pumps are single stage, single suction, vertical centrifugal pumps. The pump casings consist of a bottom suction inlet, a multi-vane diffuser, a collecting scroll, and a horizontal discharge nozzle. The pump casing is welded into the piping system, and the pump internals can be removed for inspection or maintenance without removing the casing from the piping.

The reactor coolant pump casings are designed to ASME Section III, 1968 Edition through Winter 1968 Addenda. The reactor coolant pumps were analyzed for fatigue by the original equipment manufacturer. Design cumulative usage factors for the limiting reactor coolant pump locations were calculated based on design transients, and are all less than 1.0. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the reactor coolant pumps will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.2.2.5 Pressurizer Fatigue

The pressurizer is a vertical-cylindrical vessel that is connected to the reactor outlet piping by the surge line. The vessel is protected from thermal effects by a distribution baffle on the surge pipe inside the vessel. Two ASME Code relief valves are connected to the pressurizer to relieve system overpressure. A pilot-operated relief valve limits the lifting frequency of the code relief valves. Replaceable electric heater bundles in the lower section and a water spray nozzle in the upper section maintain the steam and water at the saturation temperature corresponding to the desired Reactor Coolant System pressure.

The pressurizer is designed to ASME Section III, 1968 Edition through Summer 1968 Addenda. The pressurizer was analyzed for fatigue by the original equipment manufacturer. Design cumulative usage factors for the limiting pressurizer locations, including the surge nozzle, were calculated based on design transients, and are all less than 1.0. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the pressurizer will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.2.2.6 Once Through Steam Generators (OTSGs)

The once through steam generator design is a vertical, straight-tube-and-shell heat exchanger that produces superheated steam at approximately a constant pressure. Reactor coolant flows downward through the tubes, and steam is generated on the shell side. The parts exposed to reactor coolant system pressure are the hemispherical heads (including inlet and outlet nozzles), the tubesheets, and the straight Inconel tubes between the tubesheets. The reactor coolant side has access ports (manways and inspection openings), and a drain nozzle for the bottom head. The reactor coolant side of the unit can be vented by a vent connection on the reactor coolant inlet pipe to each unit. The unit is supported by a skirt attached to the bottom head which rests on a sliding support and provides the required freedom of movement to accommodate thermal expansion of the Reactor Coolant System.

The shell, the outside of the tubes, and the tubesheets form the boundaries of the steam-producing section of the vessel. Within the shell, the tube bundle is surrounded by a baffle, which is divided into two sections. The upper part of the annulus between the shell and baffle is the superheater outlet, and the lower part is the feedwater inlet-heating zone.

The various aspects of steam generator fatigue analysis are addressed in the subsections below.

4.3.2.2.6.1 OTSGs Fatigue

The primary (tube) and secondary (shell) sides of the once through steam generators are designed to ASME Section III, 1968 Edition through Summer 1968 Addenda. The steam generators were analyzed for fatigue by the original equipment manufacturer. The cumulative usage factors for the limiting primary and secondary side steam generators locations were calculated based on design transients, and are all less than 1.0. In addition, the steam generator remote weld plugs have a limited design life of 33 heatup-cooldown cycles to maintain a fatigue usage of less than 1.0. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the steam generators will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.2.2.6.2 OTSGs Tube Sleeves Fatigue

USAR Section 5.5.2.3 indicates that steam generator tubes that are found to be leaking may be plugged or repaired by mechanical (rolled) sleeving. Section III of the ASME Code does not provide design rules for mechanically roll-expanded attachments, and theoretical stress analyses are inadequate. In such cases, Appendix II of ASME Section III permits the use of experimental stress analysis to substantiate the critical or governing stress. The structural adequacy of the sleeve attachment to withstand cyclic loadings was demonstrated by a fatigue test per ASME Section III, Appendix II-1500. The sleeve loading transients for the fatigue test were based on the design transients. In particular, the pressure cycling portion of the fatigue test is based on the number of startup cycles for a once through steam generator (360 cycles).

Note that the steam generator tube sleeves were tested to 360 startup cycles to bound all Babcock & Wilcox 177 fuel assembly plants. Davis-Besse has only 240 startup cycles allowed in USAR Table 5.1-8, and only 128 projected startup cycles in 60 years of operation per Table 4.3-1. Consequently, Davis-Besse will not approach the tested number of cycles for the once through steam generator tube sleeves during the period of extended operation, and the TLAA associated with fatigue testing of the tube sleeves will remain valid.

Disposition: 10 CFR 54.21(c)(1)(i) The fatigue testing of the once through steam generator tube sleeves will remain valid for the period of extended operation.

4.3.2.2.6.3 OTSGs Auxiliary Feedwater Modification

The original auxiliary feedwater headers internal to the steam generators were found damaged during the 1982 refueling outage. The repair installed an external header on each steam generator, including some rerouting of piping and supports. Included in this repair was the evaluation of the eight new holes in the steam generators, the auxiliary feedwater thermal sleeves, the riser flange attachment to the shell (shell, thermal sleeve bearing area and studs), and flow induced vibration of the steam generator tubes.

The design of this 1982 modification has been included in the steam generator stress analysis referenced in Section 4.3.2.2.6.1 above. Therefore the fatigue analyses of the steam generator shell performed as part of this modification are included in the steam generator fatigue previously discussed in Section 4.3.2.2.6.1.

The analysis of the auxiliary feedwater thermal sleeve stresses provided a basis for demonstrating that the auxiliary feedwater thermal sleeve is capable of withstanding 300 cycles of auxiliary feedwater injection transients. This analysis was performed in accordance with the requirements of the ASME Code for Class I components. The riser flange attachment to the steam generator shell was also analyzed per ASME Code requirements, and was acceptable for a design life of 875 cycles of auxiliary feedwater initiation. Auxiliary feedwater initiations, Transients 30A and 30B in Table 4.3-1, are currently only at 196.5 and 224.5 cycles respectively. Transients 30A and 30B are projected to a maximum of 387 and 442 cycles, respectively, through the period of extended operation. These 60-year projections are below the 875 design cycles for the riser flange attachment but exceed the 300 design cycles for the auxiliary feedwater thermal sleeve. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

Flow induced vibration of the steam generator tubes with the new feedwater header design was also reviewed. It was concluded that the stress and deflection with the external headers was significantly less than the stress and deflection with the original internal headers; consequently flow induced vibration was not reanalyzed for this modification. Section 4.3.2.2.6.4 below, discusses the flow induced vibration analyses of the steam generator tubes.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the auxiliary feedwater header modification will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.2.2.6.4 OTSGs Tubes and Tube Stabilizers Flow Induced Vibration

Flow induced vibration of the once through steam generator tubes has been analyzed several times over the life of the Davis-Besse plant. The latest flow induced vibration analysis shows that the highest cumulative usage factor for any existing tube configuration is 0.443 for an un-repaired tube next to the open lane. Adding 20 years of operation to this tube increases the cumulative usage factor by a factor of 1.5 to a 60-year value of 0.665, which remains acceptable (< 1.0).

The cumulative usage factor for the 3/8 inch tube stabilizers is calculated using both high cycle (flow inducted vibration) and low cycle (transients) fatigue. As the cumulative usage factors are only 0.12 for the tube-to-stabilizer weld and 0.07 for the nail, the flow induced vibration portion of these cumulative usage factors can be increased by 1.5 for 60 years, and the cumulative usage factors will remain below 1.0.

Disposition: 10 CFR 54.21(c)(1)(ii) The TLAA associated with the flow induced vibration of the steam generator tubes and tube stabilizers has been projected through the period of extended operation.

4.3.2.3 Class 1 Piping and Valves

The Davis-Besse reactor coolant system piping, as well as reactor coolant pressure boundary piping in other systems, was designed to American National Standards Institute (ANSI) B31.7 Draft, February 1968 with Errata, June 1968 and also meets the design requirements of ANSI B31.7, 1969 Edition. The B31.7 Piping Code requires evaluation of transient thermal and mechanical load cycles and determination of fatigue usage for Class 1 piping. The reactor head vent and other piping designated as quality group A, B, or C is designed to ASME Section III, 1971 Edition, Class 1, 2 or 3 respectively. Only quality group D piping is designed to ANSI B31.1. Davis-Besse has no Class 1 piping designed to B31.1.

4.3.2.3.1 Class 1 Piping Fatigue

Class 1 piping at Davis-Besse includes the following piping.

<u>Reactor Coolant Piping:</u>

The reactor coolant piping connects the major components of the Reactor Coolant System, including the reactor vessel, the steam generators and the reactor coolant pumps. The reactor coolant piping has welded connections for pressure taps, temperature elements, vents, drains, decay heat removal, and emergency core cooling high-pressure injection water.

The CUFs calculated for the reactor coolant piping are based on the design transients identified in Table 4.3-1 and are all less than 1.0.

Pressurizer Surge Line:

NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," required the re-evaluation of the cyclic fatigue of the pressurizer surge line [References 4.8-10, 4.8-11, 4.8-12, and 4.8-13]. As part of this evaluation the design basis plant heatup and cooldown transients were completely redefined. Other transients were modified to include thermal stratification and striping. In addition to these changes, a number of transients were added and other modifications were made to the existing transients based on a review of the plant operating history, including the operating procedures. The surge line piping and nozzles were analyzed for license renewal, considering the effects of the reactor coolant environment. See Section 4.3.4 for the latest pressurizer surge line analyses.

Reactor Coolant Drains and Letdown Lines:

The Class 1 portion of the reactor coolant drains, designed to ASME III Class A (Class 1), extends only to the second isolation valve away from the reactor vessel. The letdown line extends from the suction of reactor coolant pump 1-1-1 (RCS Loop 1-1 Cold Leg) to the letdown cooler isolation valves. The original analysis for these vents and drains was updated based on NRC Bulletin 79-14. The CUFs calculated for the reactor coolant drains and letdown line are based on the design transients in Table 4.3-1 and are all less than 1.0.

High Pressure Injection Lines:

The Class 1 portion of the High Pressure Injection System, designated as ASME III Class A (Class 1) is entirely within the containment vessel and consists of four legs, each of which extend from the first of two isolation valves to the cold leg piping on the inlet to each of the four reactor coolant pumps. The current analysis, updated per NRC Bulletin 79-14, is based on the design transients in Table 4.3-1, and all CUFs are less than 1.0.

A thermal sleeve is provided in the high-pressure injection connection to the reactor coolant inlet piping. The analysis of the high-pressure injection nozzles determined that high-pressure injection flow tests had negligible effect on the high-pressure injection nozzles, but a significant effect on the normal makeup nozzle. The CUF for the normal makeup nozzle was calculated to be 0.558 after 40 flow tests; 0.513 usage due to the 40 flow tests and 0.045 usage due to all other transients. Projections of cycles for 60 years implies that the design cycles of 40 will be reached in year 51, with 48 cycles occurring by year 60. Projecting the CUF to a 60-year number with 50 tests, gives a

CUF of 0.686 (0.045 + 50/40 * 0.513), which implies the nozzles will still be acceptable. However, Davis-Besse monitors these cycles and will ensure action is taken before the design cycles are reached. Davis-Besse has committed (see Appendix A) to replace the high pressure injection thermal sleeves and safe ends prior to reaching the period of extended operation. Davis-Besse manages fatigue of these nozzles.

Decay Heat Removal Lines:

The Class 1 portion of the Decay Heat Removal System, designated as ASME III Class A (Class 1) is entirely within the containment vessel and consists of two legs, each of which extends from stop-check isolation valves to the reactor vessel core flood lines. The current analysis, updated per NRC Bulletin 79-14, is based on the design transients in Table 4.3-1 and all CUFs are less than 1.0.

Core Flooding Lines:

The Class 1 portion of the Core Flood System, designated as ASME III Class A (Class 1) is entirely within the containment vessel and consists of two legs, each of which extends from a core flood tank to a reactor vessel core flood nozzle. The current analysis, updated per NRC Bulletin 79-14, is based on the design transients in Table 4.3-1 and all CUFs are less than 1.0.

Pressurizer Safety/Relief Valve Lines:

The Class 1 pressurizer safety/relief valve lines are entirely within the containment, and run from the safety/relief nozzles on the top head of the pressurizer to the safety/relief valves. The CUFs calculated for the pressurizer safety/relief valve lines are based on the design transients in Table 4.3-1 and are all less than 1.0.

Class 1 Piping Summary:

All cumulative usage factors calculated for Class 1 piping are less than 1.0 based on the design transients identified in Table 4.3-1. The Fatigue Monitoring Program will monitor these transients for the period of extended operation and ensure that action is taken before the design cycles are reached. See Section 4.3.1 above for further discussion of the design cycles.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of aging on the Class 1 piping will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.2.3.2 Class 1 Valves Fatigue

A review was performed to determine if the current licensing basis for Davis-Besse contains fatigue analyses for Class 1 valves. Piping and instrumentation diagrams were reviewed to identify the Class 1 valves of four inches or greater diameter. While there is no code distinction for fatigue analyses between large bore and small bore valves, the review of the large bore valves was intended to provide a representation of the status of such analyses for all Class 1 valves. There were 12 valves of four inches or greater diameter that were identified as a result of this effort. A review of the Davis-Besse quality assurance records located the stress reports of record for each of the 12 valves, however, no associated fatigue reports were identified. Therefore, it is concluded that no fatigue analyses for Class 1 valves were performed, and there is no TLAA for Class 1 valves at Davis-Besse. This conclusion is consistent with industry practice at the time Davis-Besse was designed. Valve bodies and pump casings were considered robust compared to the piping systems in which they were located and fatigue of the attached piping was understood to bound the fatigue of the valve bodies.

Disposition:	Not a TLAA	There are no fatigue analyses for the Class 1 valves at Davis-Besse and thus there is no TLAA associated with fatigue of Class 1 valves.

4.3.2.3.3 High Energy Line Break Postulations

USAR Section 3.6.2.1 indicates that the criteria given in Standard Review Plan Sections 3.6.1 and 3.6.2, including Branch Technical Position MEB 3-1, were used in determining the pipe break locations for pipe whip restraint design. This allows the elimination of potential break locations based on cumulative usage factors being less than 0.1, if other stress criteria are also met. The cumulative usage factors calculated for Davis-Besse piping were based on the design transients that are counted by the Fatigue Monitoring Program. If any of the design cycles are approached, the Fatigue Monitoring Program will require action prior to the design cycles being reached. That action will include a review of the high energy line break location selections. As such, the effects of fatigue on the high energy line break location selection will be managed for the period of extended operation.

The identification of high energy line break locations for the hot and cold leg piping was replaced by leak-before-break criteria in 1990. See Section 4.7.1 below for a discussion of leak-before-break.

Disposition:	10 CFR 54.21(c)(1)(iii)	The effects of fatigue on the high energy line break location selection will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.3.3 NON-CLASS 1 FATIGUE ANALYSES

The specific codes and standards to which systems and components important to safety were designed are listed in USAR Table 3.2-2. Non-class 1 components that are Quality Group B or C are largely designed and constructed to the ASME Boiler and

Pressure Vessel Code, but certain components are built to other codes including B31.1, American Water Works Association, and the Draft Pump and Valve Code.

The aging management review for Davis-Besse non-Class 1 mechanical components is contained in Section 3.2. Non-Class 1 components with a maximum service temperature in excess of 220°F for carbon steel, or 270°F for stainless steel, are identified in Section 3.2 as requiring further evaluation for fatigue. That evaluation is summarized in Sections 4.3.3.1 and 4.3.3.2 below. Section 4.3.3.1 determines that all TLAAs associated with piping and in-line components (tubing, piping, thermowells, valve bodies, etc.) remain valid for the period of extended operation. Section 4.3.3.2 determines that there are no TLAAs associated with non-piping components (tanks, heat exchangers, pump & turbine casings, etc.).

4.3.3.1 Non-Class 1 Piping and In-Line Components

The design of ASME Section III Class 2 and Class 3 piping systems incorporates a stress range reduction factor for determining acceptability of piping design with respect to thermal stresses. Davis-Besse components designated as quality group D are designed to ANSI B31.1, which incorporates stress range reduction factors based upon the number of thermal cycles. A stress range reduction factor of 1.0 in the stress analyses applies for up to 7,000 thermal cycles. The allowable stress range is reduced by the stress range reduction factor if the number of thermal cycles exceeds 7,000. If fewer than 7,000 cycles are expected through the period of extended operation, then the fatigue analysis (stress range reduction factor) of record remains valid through the period of extended operation.

Thermal cycles have been projected for 60 years of plant operation in Section 4.3.1.2 above. These projections, applied to the non-Class 1 piping and in-line components indicate that 7,000 thermal cycles will not be exceeded during 60 years of operation.

- Piping connected to the Reactor Coolant System, the Main Steam System, or the Main Feedwater System will experience essentially the same transients as the Reactor Coolant System. As shown in Table 4.3-1, there are less than 2400 total thermal cycles projected in 60 years of operation. As such, systems connected to the Reactor Coolant, Main Steam, or Main Feedwater systems will not exceed 7,000 equivalent full temperature cycles during the period of extended operation, and the system piping fatigue analyses (stress range reduction factors) remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).
- Piping from the fire water storage tank heat exchanger to the fire water storage tank operates at a temperature that exceeds the fatigue threshold temperature. While cycles have not been counted on this system, it is estimated that the system is cycled four times a week for 24 weeks (October-March) out of the year, or 96 cycles a year. This is a conservative estimate because in very cold months

the system is kept running rather than being cycled. As 96 cycles per year for 60 years is 5,760 cycles, the fire water storage tank will not exceed 7,000 design cycles through the period of extended operation, and the system piping fatigue analyses (stress range reduction factors) remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

- Piping and piping components associated with the emergency diesels, the fire pump diesel engine, and the station blackout diesel require evaluation of thermal fatigue.

  Technical Specification surveillance requirements 3.8.1.2 and 3.8.1.3 require each emergency diesel generator to be started once per 31 days, or 720 starts in 60 years. Surveillance requirement 3.8.1.8 requires each emergency diesel to be run twice per year, or 120 starts in 60 years. Surveillance requirements 3.8.1.13, 3.8.1.14, and 3.8.1.15 require extended runs every two years, or 30 starts in 60 years. As these surveillance requirements may be run consecutively, or may take credit for inadvertent starts, conservatively combining these starts indicates there will be less than 870 (720 + 120 + 30) surveillance-related starts in 60 years. Unanticipated operation of the emergency diesels is less frequent than testing. Doubling the surveillance-related starts to account for unanticipated operation produces 1,740 cycles in 60 years and remains below the 7,000 cycles implicitly assumed in the analysis.

  The station blackout diesel generator is tested monthly per Technical Requirements Manual Section 8.8.2. This will account for 720 thermal cycles in 60 years. Unplanned operation of the station blackout diesel is very infrequent (less than once per year), so the total cycles in 60 years remains below the 7,000 cycles implicitly assumed in the analysis.

  Fire Hazards Analysis Report surveillance requirement 8.1.2.E.1 requires a start of the diesel fire pump engine every 31 days, or 720 times in 60 years. Surveillance requirements 8.1.2.E.4 and 8.1.2.E.5 require extended diesel runs once per cycle, conservatively estimated at 60 times in 60 years. Combining these surveillance requirements concludes there will be less than 780 surveillance-related starts in 60 years. Unanticipated operation of the fire pump diesel engine is less frequent than testing. Doubling the surveillance-related starts to account for unanticipated operation produces 1,560 cycles in 60 years and remains below the 7,000 cycles implicitly assumed in the analysis.

  As such, the emergency diesels, diesel fire pump engine, and station blackout diesel, will not exceed 7,000 equivalent full temperature cycles during the period of extended operation, and the system piping fatigue analyses (stress range reduction factors) remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

- Piping and piping components in the Gaseous Radwaste System may exceed the temperature threshold for fatigue. The piping and piping components are designed to either ASME Section III, Class 3 or ANSI B31.1. There are no explicit fatigue analyses for this piping. The only source of hot gas above the fatigue threshold is the vent of the reactor coolant drain tank. Gas vented from this tank will only exceed the fatigue threshold immediately after a safety valve or power operated relief valve lift. As shown in Table 4.3-1, Events 26A, 26B, and 26C, only 72 lifts of these valves are expected in 60 years. As such, the Gaseous Radwaste System will not exceed 7,000 equivalent full temperature cycles during the period of extended operation, and the system piping fatigue analyses (stress range reduction factors) remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).
- Piping and piping components associated with the containment air systems may be exposed to maximum operating temperatures that exceed the threshold values for fatigue, and therefore require further evaluation of thermal fatigue. The subject piping is designed to ASME Section III Class 2 or ANSI B31.1. There are no explicit fatigue analyses for this piping. The containment air temperature is restricted to less than 120°F per Technical Specification 3.6.5. The maximum operating temperature for the containment air systems is 264°F; which corresponds to the containment design temperature. These systems will only see that temperature following the containment design transient (LOCA), and will only see that temperature once in the life of the plant. As such, the containment air systems will not exceed 7,000 equivalent full temperature cycles during the period of extended operation, and the system piping fatigue analyses (stress range reduction factors) remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).
- Piping and piping components in the sampling systems may exceed the temperature threshold for fatigue, and therefore require further evaluation of thermal fatigue. These sample pipes, valves, and tubing are used for collecting samples of feedwater or main steam and for routing reactor coolant to the Post Accident Sampling System.

  Sample piping to the Post Accident Sampling System would be used only in the case of a design basis accident; and thus no cycles are anticipated. The lines are occasionally used as a test, less than once per year, or 60 cycles in 60 years. The lines may also be used to degasify the Reactor Coolant System (pressurizer) but this is defined as an "Infrequent or Special Operation". An estimate of "infrequent operation" is less than once per fuel cycle, or 30 times in 60 years. Consequently this piping and piping components are expected to see less than 90 cycles in 60 years of operation. As this is well below the 7,000 cycles in any implicit fatigue analyses, the system piping stress analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Placing the secondary sample panel in service is an infrequent operation performed after each refueling outage. As such the sample panel will only heatup and cooldown when the secondary plant heats up and cools down, which per Table 4.3-1 is projected to 128 cycles in 60 years. Even doubling the cycles to allow for unplanned isolations and restarts, this system will experience only 256 cycles in 60 years. As such, any implicit fatigue analyses in the system piping stress analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

- Piping and piping components of the Auxiliary Steam System may be exposed to maximum operating temperatures that exceed the threshold for further evaluation of thermal fatigue. The Auxiliary Steam System is supplied from the Main Steam System during normal operation and by the auxiliary boiler when the plant is off line (including during startups). Because the auxiliary boiler sometimes maintains temperature and pressure in the Auxiliary Steam System when the plant is off line, the Auxiliary Steam System will see fewer transients than are experienced by the overall plant. As shown in Table 4.3-1, the Main Steam System (and Reactor Coolant System) is projected to see only 1,915 total thermal cycles in 60 years of operation. As such, the Auxiliary Steam System will not exceed 7,000 equivalent full temperature cycles during the period of extended operation, and the system piping fatigue analyses (stress range reduction factor)s remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).
- Piping and piping components of the Station Heating System may be exposed to maximum operating temperatures that exceed the threshold for further evaluation of thermal fatigue. The Station Heating System is a hot water system with a primary loop heated by the Auxiliary Steam System and secondary loops heated by the primary loop. This system is normally in service only during the heating season (winter). The Station Heating System could cycle several times per year as environmental conditions change. Cycling 20 times per year produces 1,200 cycles in 60 years, therefore the Station Heating System will remain below the 7,000 cycles. As such, the Station Heating System will not exceed 7,000 equivalent full temperature cycles during the period of extended operation, and the system piping fatigue analyses (stress range reduction factors) remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Disposition: 10 CFR 54.21(c)(1)(i) The TLAAs associated with fatigue of non-Class 1 piping and in-line components will remain valid for the period of extended operation.

4.3.3.2 Non-Class 1 Major Components

Fatigue need not be addressed for non-Class 1 vessels, heat exchangers, storage tanks, and pumps, unless these components were designed to ASME Section VIII Division 2 or ASME Section III Subsection NC-3200. For those non-Class 1 non-piping components possibly subject to fatigue, a review of component design codes was conducted to determine if fatigue analyses of the components were required. If no fatigue analysis was required, then no TLAA for fatigue exists.

While most Class 1 components are designed in accordance with ASME Section III, non-Class 1 pressure vessels, heat exchangers, tanks, and pumps are often designed in accordance with other industry codes and standards, reactor designer specifications, and architect engineer specifications. ASME Section III Subsection NC-3200 and ASME Section VIII Division 2 include fatigue design requirements, and include provisions for "exemption from fatigue," which is actually a simplified fatigue evaluation based on materials, configuration, temperature, and cycles. If cyclic loading and fatigue usage for a component could be significant, then ASME Section VIII Division 2 or NC-3200 would have been specified.

Due to conservatism in ASME Section VIII Division 1 and ASME Section III NC-3100 and ND-3000, detailed fatigue analysis is not required. Also, fatigue analyses are not required for NC and ND pumps and storage tanks (< 15 psig), or for other design codes (e.g., ASME Section VIII Division 1, AWWA, MSS, NEMA). Components designed and fabricated to these codes require no fatigue analyses for the period of extended operation.

The non-Class 1 non-piping components identified in Sections 3.2, 3.3 and 3.4 as requiring further evaluation for fatigue are discussed below.

- The decay heat removal coolers, decay heat removal pumps, and borated water storage tank heater are the only non-piping components in the Decay Heat Removal / Low Pressure Injection System that may exceed the fatigue threshold temperature. The decay heat removal coolers are designed to ASME Section III-C (tube side) and ASME Section VIII (shell side). The decay heat removal pumps are designed to the draft ASME Code for pumps and valves 1968, Class 2. The borated water storage tank heater is designed to ASME Section VIII Division 1 (tube side) and ASME Section VIII (shell side)

  No fatigue analyses exist for these components, and therefore, there are no TLAAs related to fatigue. These components require no further fatigue evaluation for period of extended operation.

- The auxiliary feedwater pump turbine casings are the only non-piping components within the evaluation boundaries of the Main Steam System that exceed the fatigue threshold temperature. There are no design codes associated with these turbines, only the standards of the American Society for Testing and Materials and National Electrical Manufacturers Association.

No fatigue analyses exist for the auxiliary feedwater pump turbine casings, and therefore, there are no TLAAs related to fatigue. These components require no further fatigue evaluation for the period of extended operation.

- The fire water storage tank heat exchanger is the only non-piping component within the evaluation boundaries of the Fire Protection System that exceeds the fatigue threshold temperature. This heat exchanger was fabricated in accordance with ASME Section VIII Division 1.

  No fatigue analysis exists for the fire water storage tank heat exchanger, and therefore, there is no TLAA related to fatigue. This component requires no further fatigue evaluation for the period of extended operation.

- The waste gas surge tank is the only non-piping component within the evaluation boundaries of the Gaseous Radwaste System that exceeds the fatigue threshold temperature. The waste gas surge tank is built to ASME Section III, Class C.

  No fatigue analysis exists for the waste gas surge tank, and therefore, there is no TLAA related to fatigue. This component is acceptable for period of extended operation without further evaluation.

- The pressurizer quench tank is the only non-piping component within the boundaries of the Reactor Coolant Drains and Vents System that may exceed the threshold temperature requiring further evaluation of thermal fatigue. The design code for the pressurizer quench tank is ASME Section III Class 3.

  No fatigue analysis exists for the pressurizer quench tank, and therefore, there is no TLAA related to fatigue. This component requires no further fatigue evaluation for the period of extended operation.

- The Intake Structure Unit Heater heat exchangers are supplied by low pressure steam and may exceed the threshold temperature of thermal fatigue. No fatigue analysis exists for these nonsafety-related components, and therefore, there is no TLAA related to fatigue. These components require no further fatigue evaluation for the period of extended operation.

- The evaporator package condensate drain pumps, the degasifier package drain pumps, and the condensate pumps all may reach temperatures of approximately 300°F. No fatigue analysis exists for these nonsafety-related pumps, and therefore, there is no TLAA related to fatigue. These components require no further fatigue evaluation for the period of extended operation.

- The 10 psig condensate tank may reach 298°F. The 10 psig condensate tank is built to ASME Section VIII. No fatigue analysis exists for this nonsafety-related tank, and therefore, there is no TLAA related to fatigue. This component requires no further fatigue evaluation for the period of extended operation.

Disposition: Not a TLAA — There are no fatigue analyses, and hence no TLAAs, associated with the non-Class 1 non-piping components.

4.3.4 EFFECTS OF REACTOR COOLANT ENVIRONMENT ON FATIGUE

4.3.4.1 Background

Industry test data indicate that certain environmental effects (such as temperature and dissolved oxygen content) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. The ASME design fatigue curves were based on laboratory tests in air and at low temperatures. Although the failure curves derived from laboratory tests were adjusted to account for effects such as data scatter, size effect, and surface finish, these adjustments may not be sufficient to account for actual plant operating environments.

No immediate NRC staff or licensee action is necessary to deal with the environmentally assisted fatigue issue. However, because metal fatigue effects increase with service life, environmentally assisted fatigue is evaluated for license renewal. Guidance for performing this evaluation is provided in NUREG/CR-6260 "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," and EPRI Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application."

NUREG/CR-6260 identifies locations of interest for consideration of environmental effects in several types of nuclear plants. Section 5.3 of NUREG/CR-6260 reviews the following locations for Babcock & Wilcox pressurized water reactors.

- Reactor vessel shell and lower head; including the instrumentation nozzles
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line (including pressurizer surge nozzle and hot leg surge nozzle)
- High pressure injection/makeup nozzle
- Reactor vessel core flood nozzle
- Decay heat removal Class 1 piping

Evaluations performed for the period of extended operation do not indicate that 40-year cumulative usage factors will exceed the fatigue limit (1.0) because the environmentally assisted fatigue adjustment is not applied during the initial 40 years of operation, consistent with the closure of Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components for 60-year Plant Life."

4.3.4.2 Davis-Besse Evaluation

The effect of the reactor coolant environment on fatigue usage has been evaluated for the six locations identified in NUREG/CR-6260. An environmentally assisted fatigue correction factor, F_{en}, was determined using material specific guidance contained in NUREG/CR-6583 "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," in NUREG/CR-5704 "Effects of LW Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," and in NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials." Environmentally adjusted cumulative usage factors (U_{en}), which include the effect of reactor water environment, were obtained by multiplying the F_{en} times the in-air CUFs.

The following bounding F_{en} values were calculated: 1.74 for carbon steel, 2.45 for low-alloy steel, 15.35 for stainless steel with $T \geq 200$ °C, and 2.55 for stainless steel with $T < 200$°C. An F_{en} value of 4.16 was calculated for the nickel-based alloy incore instrument nozzles. These F_{en} values were applied to either design CUFs or adjusted CUFs at all NUREG/CR-6260 locations with the exception of the surge line piping and high pressure injection/makeup (HPI/MU) nozzle safe end. The surge line piping and HPI/MU nozzle safe end were evaluated using an integrated F_{en} approach consistent with MRP-47, Revision 1, Section 4.2.

Environmentally-adjusted U_{en} factors are summarized in Table 4.3-2. Each location listed in Table 4.3-2 is discussed individually below.

Location 1 is the reactor vessel shell and lower head, including the nickel-based alloy (NBA) incore instrument nozzles.

Evaluations for the vessel shell and lower head base metal and welds are based on application of bounding F_{en} environmental penalty factors to design CUFs. The maximum design CUF for the clad alloy steel reactor vessel head is 0.024. Adjusting this CUF by a bounding F_{en} of 2.45 for low alloy steel (LAS) results in an U_{en} of 0.059.

Evaluations for the nickel-based alloy incore instrument nozzles are based on application of an F_{en} environmental penalty factor, calculated in accordance with NUREG/CR-6909, to CUFs that were adjusted using the new in-air fatigue curves reported in NUREG/CR-6909. The maximum design CUF for the nickel-based alloy incore instrument nozzle is 0.77. The original design CUF of 0.77 was conservative due to the use of ASME III, Figure N-415a (applies to LAS), to determine allowable cycles versus Figure N-415b (applies to stainless steel and nickel-based alloy). This original design CUF was reduced to 0.206 by applying the alternating stresses from the original design calculation to the new in-air design curve for stainless steel in NUREG/CR-6909. The new design curve for austenitic stainless steel may also be used for nickel-based alloy materials. Adjusting the revised CUF of 0.206 by an F_{en} of 4.16 for nickel-based alloy results in an U_{en} of 0.857.

Location 2 is the reactor vessel inlet and outlet nozzles.

Evaluations for the reactor vessel inlet and outlet nozzles are based on application of bounding F_{en} environmental penalty factors to design CUFs that were adjusted by identifying incremental fatigue contribution attributed to the full NSSS design transient cycles and reducing those incremental contributions based on the 60-year cycle projections presented in Table 4.3-1.

The maximum design CUF for the clad low alloy steel RV inlet nozzles is 0.829. This design CUF was reduced to 0.146 by considering incremental usage and utilizing the 60-year cycle projections from Table 4.3-1. Adjusting the revised CUF of 0.146 by a bounding F_{en} of 2.45 for LAS results in a U_{en} of 0.358. The maximum design CUF for the clad low alloy steel RV outlet nozzles is 0.768. This design CUF was reduced to 0.335 by considering incremental usage and utilizing the 60-year cycle projections. Adjusting the revised CUF of 0.335 by a bounding F_{en} of 2.45 for LAS results in a U_{en} of 0.821.

Location 3 is the pressurizer surge line, which includes the hot leg surge nozzle, surge line piping, and pressurizer surge nozzle.

Hot Leg Surge Nozzle - Includes the stainless steel clad carbon steel surge nozzle, Alloy 82/182 weld buildup (buttering) on the outboard end of the nozzle, Alloy 82/182 weld that connects the weld buildup to the stainless steel pipe, and the Alloy 52/152 weld overlay on the outer diameter of the nozzle that extends from near the end of the taper region of the hot leg surge nozzle to just beyond the Alloy 82/182 weld.

The bounding environmentally adjusted cumulative usage factor for the hot leg surge nozzle is as follows:

- Evaluations for the stainless steel clad carbon steel nozzle are based on application of bounding F_{en} environmental penalty factors to design CUFs. The maximum design CUF occurs at the inside radius of the carbon steel nozzle at 0.445. Adjusting this CUF by a bounding F_{en} of 1.74 for carbon steel results in a U_{en} of 0.774.

Surge Line Piping - Includes stainless steel pipe, stainless steel fittings, and stainless steel welded joints between the outboard end of the hot leg surge nozzle weld overlay up to the stainless weld that connects the pressurizer surge line to the pressurizer nozzle stainless steel safe end.

The ASME Section III structural/stress analyses performed in the 1990 – 1992 timeframe (BAW-2127 [References 4.8-10 through 4.8-13]) for the stainless steel surge line piping was used to obtain U_{en} values for the surge line. The 60-year transient projections were used for the evaluation with the exception of the 60-year projection of heatup/cooldowns (HU/CDs), where a best estimate number of 114 total was used. The surge line piping was evaluated using an integrated F_{en} approach consistent with MRP-47, Revision 1, Section 4.2.

For the stainless steel surge line piping, the equations for the fatigue penalty factors F_{en} were taken from NUREG/CR-5704. The F_{en} values are a function of dissolved oxygen (DO) level, metal service temperature and strain rate, as described in MRP-47, Revision 1, Section 4.2. The effects of metal service temperature were considered, transformed strain rates were assumed to be at saturation, and dissolved oxygen was considered as being less than 0.05 ppm.

Transformed Strain Rate

Transformed strain rates were assumed to be at the saturation value of ln(0.001). This corresponds to a strain rate of 0.0004%/sec or less.

Transformed Metal Service Temperature

For each Peak or Valley, the metal temperature is known from the Surge Line Functional Specification. For each load set pair, the F_{en} values were calculated based on the varying metal temperature values from the valley to the peak will be integrated. The multiplication of the resulting F_{en} factor - after integration - by the usage factor in air for that particular load set pair (from the Valley to the Peak) results in the usage factor with consideration of the environmental effects for that particular load set pair. This means that for each load set pair: $U_{en} = F_{en} * U(\text{in-air})$.

For each integration point from the Valley to the Peak, the transformed temperature T* is calculated as specified for stainless steel in Subsection 4.2.4 of MRP-47, Revision 1: $T^* = 0.0$ for $T < 392^{o}F$, and $T^* = 1.0$ for $T \geq 392^{o}F$.

Transformed Dissolved Oxygen

For the stainless steel surge line, it will be assumed that dissolved oxygen is less than 0.05 ppm. $O^* = 0.260$.

Surge Line Fatigue Calculation

Using the methodology described above, the ASME Section III structural/stress analyses performed in the 1990 – 1992 timeframe (BAW-2127) for the stainless steel surge line piping was re-evaluated to extract the variations of metal service temperature to calculate environmental correction factors F_{en}. With regard to the methodology discussed above, the following are relevant to the calculation of environmentally-adjusted CUFs for the surge line.

- In the main fatigue usage calculations the F_{en} values are calculated as a function of the temperature changes between the Valley and the Peak [Integration of the F_{en} values ranged between 2.55 when metal temperature is less than 392°F to a maximum of 15.35 when metal temperature equals or exceeds 392°F]. In addition, all the F_{en} calculations are based on the most severe strain rate of 0.0004 % / sec, which is the "saturation strain rate."
- In the fatigue usage calculations for the low stratification transients, the most severe F_{en} of 15.35 is used.
- For all the full-flush cycles, the most severe F_{en} of 15.35 is used.
- For thermal striping by itself (thermal striping fluctuations), the most severe strain amplitude is less than 0.097% and F_{en} is equal to 1.0 for thermal striping.

Surge Line Fatigue Results

The bounding environmentally adjusted cumulative usage factors for the surge line are as follows:

- The maximum design CUF for the stainless steel pipe adjacent to the outboard end of the hot leg surge nozzle is 0.179. Using the integrated F_{en} approach described above, the U_{en} for the stainless steel pipe adjacent to the outboard end of the hot leg surge nozzle weld overlay is 0.387 with a global F_{en} of 5.83. An adjusted CUF of 0.07 is obtained by dividing the U_{en} of 0.387 by the global F_{en} of 5.83.
- The maximum design CUF for the elbows is 0.643. Using the integrated F_{en} approach described above, the maximum U_{en} for the elbows is 0.996 with a global F_{en} of 4.17. An adjusted CUF of 0.239 is obtained by dividing the U_{en} of 0.996 by the global F_{en} of 4.17.
- The maximum design CUF for the straight pipe is 0.764. Using the integrated F_{en} approach described above, the maximum U_{en} for the straight pipe is 0.846 with a global F_{en} of 2.52. An adjusted CUF of 0.336 is obtained by dividing the U_{en} of 0.846 by the global F_{en} of 2.52.
- The maximum design CUF for the stainless steel weld that connects the surge line to the pressurizer surge nozzle safe end is 0.51. Using the integrated F_{en} approach described above, the U_{en} for the stainless steel weld that connects the surge line to the pressurizer surge nozzle safe end is 0.644 with a global F_{en} of 8.84. An adjusted CUF of 0.073 is obtained by dividing the U_{en} of 0.644 by the global F_{en} of 8.84.

Pressurizer Surge Nozzle - Includes the stainless steel clad carbon steel surge nozzle, Alloy 82/182 weld buttering on the outboard end of the nozzle, Alloy 82/182 weld that connects the buttering to the stainless steel safe end, the stainless steel safe end, and the Alloy 52/152 weld overlay on the outer diameter of the nozzle that extends from the end of the taper region of the pressurizer surge nozzle to just beyond the Alloy 82/182 weld.

The bounding environmentally adjusted cumulative usage factors for the pressurizer surge nozzle are as follows:

- Evaluations for the stainless steel clad carbon steel nozzle are based on application of bounding F_{en} environmental penalty factors to design CUFs. For the stainless steel clad carbon steel nozzle the maximum design CUF occurs at the inside radius of the carbon steel nozzle is 0.182. Adjusting this CUF by a bounding F_{en} of 1.74 for carbon steel results in a U_{en} of 0.317.
- Evaluations for the stainless steel safe end are based on application of bounding F_{en} environmental penalty factors to design CUFs adjusted by identifying incremental fatigue contribution attributed to the full NSSS design transient cycles and reducing those incremental contributions based on the 60-year cycle projections. The maximum design CUF for the stainless steel safe end at the inside surface is 0.108. This design CUF was reduced to 0.058 by considering incremental usage and utilizing the 60-year cycle projections. Adjusting this CUF by a bounding F_{en} of 15.35 for stainless steel results in a U_{en} of 0.892.

Location 4 is the high pressure injection/makeup nozzle and stainless steel safe end.

The stainless steel clad carbon steel nozzle is connected to a stainless steel safe end by an Alloy 82/182 weld. Adjustments of design CUFs were made for the HPI/MU nozzle and associated safe end by removing conservatisms in the original design calculation yet maintaining the full set of 40-year NSSS design cycles. In addition, the stainless steel safe end was evaluated using an integrated F_{en} approach consistent with MRP-47, Revision 1, Section 4.2

For the stainless steel clad carbon steel nozzle the maximum design CUF is 0.589. This design CUF was reduced to 0.348 by removing conservatisms in the design analysis yet retaining the full set of NSSS design transients. Adjusting this CUF by a bounding F_{en} of 1.74 for carbon steel results in a U_{en} of 0.606.

The maximum design CUF for the stainless steel safe end is 0.664. This design CUF was reduced to 0.550 by removing conservatisms in the design analysis yet retaining the full set of NSSS design transients. Adjusting this CUF using an integrated F_{en} based on the methodology in MRP-47, Section 4.2.2, yields a U_{en} of 4.417, which is >1.0 and is unacceptable for the period of extended operation. Both the HPI/MU nozzle stainless steel safe end and associated Alloy 82/182 weld have environmentally adjusted CUFs greater than 1.0. and are therefore, unacceptable for the period of extended operation.

Location 5 is the reactor vessel core flood nozzle.

Evaluations of the core flood nozzle are based on application of bounding F_{en} environmental penalty factors to design CUF. As specified in the NUREG/CR-6260, the limiting location for B&W plants is the stainless steel clad low alloy steel nozzle. The maximum design CUF for the stainless steel clad low alloy steel core flood nozzle is 0.0504. Adjusting this CUF by a bounding F_{en} of 2.45 for LAS results in a U_{en} of 0.123.

Location 6 is the decay heat removal system Class 1 piping.

The limiting location is the decay heat return line to core flood system tee. The evaluation is based on application of a bounding F_{en} environmental penalty factor to the design CUF. The maximum design CUF for the stainless steel tee is 0.233. Adjusting this CUF by a bounding F_{en} of 2.55 for stainless steel at fluid temperatures less than 200 °C (392 °F) results in a U_{en} of 0.595. The decay heat system cut in temperature is 280 °F, which is well below the threshold of 200 °C (392 °F).

Table 4.3-2 Davis-Besse CUFs for NUREG/CR-6260 Locations

NUREG/CR-6260 generic locations		Davis-Besse plant-specific locations	Material type	Design CUFs	Adjusted CUFs	F_{en}	U_{en}
1	Reactor vessel shell and lower head	Vessel shell and lower head	LAS	0.024	NA[8]	2.45	0.059
		Incore instrument nozzle	NBA	0.770	0.206[5]	4.16	0.857
2	Reactor vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.829	0.146[1]	2.45	0.358
		Reactor vessel outlet nozzle	LAS	0.768	0.335[1]	2.45	0.821
3	Pressurizer surge line	Hot leg surge nozzle inside radius	CS	0.445	NA[8]	1.74	0.774
		Piping adjacent to outboard end of hot leg surge nozzle	SS	0.179	0.07[2]	5.83	0.387
		Piping elbows	SS	0.643	0.239[2]	4.17	0.996
		Piping straights	SS	0.764	0.336[2]	2.52	0.846
		Piping to pressurizer surge nozzle safe end weld,	SS	0.51	0.073[2]	8.84	0.644
		Pressurizer surge nozzle inside radius	CS	0.182	NA[8]	1.74	0.317
		Pressurizer surge nozzle, safe end	SS	0.108	0.058[1]	15.35	0.892
4	HPI/Makeup nozzle	HPI/Makeup nozzle	CS	0.589	0.348[3]	1.74	0.606
		HPI/Makeup nozzle safe end	SS	0.664	0.550[4]	8.03[6]	4.417[7]
5	Reactor vessel core flood nozzle	Nozzle	LAS	0.0504	NA[8]	2.45	0.123
6	Decay heat Class 1 piping	Decay heat to core flood tee	SS	0.233	NA[8]	2.55	0.595

1. Adjusted CUF obtained by identifying incremental fatigue contribution attributed to the full NSSS design transient cycles for design CUF and reducing those incremental contributions based on the 60-year cycle projections.
2. Adjusted CUF obtained by dividing U_{en} by global F_{en}. Global F_{en} calculated using method from Section 4.2 of MRP-47, Revision 1 as described above for the pressurizer surge line.
3. Design CUF reduced from 0.589 to 0.348 by removing conservatisms in the original calculation. Full set of design cycles were used for the calculation.
4. Design CUF reduced from 0.664 to 0.550 by removing conservatisms in the original calculation. Full set of design cycles were used for the calculation.
5. Adjusted CUF obtained by applying the alternating stresses from the original design calculation to the new in-air design curve in NUREG/CR-6909 for stainless steel.
6. This is a global F_{en} obtained by dividing U_{en} by the CUF (4.417/ 0.550).
7. 4.417 is >1.0 and is unacceptable for the period of extended operation. (See Section 4.3.4.2, Location 4).
8. Adjusted CUF was not required. Design CUF multiplied by F_{en} resulted in an U_{en} of < 1.0.

4.3.4.3 Management of Environmentally Assisted Fatigue

As indicated in Table 4.3-2, the environmentally adjusted CUF for most locations is less than 1.0. However, HPI/MU nozzle stainless steel safe end and associated Alloy 82/182 weld have environmentally adjusted CUFs greater than 1.0. FENOC will replace the HPI/MU nozzle safe end and associated Alloy 82/182 weld prior to entering the period of extended operation. 60 year cycle projections were used in the evaluation of U_{en} for the RV inlet and outlet nozzles, HPI/MU nozzles safe end, and pressurizer surge nozzle and attached safe end. Sixty-year cycle projections and a best estimate prediction of total HU/CDs of 114 at 60 years were used in the evaluation of U_{en} for the pressurizer surge line. The remaining locations are qualified for environmentally-assisted fatigue for the full set of NSSS design cycles.

The Davis-Besse Fatigue Monitoring Program will manage the effects of environmentally assisted fatigue for each NUREG/CR-6260 location by counting the design transients on which these environmentally adjusted analyses are based, and assuring that appropriate action is taken prior to any transient approaching its analyzed number of cycles.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of environmentally assisted fatigue will be managed for the period of extended operation by the Fatigue Monitoring Program.

A commitment is provided in Appendix A to replace all four high pressure injection / makeup nozzle safe ends prior to the period of extended operation. In addition, FENOC commits to evaluate the environmental effects of the replacement HPI nozzle safe ends and associated welds in accordance with NUREG/CR-6260 and the guidance of EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application. Any nickel-based alloy locations will be evaluated in accordance with NUREG/CR-6909.

4.4 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The Davis-Besse Environmental Qualification (EQ) of Electrical Components 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, environmentally qualified components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the limits established in the evaluation. The EQ program ensures that the environmentally qualified components are maintained in accordance with their qualification bases. Equipment qualification evaluations for environmentally qualified components that specify a qualification of at least 40 years are considered TLAAs for license renewal.

Under 10 CFR 54.21(c)(1)(iii) the Environmental Qualification program, which implements the requirements of 10 CFR 50.49 (as further defined and clarified by the Division of Operating Reactors Guidelines, NUREG-0588, Regulatory Guide 1.89 Revision 1, and Regulatory Guide 1.97 Revision 3), 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 Environmental Qualification program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). A discussion of the environmentally qualified component re-analysis attributes is included in the description of the Environmental Qualification (EQ) of Electrical Components Program.

Continued implementation of the Environmental Qualification (EQ) of Electrical Components Program for the period of extended operation ensures that the requirements of 10 CFR 50.49 will continue to be met.

Disposition: 10 CFR 54.21(c)(1)(iii) Environmental qualification of electrical equipment will be managed by the Environmental Qualification (EQ) of Electrical Components Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

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4.5 CONCRETE CONTAINMENT TENDON PRESTRESS

The Davis-Besse containment structure does not include pre-stressed tendons. As described in USAR Section 1.2.10.2, the Davis-Besse containment vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom. The containment vessel is completely enclosed by a reinforced concrete shield building having a cylindrical shape with a shallow dome roof. An annular space is provided between the wall of the containment vessel and the wall of the shield building, and between the top of the containment vessel and the dome of the shield building. With the exception of the concrete under the containment vessel there are no structural ties between the containment vessel and the shield building. Above the foundation slab there is virtually unlimited freedom for differential movement between the containment vessel and the shield building.

Disposition: Not applicable

TLAAs for tendon prestress are not applicable for Davis-Besse, which has a free-standing metal containment.

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4.6 CONTAINMENT FATIGUE ANALYSES

The containment system for the station utilizes a free-standing containment vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel structure designed to withstand a postulated loss-of-coolant accident and to confine a postulated release of radioactive material. The Davis-Besse containment does not have a containment liner plate.

The containment, including the vessel, the penetrations, the relief valves, and internal structures, was reviewed for license renewal. The only TLAAs identified were for the containment vessel and the permanent canal seal plant, which are discussed below.

4.6.1 CONTAINMENT VESSEL

The containment vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom which houses the reactor vessel, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, reactor coolant pumps, steam generators, core flooding tanks, letdown coolers, and normal ventilating system. The containment vessel is a Class B vessel as defined in the ASME Section III, Paragraph N-132, 1968 Edition through Summer 1969 Addenda.

The containment vessel is designed to resist dead loads, LOCA loads, operating loads, external pressure load, temperature and pressure, impingement force and missiles, wind loads, seismic loads, gravity loads, and live loads. The containment vessel meets the requirements of ASME Section III, Paragraph N-415.1; thereby justifying the exclusion of cyclic or fatigue analyses in the design of the containment vessel. Analysis of 400 pressure cycles (from -25 to 120 psi) and 400 temperature cycles (from 30°F to 120°F) were performed against the requirements of ASME Section III, Paragraph N-415.1. To date, the containment vessel has not seen any pressure cycles from -25 to 120 psi. The values of 400 pressure and temperature cycles used to exclude fatigue analyses will not be exceeded for 60 years of operation. Thus, the TLAAs associated with exclusion of fatigue analyses for the containment vessel will remain valid for the period of extended operation.

Disposition:	10 CFR 54.21(c)(1)(i)	The TLAAs excluding the containment vessel from fatigue analysis per ASME Section III, Paragraph N415-1 will remain valid through the period of extended operation.

4.6.2 CONTAINMENT PENETRATIONS

Penetrations (of the containment vessel) conform to the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

Piping penetrations (of the containment vessel) are either large diameter, high energy, hot piping (main steam and feedwater lines) or small diameter lower energy piping (general piping). Each main steam and main feedwater containment penetration consists of 1) process pipe, 2) guard pipe, 3) flued head, and 4) penetration bellows assembly.

Consistent with the exclusion of cyclic fatigue analyses in containment vessel design (see Section 4.6.1), a search of the Davis-Besse CLB did not identify any pressurization cycles or fatigue analyses for containment penetration assemblies.

Disposition:	Not a TLAA	There are no fatigue analyses, and hence no TLAA, associated with the containment vessel penetration assemblies.

4.6.3 PERMANENT CANAL SEAL PLATE

The permanent canal seal plate (also known as permanent reactor cavity seal plate) spans the gap between the reactor vessel and the fuel transfer canal floor, and retains water in the canal when the canal is flooded. The permanent canal seal plate is made up of a support structure that rests on the shield plate and reactor vessel seal ledge and a seal membrane that covers the support structure and is welded to the shield plate and reactor vessel seal ledge. Eight access ports and covers are equally spaced around the permanent canal seal plate to allow for sufficient air flow during normal operations. Multiple shield plate holddown clamps are installed to ensure the shield plate will not fail due to heatup and cooldown loads or core flood line break loads.

The fatigue analysis of the permanent canal seal plate seal membrane, which was installed 2004, shows that the maximum fatigue usage factor, at the inner leg to the reactor vessel seal ledge weld, is based on 50 full heatup/cooldown cycles. As shown in Table 4.3-1, Transient 31A, the permanent canal seal plate is projected to experience 51 heatup/cooldown cycles between installation in 2004 and the end of the period of extended operation. However, the number of occurrences of permanent canal seal plate heatup and cooldown is tracked by the Fatigue Monitoring Program to ensure that action is taken before the analyzed numbers of transients are reached. As such, the effects of aging due to fatigue of the permanent canal seal plate seal membrane are managed for the period of extended operation.

Disposition:	10 CFR 54.21(c)(1)(iii)	The effects of fatigue on the permanent canal seal plate seal membrane will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

4.7.1 LEAK-BEFORE-BREAK

The Reactor Coolant System has been evaluated using the criteria of Standard Review Plan Section 3.6.3, Leak-Before-Break, evaluation procedures (see USAR Sections 3.6.2.2.1 and 3.8.2.3.4.d). In conjunction with General Design Criterion 4 of 10 CFR 50 Appendix A, this allows the exclusion of the dynamic effects of a postulated pipe rupture and excludes cold leg and hot leg breaks from the reactor vessel cavity pressurization analysis post-LOCA.

The leak-before-break (LBB) concept relies on the plant's ability to detect leakage from a through-wall flaw and then take appropriate action before that flaw grows to the point of pipe failure. Topical report BAW-1847 Revision 1 [Reference 4.8-1] presents the LBB topical evaluation of Reactor Coolant System primary piping (36 inch hot leg piping and 28 inch cold leg piping) under normal plus faulted loading conditions over the current term of operation (i.e., 40 years). Report BAW-1847 Revision 1 showed that postulated flaws producing detectable leakage exhibit stable growth, and thus, allow a controlled plant shutdown before any potential exists for catastrophic piping failure. The inputs to these analyses include Reactor Coolant System piping structural loads, leakage flaw size determination, and Reactor Coolant System piping material properties.

The LBB analysis reported in BAW-1847 Revision 1 was performed in accordance with the guidance provided in Section 5.2, Item (d), of NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks."

FENOC received relief to install Alloy 52 weld overlays on the reactor coolant pump suction and discharge nozzles Alloy 82/182 dissimilar metal welds for mitigation of primary water stress corrosion cracking (PWSCC). These welds are located in piping approved for LBB. Therefore, an updated LBB evaluation to reflect the new weld configuration with the weld overlays in place was submitted and has been approved by the NRC [Reference 4.8-17]. These weld overlays were installed during the Cycle 16 refueling outage.

The LBB analysis includes fatigue flaw growth analysis, thermal aging analyses for cast austenitic stainless steel, and PWSCC analyses that could be influenced by time. The time-limited aspects of fatigue flaw growth, thermal aging and PWSCC are addressed separately in the subsections below.

4.7.1.1 Fatigue Flaw Growth

The LBB analysis postulated surface flaws at the piping system locations with the highest stress coincident with the lower bound of the material properties for base metal and welds. The fatigue crack growth analysis for postulated flaws was performed to demonstrate that a surface flaw is likely to propagate in the through-wall direction and develop an identifiable leak before it will propagate circumferentially around the pipe to such an extent that it could cause a double-ended pipe rupture under faulted conditions. The fatigue flaw growth analysis used plant design transients. The updated analysis used 1.5 times the design cycles for the reactor coolant pump suction and discharge weld overlays.

The transient cycles are being monitored by the Fatigue Monitoring Program. If a transient cycle count approaches the allowable design limit, corrective actions are taken. Therefore, the effects of fatigue flaw growth on piping approved for LBB will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue flaw growth on piping approved for LBB will be managed by the Fatigue Monitoring Program for the period of extended operation.

4.7.1.2 Thermal Aging

The only stainless steels in the leak-before-break analysis are the safe ends welded to the reactor coolant pump casings and the pump casings themselves; with the pump casings the only cast stainless steel. The reactor coolant pump casings at Davis-Besse, including the suction and discharge nozzles, are annealed SA 351 CF-8M, and were statically cast.

The updated LBB analysis was based on saturated embrittlement of the cast austenitic stainless steel (CASS) casings such that there is no embrittlement TLAA.

An aging management review of the Reactor Coolant System, including the reactor coolant pumps, has been performed for license renewal (see Section 3.1). Reduction of fracture toughness due to thermal embrittlement of CASS components is an aging effect requiring management for the reactor coolant pump casings and is managed by the Inservice Inspection Program. The acceptability of a 10-year inspection interval for these weld overlays was demonstrated in the updated LBB analysis. This analysis does not justify operation of the weld overlays for the life of the plant, but for the 10 years between inspections. Therefore, the effects of thermal aging on CASS components in the approved LBB piping will be managed by the Inservice Inspection Program for the period of extended operation.

Disposition: Not a TLAA. The effects of thermal aging on CASS components in the approved LBB piping will be managed by the Inservice Inspection Program for the period of extended operation.

4.7.1.3 Primary Water Stress Corrosion Cracking

FENOC received relief to install weld overlays on certain Alloy 600 components and Alloy 82/182 dissimilar metal welds for mitigation of PWSCC. As presented in Section 4.7.1, this relief included Alloy 82/182 dissimilar metal welds that are located in piping approved for LBB. FENOC updated the original leak-before-break calculations for Davis-Besse with an evaluation demonstrating that the weld overlays resolve the concerns for original welds susceptibility to primary water stress corrosion cracking. Critical crack sizes and leakage rates with the weld overlay in place were evaluated to demonstrate that margins exist for detection of leakage, i.e., the conclusions of the existing leak-before-break analysis remain valid.

For license renewal, an aging management review of the Reactor Coolant System, including the nickel-alloy weld locations, has been performed (see Section 3.1). Cracking due to PWSCC is an aging effect requiring management for the period of extended operation and is managed by the Inservice Inspection Program and Nickel-Alloy Management Program.

Disposition: Not a TLAA. The effects of PWSCC on the Reactor Coolant System piping will be managed by the Inservice Inspection Program and Nickel-Alloy Management Program for the period of extended operation.

4.7.2 METAL CORROSION ALLOWANCE FOR PRESSURIZER INSTRUMENT NOZZLES

USAR Section 5.2.3.2 indicates that pressurizer nozzle repairs and replacements have resulted in a portion of the carbon steel pressurizer nozzle bore being exposed to reactor coolant. This resulted in an increase of the general corrosion rate of the pressurizer shell base metal in the nozzle bores from zero to 1.42 thousandths of an inch (mils) per year. Over the 9 years from the installation of this modification to the end of the original licensed period, this will result in a loss of 13 mils of the pressurizer carbon steel shell in the nozzle annular regions. The allowable radial corrosion limit, calculated per ASME Section III, is 293 mils for the level instrument nozzles, 493 mils for the sample nozzle and 495 mils for the vent and thermowell nozzles. This corrosion analysis is a TLAA.

The projected loss of material can be extrapolated to 60-years by multiplying the 1.42 mils per year corrosion rate times the 29 years from the date of installation to the end of the period of extended operation. The projected loss of 41.2 mils (29 x 1.42) remains below the allowable radial corrosion limits.

Disposition: 10 CFR 54.21(c)(1)(ii) The metal corrosion allowance TLAA for the pressurizer nozzle annular regions has been projected through the period of extended operation.

4.7.3 REACTOR VESSEL THERMAL SHOCK DUE TO BORATED WATER STORAGE TANK WATER INJECTION

USAR Section 5.2 addresses integrity of the reactor coolant pressure boundary and the analysis to demonstrate that the reactor vessel can safely accommodate the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at the end of the vessel's design life. The analysis documents the reactor vessel integrity during a small steam line break, which creates a pressurized thermal shock condition. This transient generates the greatest level of stress in the reactor vessel. Technical Specifications allow the borated water storage tank (BWST) water temperature to be as low as 35°F. The analysis was revised for license renewal to use reactor vessel embrittlement values that bound the period of extended operation.

The revised fracture mechanics analysis evaluated the integrity of the reactor vessel against PTS for 52 EFPY considering the 35°F minimum temperature for the BWST. Several locations in the reactor vessel were analyzed for PTS, and all locations have demonstrated service life greater than 52 EFPY. Flaws do not initiate for any of the postulated flaw depths. The minimum critical margin to applied pressure margin is 2.21 at the nozzle belt forging.

In addition, as addressed in Section 4.2.3, the vessel's compliance with 10 CFR 50.61 has been assessed. All RT_{PTS} values are below the screening criteria at 60 years.

Disposition: 10 CFR 54.21(c)(1)(ii) The reactor vessel integrity analysis has been projected to the end of the period of extended operation.

4.7.4 HIGH PRESSURE INJECTION/MAKEUP NOZZLE THERMAL SLEEVES

During the Cycle 5 refueling outage, Davis-Besse discovered a failed thermal sleeve for high pressure injection (HPI)/makeup nozzle A-1. Corrective actions included assessment and preservation of the structural integrity of the nozzle, which had experienced thermal cycling due to the thermal sleeve failure. The makeup flow path was re-routed from nozzle A-1 to nozzle A-2 during the Cycle 6 refueling outage (1990) as one of the corrective actions. Fracture mechanics analysis of thermal sleeve life under various makeup flow cycling conditions predicted a thermal sleeve lifetime exceeding 20 eighteen-month operating cycles under current makeup flow control conditions.

Since that analysis, Davis-Besse had an extended (approximately two year) Cycle 13 refueling outage, converted to a 24-month fuel cycle, and performed a measurement uncertainty recapture power uprate. The corresponding predicted end-of-life for the HPI/makeup nozzle thermal sleeve is approximately 2022, based on the predicted number of makeup thermal cycles. The current operating license for Davis-Besse will expire in April of 2017. Davis-Besse will replace all four makeup nozzle thermal sleeves prior to the period of extended operation. The commitment to replace these thermal sleeves is found in Appendix A to this application.

Disposition: 10 CFR 54.21(c)(1)(iii) Cracking of the HPI/makeup thermal sleeve will be managed through the period of extended operation by the Fatigue Monitoring Program. In addition, a FENOC commitment to replace the thermal sleeves prior to the period of extended operation is contained in Appendix A of the License Renewal Application.

4.7.5 INSERVICE INSPECTION – FRACTURE MECHANICS ANALYSES

10 CFR 50.55a(g) requires an Inservice Inspection program to verify the integrity of the reactor coolant pressure boundary. ASME Section XI, Table IWB-2500-1 requires the use of nondestructive examination techniques to detect and characterize flaws. Flaws detected during examination are compared to acceptance standards established in ASME Section XI. Unacceptable flaws require detailed analyses, repair, or replacement.

Acceptance via fracture mechanics analysis requires a prediction of flaw growth considering a chosen evaluation period, i.e., no shorter than the time until the next inspection following discovery of the flaw or as long as the remaining service life of the plant. Flaw indications that are determined not to grow beyond acceptance limits during the evaluation period are justified for continued operation. Fracture mechanics analyses performed for the life of the plant are TLAAs that typically involve the same design transient cycle assumptions considered in the current licensing basis.

A search of Davis-Besse inservice inspection reports and docketed correspondence was performed. Two flaw growth analyses were identified as TLAAs and are evaluated below.

4.7.5.1 Reactor Coolant System Loop 1 Cold Leg Drain Line Weld Overlay Repair

FENOC performed a full structural overlay repair for an axial indication found on the Reactor Coolant System Loop 1 cold leg drain line during the Cycle 14 refueling outage. The structural weld overlay of the cold leg drain nozzle was designed consistent with the requirements of ASME Section XI; Code Case N-504-2; non-mandatory Appendix

Q; and was supplemented by additional design considerations specific to the unique nature of the geometry and materials of the cold leg drain nozzle-to-elbow weld.

The overlay is designed as a full structural overlay that assumes the as-found flaw propagates to a 100% through-wall 360-degree crack rather than performing a crack growth analysis of the as-found flaw. Thus there is no time dependency in the weld overlay design.

The fatigue analysis for the repaired configuration conservatively estimated cycles for 60 years at 1.5 times the original design cycles. Because this analysis is based on a specific number of cycles, it is a TLAA. The Fatigue Monitoring Program manages the effects of fatigue on the reactor coolant system drain line weld overlay repair by counting the thermal cycles incurred through the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the reactor coolant system cold leg drain line nozzle weld overlay repair will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.7.5.2 OTSG 1-2 Flaw Evaluations

During the Cycle 5 refueling outage (May 1988) a number of flaw indications were detected in steam generator 1-2, both in the shell near the steam outlet nozzle and in the shell welds near the lower tubesheet-to-shell juncture. Two of the indications in the shell near the steam outlet nozzle were evaluated according to ASME Section XI, with the remaining shell indications bounded by those evaluated. Five of the indications in the shell welds near the lower tubesheet-to-shell juncture were evaluated, with the remaining shell weld indications bounded by those evaluated.

Simplified evaluation of fatigue crack growth, based on 240 heatup and cooldown cycles, concluded that there would be only slight crack growth, and the indications were found to be acceptable by ASME Section XI, IWB-3612 standards. Because these analyses are based on a specific number of cycles, they are TLAAs. The Fatigue Monitoring Program manages the effects of fatigue on steam generator flaw evaluations by counting the thermal cycles incurred through the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(iii) The effects of fatigue on the steam generator flaw growth will be managed for the period of extended operation by the Fatigue Monitoring Program.

4.8 REFERENCES

4.8-1 AREVA NP Document BAW-1847, "Leak-Before-Break Evaluation of Margins Against Full Break for RCS Piping of B&W Designed NSS," Revision 1

4.8-2 AREVA NP Document BAW-2178P-A, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group, for Level C & D Service Loads," Revision 0

4.8-3 AREVA NP Document BAW-2192P-A, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group, for Level A & B Conditions," Revision 0

4.8-4 AREVA NP Document BAW-2222, "Reactor Vessel Working Group Response to Closure Letters to NRC Generic Letter 92-01, Revision 1," June 1994

4.8-5 AREVA NP Document BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity, B&W Owners Group – Reactor Vessel Working Group," Revision 1

4.8-6 AREVA NP Document BAW-2241P-A, "Fluence and Uncertainty Methodologies," April 1999 (NRC Safety Evaluation Report included)

4.8-7 AREVA NP Document BAW-10013-A, "Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones under Austenitic Stainless Steel Weld Cladding," Last Revised February 15, 1972

4.8-8 AREVA NP Document BAW-10046A, "Method of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," Revision 4

4.8-9 Electric Power Research Institute (EPRI) Report TR-105090, "Guidelines to Implement the License Renewal Technical Requirements of 10CFR54 for Integrated Plant Assessments and Time-Limited Aging Analyses," November 1995

4.8-10 BAW-2127, "Final Submittal for Nuclear Regulatory Commission Bulletin 88-11 'Pressurizer Surge Line Thermal Stratification'," December 1990

4.8-11 BAW-2127, Supplement 1, "Plant-Specific Analysis in Response to Nuclear Regulatory Commission Bulletin 88-11 'Pressurizer Surge Line Thermal Stratification,' Davis-Besse Nuclear Power Station Unit 1," September 1991 - replaced by Supplement 3

4.8-12 BAW-2127, Supplement 2, "Pressurizer Surge Line Thermal Stratification for the B&W 177-FA Nuclear Plants, Summary Report, Fatigue Stress Analysis of the Surge Line Elbows," May 1992

4.8-13 BAW-2127, Supplement 3, "Plant-Specific Analysis in Response to Nuclear Regulatory Commission Bulletin 88-11 'Pressurizer Surge Line Thermal Stratification,' Davis-Besse Nuclear Power Station Unit 1," December 1993

4.8-14 AREVA NP Document BAW-2308-01-A, "Initial RT_{NDT} Of Linde 80 Weld Materials," August 2005 (NRC SER Included)

4.8-15 AREVA NP Document BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," August 1999 (NRC SER included)

4.8-16 FENOC Letter L-09-225, Barry S. Allen to USNRC Document Control Desk, "Supplemental Information Related to a License Amendment Request to Incorporate the Use of Alternate Methodologies for the Development of Reactor Pressure Vessel Pressure-Temperature Limit Curves, and Request for Exemption from Certain Requirements Contained in 10 CFR 50.61 and 10 CFR 50, Appendix G (TAC No. ME1127) – License Amendment Request, ME1128 – Exemption Request)," December 18, 2009 (ADAMS, ML093570103)

4.8-17 NRC Letter, Michael Mahoney (NRC), to Barry S. Allen (FENOC), Davis-Besse Nuclear Power Station, Unit 1 - Issuance of Amendment Regarding Application To Update the Leak-Before-Break Evaluation for the Reactor Coolant Pump Suction and Discharge Nozzle Dissimilar Metal Welds (TAC No. ME2310), March 26, 2010

APPENDIX A

UPDATED SAFETY ANALYSIS REPORT SUPPLEMENT

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A.0 INTRODUCTION

This appendix provides the information to be submitted in an Updated Safety Analysis Report (USAR) Supplement as required by 10 CFR 54.21(d) for the Davis-Besse License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Section 3 contains the results of the aging management reviews. The programs and activities credited to manage the effects of aging are described in Appendix B. Section 4 documents the evaluations of time-limited aging analyses for the period of extended operation. Section 3, Section 4, and Appendix B have been used to prepare the program and activity descriptions that are contained in this appendix.

This appendix is divided into three sections:

- Section A.1 contains summary descriptions of the programs and activities credited to manage the effects of aging during the period of extended operation;
- Section A.2 contains summaries of the evaluations of time-limited aging analyses for the period of extended operation;
- Section A.3 contains a listing of the commitments associated with license renewal.

The information presented in these three sections will be incorporated into the Davis-Besse USAR following issuance of the renewed operating license in accordance with 10 CFR 50.71(e).

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A.1 SUMMARY DESCRIPTIONS OF AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The license renewal integrated plant assessment and evaluation of time-limited aging analyses identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. This section describes the aging management programs and activities identified during the integrated plant assessment and evaluation of time-limited aging analyses. The aging management programs and activities will be implemented as identified in the list of license renewal commitments (see Table A-1). The aging management programs identified as necessary in association with the evaluation of time-limited aging analyses are described in Sections A.1.14 and A.1.16.

Three elements of an effective aging management program that are common to each of the aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the Quality Assurance Program Manual for Davis-Besse, which implements the requirements of 10 CFR 50, Appendix B. The corrective actions, confirmation process, and administrative controls in the Quality Assurance Program Manual, to be applied to the credited aging management programs and activities for the structures and components determined to require aging management, are consistent with the related discussions in the Appendix on Quality Assurance for Aging Management Programs in NUREG-1801, Volume 2.

A.1.1 10 CFR PART 50, APPENDIX J PROGRAM

The 10 CFR Part 50, Appendix J Program monitors Containment leak rate. Containment leak rate tests are required to assure that: (a) leakage through primary Containment, and systems and components penetrating primary Containment, shall not exceed allowable values specified in the Technical Specifications, and (b) periodic surveillance of primary Containment penetrations and isolation valves is performed so that proper maintenance and repairs are made. Appendix J, Option B, is utilized. The Containment leak rate tests are performed in accordance with the guidelines contained in NRC Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program [Reference A.1-1], as modified by approved exceptions; and NEI 94-01, Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J [Reference A.1-2].

A.1.2 ABOVEGROUND STEEL TANKS INSPECTION PROGRAM

The Aboveground Steel Tanks Inspection Program manages the effects of corrosion on the external surfaces and inaccessible locations of the steel fire water storage tank and diesel oil storage tank. The Aboveground Steel Tanks Inspection Program is a condition monitoring program that consists of periodic visual inspections of tank external surfaces and volumetric examinations of tank bottoms.

A.1.3 AIR QUALITY MONITORING PROGRAM

The Air Quality Monitoring Program is a preventive program that is implemented via periodic sampling of the air for hydrocarbons, dew point, and particulates. The Air Quality Monitoring Program ensures that the system remains dry and free of contaminants, such that there are no aging effects which require management.

A.1.4 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program is a combination of existing activities that rely on manufacturer and vendor information, as well as on industry recommendations, such as contained in EPRI Reports TR-104213 [Reference A.1-3] and TR-111472 [Reference A.1-4], for a comprehensive bolting and bolting maintenance program addressing proper selection, assembly, and maintenance of bolting for pressure-retaining closures and structural connections. The program also includes preventive measures to preclude or minimize loss of preload and cracking.

The Bolting Integrity Program includes, through the Inservice Inspection Program, Inservice Inspection (ISI) Program – IWE, Inservice Inspection (ISI) Program – IWF, Structures Monitoring Program, and External Surfaces Monitoring Program, the periodic inspection of bolting for indications of degradation such as leakage, loss of material due to corrosion, loss of preload, and cracking.

A.1.5 BORAL® MONITORING PROGRAM

The Boral® Monitoring Program detects degradation of Boral® neutron absorbers in the spent fuel storage racks by in situ testing. From the monitoring data, the stability and integrity of Boral® in the storage cells are assessed.

A.1.6 BORIC ACID CORROSION PROGRAM

The Boric Acid Corrosion Program manages the effects of boric acid leakage on the external surfaces of in-scope structures and components potentially exposed to boric acid leakage. The Boric Acid Corrosion Program is a condition monitoring program consisting of visual inspections.

A.1.7 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program manages the effects of corrosion on the external surfaces of piping, tanks and associated bolting exposed to a buried (soil) environment. The Buried Piping and Tanks Inspection Program is a combination of a mitigation program (consisting of protective coatings) and a condition monitoring program (consisting of visual inspections).

A.1.8 CLOSED COOLING WATER CHEMISTRY PROGRAM

The Closed Cooling Water Chemistry Program mitigates damage due to loss of material, cracking, and reduction in heat transfer of components that are within the scope of license renewal and contain closed cooling water. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material, cracking, or reduction in heat transfer through proper monitoring and control of corrosion inhibitor concentrations consistent with the current EPRI water chemistry guideline.

The Closed Cooling Water Chemistry Program includes corrosion rate measurement at selected locations in the closed cooling water systems and is supplemented by the One-Time Inspection, which provides verification of the effectiveness of the program in managing the effects of aging.

A.1.9 COLLECTION, DRAINAGE, AND TREATMENT COMPONENTS INSPECTION PROGRAM

The Collection, Drainage, and Treatment Components Inspection Program consists of visual inspections of the surfaces of in-scope steel and other metal components exposed to raw (untreated) water, that are not covered by other aging management programs, for evidence of loss of material, as well as cracking of susceptible materials, or reduction in heat transfer for susceptible components. This program is implemented via opportunistic inspections during periodic maintenance, repair, and surveillance activities when the surfaces are made available for inspection, or via focused inspection. These inspections ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation.

A.1.10 CRANES AND HOISTS INSPECTION PROGRAM

The Cranes and Hoists Inspection Program manages loss of material for structural components of cranes (including bridge, trolley, rails, and girders), monorails, and hoists within the scope of license renewal through periodic visual inspection of structural members for signs of corrosion and wear. The cranes, monorails and hoists within the scope of license renewal are those defined by NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and light load handling systems related to refueling.

The Cranes and Hoists Inspection Program is based on guidance contained in ANSI B30.2 [Reference A.1-5] for overhead and gantry cranes, ANSI B30.11 [Reference A.1-6] for monorail systems and underhung cranes, and ANSI B30.16 [Reference A.1-7] for overhead hoists.

A.1.11 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS INSPECTION

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection detects and characterizes the aging of metallic electrical connections within the scope of license renewal. The one-time inspection uses thermography (augmented by the optional use of contact resistance testing) to detect loose or degraded connections that lead to increased resistance for a representative sample of metallic electrical connections in various plant locations.

A.1.12 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging of cables and connections that are not required to be environmentally qualified but are within the scope of license renewal and subject to adverse localized environments.

Cables and connections subject to an adverse localized environment are managed by visual inspection. Accessible electrical cables and connections installed in adverse localized environments are visually inspected for signs of accelerated age-related degradation such as embrittlement, discoloration, cracking, or surface contamination.

A.1.13 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS PROGRAM

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program manages the potential loss of insulation resistance for high voltage, low current, sensitive instrument circuits that are subject to adverse localized environments (heat, radiation, and moisture in the presence of oxygen). The program is applicable to in-scope neutron monitoring and radiation monitoring circuits and utilizes testing of the cable assemblies for the subject circuits to determine if the cable insulation resistance is degrading.

A.1.14 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRICAL COMPONENTS PROGRAM

The Environmental Qualification (EQ) of Electrical Components Program implements the requirements of 10 CFR 50.49 (as further defined and clarified by the Division of Operating Reactors (DOR) Guidelines [Reference A.2-10], NUREG-0588 [Reference A.2-11], Regulatory Guide 1.89 [Reference A.2-12], and Regulatory Guide 1.97 [Reference A.2-13]). The program demonstrates that subject electrical components located in harsh plant environments are qualified to perform their safety functions in those harsh environments, consistent with 10 CFR 50.49 requirements. The program manages component thermal, radiation, and cyclical aging, as applicable, through the use of aging evaluations. The program requires action to be taken before individual components in the scope of the program exceed their qualified life. Actions taken to maintain qualification include replacement of piece parts, replacement of complete components, or reanalysis.

As required by 10 CFR 50.49, EQ components not qualified to the end of the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Some aging evaluations for EQ components specify a qualification of at least 40 years and are considered time-limited aging analyses for license renewal. The program ensures that these EQ components are maintained within the bounds of their qualification bases.

Reanalysis of an aging evaluation to extend a component qualification is performed on a routine basis as part of the program. Important attributes for the reanalysis of an aging evaluation include analytical models, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met).

A.1.15 EXTERNAL SURFACES MONITORING PROGRAM

The External Surfaces Monitoring Program is a condition monitoring program that consists of periodic visual inspections and surveillance activities of in-scope mechanical component external surfaces to manage loss of material, including loss of material for internal surfaces where the environment is the same as the external environment.

In addition, the External Surfaces Monitoring Program includes opportunistic inspection of external component surfaces that are inaccessible or not readily visible during either normal plant operations or refueling outages.

Also, the External Surfaces Monitoring Program, supplemented by the One-Time Inspection, includes inspection and surveillance of elastomers and polymers that are exposed to air-indoor uncontrolled and air-outdoor environments, but are not replaced on a set frequency or interval (i.e., are long-lived), for evidence of cracking and change in material properties (hardening and loss of strength).

The External Surfaces Monitoring Program also includes inspection of control room emergency ventilation system air-cooled condensing unit cooling coil tubes and fins and station blackout diesel generator radiator tubes and fins (exposed to an air-outdoor environment) for conditions that could result in a reduction in heat transfer, evidenced by visible dirt or other foreign material buildup on tube and fin external surfaces.

A.1.16 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program manages fatigue of select primary and secondary components; including the reactor vessel, reactor internals, pressurizer and steam generators; by tracking thermal cycles as required by Technical Specification 5.5.5, "Component Cyclic or Transient Limit."

The Fatigue Monitoring Program uses the systematic counting of plant transient cycles to ensure that the numbers of design cycles are not exceeded, thereby ensuring that component fatigue usage limits are not exceeded.

The Fatigue Monitoring Program acceptance criteria are to maintain the number of counted transient cycles below the analyzed number of cycles for each transient. The program periodically updates the cycle counts. When the accumulated cycles approach the design cycles, corrective action is taken to ensure the design cycles is not exceeded. Corrective action may include update of the fatigue usage calculation. Any re-analysis uses an NRC-approved version of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) or NRC-approved alternative (e.g., NRC-approved ASME Code case) to determine a valid cumulative usage factor.

For license renewal, the effects of the reactor coolant environment on component fatigue life are addressed by assessing the impact of the environment on a sample of critical components, as identified in NUREG/CR-6260 [Reference A.1-8]. Environmental effects are evaluated in accordance with NUREG/CR-6260 and the guidance of EPRI Technical Report MRP-47 [Reference A.1-9]. Components identified in NUREG/CR-6260 are evaluated using material specific guidance presented in NUREG/CR-6583 [Reference A.1-10] and NUREG/CR-5704 [Reference A.1-11].

A.1.17 FIRE PROTECTION PROGRAM

The Fire Protection Program is a combination condition and performance monitoring program, comprised of tests and inspections that follow the applicable National Fire Protection Association (NFPA) recommendations. The Fire Protection Program manages, through visual inspections and functional tests, as appropriate, the aging effects on fire barrier penetration seals, fire wraps, fire-rated doors and fire barrier walls, ceilings, and floors that perform a current licensing basis fire barrier intended function. The Fire Protection Program also supplements the Fuel Oil Chemistry Program for managing the aging effects on the diesel fire pump fuel oil supply line.

A.1.18 FIRE WATER PROGRAM

The Fire Water Program (sub-program of the overall Fire Protection Program) is credited with aging management of the fire water supply and water-based fire suppression components in the scope of license renewal. Periodic inspection and testing of the fire water supply and water-based fire suppression systems provide reasonable assurance that the supply and suppression components will remain capable of performing their intended functions. Periodic inspection and testing activities include hydrant and hose station inspections, fire main flushes, flow tests, tank inspections, and sprinkler system inspections. The Fire Water Program is a condition monitoring program that comprises tests and inspections based on NFPA recommendations.

The Fire Water Program also includes: 1) NFPA 25 [Reference A.1-18] identified sprinkler head sampling or replacement prior to 50 years of service (in-place), 2) periodic ultrasonic testing (or internal visual inspection, if certain conditions are met) of representative above-ground piping that contains, or has contained, stagnant water, and 3) opportunistic or focused internal visual inspection of buried fire protection piping.

A.1.19 FLOW-ACCELERATED CORROSION (FAC) PROGRAM

The Flow-Accelerated Corrosion (FAC) Program manages loss of material for steel components that are within the scope of license renewal and are exposed to single-phase water above 190°F or two phase steam at any temperature in systems that are susceptible to flow-accelerated corrosion, also called erosion-corrosion. The Flow-Accelerated Corrosion (FAC) Program combines the elements of predictive analysis, baseline inspections, and periodic inspections (to monitor wall-thinning) to monitor and predict wall thickness in susceptible locations. The program is a condition monitoring program that implements the recommendations of NRC Generic Letter 89-08, Erosion/Corrosion – Induced Pipe Wall Thinning [Reference A.1-17] and follows the guidance and recommendations of EPRI NSAC-202L [Reference A.1-12], to ensure that the integrity of piping systems susceptible to flow-accelerated corrosion is maintained.

A.1.20 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program monitors and maintains fuel oil quality in order to mitigate damage due to loss of material, as well as due to cracking of susceptible materials, for the storage tanks and associated piping and components containing fuel oil that are within the scope of license renewal. The program includes verifying the quality of new fuel oil, periodic sampling of stored diesel fuel oil, and periodic cleaning and inspection for emergency diesel generator, diesel fire pump, and station blackout diesel generator fuel oil tanks and associated components. The Fuel Oil Chemistry Program manages the presence of contaminants, such as water or microbiological organisms, that could lead to the onset and propagation of loss of material or cracking (of susceptible material) through proper monitoring and control of fuel oil contamination consistent with plant Technical Specifications and ASTM International (ASTM) standards for fuel oil. The Fuel Oil Chemistry Program is a mitigation program.

The effectiveness of the Fuel Oil Chemistry Program is verified by the One-Time Inspection, which includes ultrasonic thickness measurement of a sample of fuel oil tank bottoms.

A.1.21 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging of inaccessible medium-voltage electrical cables that are not required to be environmentally qualified but are susceptible to aging effects caused by moisture and voltage stress, such that there is reasonable assurance that the cables will perform their intended function in accordance with the current licensing basis during the period of extended operation.

Inaccessible medium-voltage cables within the scope of the program and exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation.

The program also requires periodic inspection of electrical manholes associated with in-scope medium-voltage cables for water accumulation, and requires the removal of water from the electrical manholes as necessary.

A.1.22 INSERVICE INSPECTION (ISI) PROGRAM – IWE

The Inservice Inspection (ISI) Program – IWE establishes responsibilities and requirements for conducting ASME Code, Section XI, Subsection IWE (IWE) inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWE includes examination and testing of accessible surface areas of the steel containment; containment hatches and airlocks; seals, gaskets and moisture barriers; and containment pressure-retaining bolting in accordance with the requirements of IWE.

The inservice examinations conducted throughout the service life of Davis-Besse will comply with the requirements of the ASME Code Section XI edition and addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC.

A.1.23 INSERVICE INSPECTION (ISI) PROGRAM – IWF

The Inservice Inspection (ISI) Program – IWF establishes responsibilities and requirements for conducting ASME Code, Section XI, Subsection IWF (IWF) inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWF includes visual examination of supports based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Classes 2 and 3). The primary inspection method is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. Supports determined to be unacceptable for continued service requiring corrective actions are re-examined during the next inspection period in accordance with the requirements of IWF.

The inservice examinations conducted throughout the service life of Davis-Besse will comply with the requirements of the ASME Code Section XI edition and addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC.

A.1.24 INSERVICE INSPECTION PROGRAM

The Inservice Inspection Program manages cracking of reactor coolant pressure boundary components and once-through steam generator secondary-side components. The Inservice Inspection Program also manages reduction in fracture toughness of cast austenitic stainless steel pump casings and valve bodies. In addition, the Inservice Inspection Program, in conjunction with the PWR Water Chemistry Program, manages loss of material for once-through steam generator secondary-side components.

The Inservice Inspection Program is a condition monitoring program that meets the inservice inspection requirements specified by the ASME Code, Section XI, Division 1, including Subsections IWB, IWC, and IWD, as modified by 10 CFR 50.55a. The Inservice Inspection Program includes augmented examinations that correspond to commitments made to the regulatory authorities beyond the ASME Code requirements.

The inservice examinations (and pressure tests) conducted throughout the service life of Davis-Besse will comply with the requirements of the ASME Code Section XI, Subsections IWB, IWC, and IWD, edition and addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC.

A.1.25 LEAK CHASE MONITORING PROGRAM

The Leak Chase Monitoring Program is a condition monitoring program, consisting of observation and activities to detect leakage from the spent fuel pool, the fuel transfer pit, and the cask pit liners due to age-related degradation.

The Leak Chase Monitoring Program includes periodic monitoring of the spent fuel pool, the fuel transfer pit, and the cask pit liners leak chase system. Periodic monitoring of leakage from the leak chase system permits early determination and localization of leakage.

A.1.26 LUBRICATING OIL ANALYSIS PROGRAM

The Lubricating Oil Analysis Program mitigates age-related degradation due to loss of material and reduction in heat transfer due to fouling for plant components that are within the scope of license renewal and that are exposed to a lubricating oil environment. The program requires management of the relevant conditions that could lead to the onset and propagation of loss of material due to crevice, galvanic, general, or pitting corrosion, or reduction in heat transfer due to fouling, through monitoring of the lubricating oil consistent with various manufacturers' recommendations and industry standards. The Lubricating Oil Analysis Program is a mitigation program.

The Lubricating Oil Analysis Program is supplemented by the One-Time Inspection, which provides verification of the effectiveness of the program in mitigating the effects of aging.

A.1.27 MASONRY WALL INSPECTION

The Masonry Wall Inspection, implemented as part of the Structures Monitoring Program, consists of inspection activities to detect cracking of masonry walls and degradation of steel edge supports and bracing on masonry walls within the scope of license renewal. Masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program. The Masonry Wall Inspection performs visual inspection of external surfaces of masonry walls.

A.1.28 NICKEL-ALLOY MANAGEMENT PROGRAM

The Nickel-Alloy Management Program manages primary water stress corrosion cracking (PWSCC) and stress corrosion cracking / intergranular attack (SCC/IGA) of nickel-alloy pressure boundary components other than reactor vessel closure head nozzles and steam generator tubes. The Nickel-Alloy Management Program is a combination mitigative and condition monitoring program.

The Nickel-Alloy Management Program uses a number of inspection techniques to detect cracking, including volumetric and bare metal visual examinations. The Nickel-Alloy Management Program implements the inspections of components through the Inservice Inspection Program. Component evaluations, examination methods, scheduling, and site documentation comply with 10 CFR 50, the ASME Code, NRC bulletins and generic letters, and staff-approved industry guidelines related to nickel-alloy issues. The Nickel-Alloy Management Program includes mitigation and repair activities to ensure long-term operability of nickel-alloy components.

A.1.29 NICKEL-ALLOY REACTOR VESSEL CLOSURE HEAD NOZZLES PROGRAM

The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program manages cracking of the control rod drive nozzles and welds in the reactor vessel closure head, and the Boric Acid Corrosion Program manages wastage of associated reactor vessel closure head surfaces. The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program ensures that inservice inspections of all nickel-alloy reactor vessel closure head penetration nozzles, and associated reactor vessel closure head surfaces, will continue to be performed in accordance with ASME Code Case N-729-1 as modified by 10 CFR 50.55a Section (g)(6)(ii)(D).

A.1.30 ONE-TIME INSPECTION

One-Time Inspection performs inspections to verify the effectiveness of the Closed Cooling Water Chemistry Program, the Fuel Oil Chemistry Program, the Lubricating Oil Analysis Program, and the PWR Water Chemistry Program, or confirms the absence of aging effects. One-time inspections address situations where: 1) an aging effect is not expected to occur, but it cannot be ruled out with reasonable assurance, 2) an aging effect is expected to progress very slowly in the specified environment, but the local environment may be more adverse, or 3) the characteristics of the aging effect include a long incubation period.

The elements of One-Time Inspection include:

- Determination of a representative sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience;
- Identification of the inspection locations in the system or component based on the aging effect, or based on the areas susceptible to concentration of contaminants that promote certain aging effects;
- Determination of the examination technique, including acceptance criteria that is effective in managing the aging effect for which the component is examined; and
- Evaluation of the need for follow-up examinations to monitor the progression of any age-related degradation.

When evidence of an aging effect is revealed by a one-time inspection, the routine evaluation of the inspection results triggers additional actions to assure the intended function of affected components will be maintained through the period of extended operation.

A.1.31 OPEN-CYCLE COOLING WATER PROGRAM

The Open-Cycle Cooling Water Program manages loss of material due to crevice, galvanic, general, pitting and microbiologically-influenced corrosion; and erosion for in-scope components in the Service Water System and components connected to or cooled by the Service Water System (including the cooling tower makeup water relative to the Circulating Water System), along with cracking of susceptible materials. The program manages fouling due to particulates (e.g., corrosion products) and biological material (micro- and macro-organisms) resulting in reduction in heat transfer for heat exchangers (including condensers, coolers, cooling coils, and evaporators) within the scope of the program.

The Open-Cycle Cooling Water Program consists of inspections, surveillances, and testing to detect and evaluate cracking, fouling, and loss of material, combined with chemical treatments and cleaning activities to minimize cracking, fouling, and loss of material. The program is a combination condition and performance monitoring, and mitigation program that implements the recommendations of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment" [Reference A.1-17] for safety-related equipment in the scope of the program and manages loss of material for in-scope nonsafety-related components that contain service water or cooling tower makeup water.

A.1.32 PWR Reactor Vessel Internals Program

The PWR Reactor Vessel Internals Program manages change in dimension due to void swelling; cracking due to flaw initiation and growth, SCC/IGA, and irradiation-assisted stress corrosion cracking (IASCC); loss of preload due to stress relaxation; reduction in fracture toughness due to radiation and thermal embrittlement; and loss of material due to wear. The PWR Reactor Vessel Internals Program is a condition monitoring program.

The PWR Reactor Vessel Internals Program is based upon the examination requirements for Babcock & Wilcox (B&W) designed pressurized water reactors (PWRs) provided in EPRI Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)" [Reference A.1-14], along with the implementation guidance described in NEI 03-08, "Guideline for the Management of Materials Issues" [Reference A.1-15]. MRP-227 has been submitted to the NRC for review and approval. Following NRC approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and re-issued as MRP-227-A. The PWR Reactor Vessel Internals Program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A.

The EPRI inspection and evaluation guidelines establish the augmented ASME Code Section XI inservice inspection requirements that will be used to monitor for the aging effects that are applicable to certain susceptible or limiting reactor vessel internals components for B&W designed PWRs.

A.1.33 PWR Water Chemistry Program

The PWR Water Chemistry Program mitigates damage due to loss of material, cracking, and reduction in heat transfer of components that are within the scope of license renewal and contain, or are exposed to, treated water or steam in the primary, secondary, or auxiliary systems. The program includes periodic monitoring and control of the known detrimental contaminants that could lead to, or are indicative of, conditions for the onset and propagation of loss of material, cracking, or reduction in heat transfer through proper monitoring and control of chemical concentrations consistent with EPRI primary and secondary water chemistry guidelines.

In addition, the PWR Water Chemistry Program is credited in conjunction with the Nickel-Alloy Management Program, Inservice Inspection Program, Nickel-Alloy Reactor Vessel Closure Head Nozzles Program, PWR Reactor Vessel Internals Program, Steam Generator Tube Integrity Program, and Small Bore Class 1 Piping Inspection to manage the effects of aging for reactor vessel, reactor vessel internals, reactor coolant pressure boundary, and steam generator components.

The PWR Water Chemistry Program is also supplemented by a One-Time Inspection to provide verification of the effectiveness of the program in managing the effects of aging.

A.1.34 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program manages cracking and loss of material for the reactor head closure stud assemblies (studs, nuts, and washers). The Reactor Head Closure Studs Program is a combination mitigative and condition monitoring program.

The Reactor Head Closure Studs Program includes the preventive measures of NRC Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs," [Reference A.1-21] to mitigate cracking, including the use of a stable lubricant that is compatible with the fastener material and the environment. The program provides a specific precaution against the use of compounds containing sulfur (sulfide), including molybdenum disulfide (MoS_2), as a lubricant for the reactor head closure stud assemblies. An approved lubricant is applied to the threaded areas of studs and nuts and to the concave and convex faces of the spherical washers during each assembly.

The Reactor Head Closure Studs Program examines reactor vessel stud assemblies in accordance with the examination and inspection requirements specified in the ASME Code, Section XI, Subsection IWB (1995 Edition through the 1996 Addenda) and approved ASME Code Cases. Visual examinations (VT-2) for leak detection are performed during system pressure tests.

The Reactor Head Closure Studs Program inspections are implemented by the Inservice Inspection Program. The Inservice Inspection Program will continue to comply with the requirements of the ASME Code Section XI edition and addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC.

A.1.35 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is a condition monitoring program that manages reduction of fracture toughness for the low alloy steel reactor vessel shell and welds in the beltline region. Davis-Besse participates in the Pressurized Water Reactor Owners Group (PWROG) Master Integrated Reactor Vessel Surveillance Program (MIRVSP) which includes all seven operating B&W 177-fuel assembly plants and six participating Westinghouse-designed plants having B&W fabricated reactor vessels. The MIRVSP is an NRC-approved program that implements the requirements of Appendix H to 10 CFR Part 50.

Data resulting from the Reactor Vessel Surveillance Program is used to:

- determine pressure-temperature limits, minimum temperature requirements, and end of life upper shelf energy (USE) in accordance with the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements," and

- determine end of life reference temperature for pressurized thermal shock (RT_{PTS}) values in accordance with 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock."

Six surveillance capsules containing Davis-Besse specific materials were inserted into the reactor before initial plant startup. These capsules were designated as TE1-A through TE1-F. The requirements of 10 CFR 50 Appendix H were met by the first four capsules having been withdrawn and tested. The remaining two capsules, TE1-C and TE1-E, have been removed and the materials have not been tested. Capsule TE1-C contains the Davis-Besse limiting material and has been exposed to a fluence slightly above the 60-year projected fluence for the Davis-Besse plant. The Reactor Vessel Surveillance Program is enhanced to require testing of capsule TE1-C. Capsule TE1-E has been discarded.

Since Davis-Besse does not have plant-specific surveillance capsules remaining inside the reactor vessel, ex-vessel cavity dosimetry is used to monitor neutron fluence.

A.1.36 SELECTIVE LEACHING INSPECTION

The Selective Leaching Inspection detects and characterizes the conditions on internal and external surfaces of subject components exposed to raw water, treated water, soil, and moist air (including condensation) environments. The inspection provides direct evidence through visual inspection, hardness measurement, or other appropriate examinations (such as chipping, scraping, or other mechanical means), of whether, and to what extent, loss of material due to selective leaching has occurred.

A.1.37 SMALL BORE CLASS 1 PIPING INSPECTION

The Small Bore Class 1 Piping Inspection will detect and characterize cracking of small bore ASME Code Class 1 piping less than 4 inches nominal pipe size (NPS 4), which includes pipe, fittings, and branch connections. The Small Bore Class 1 Piping Inspection is a condition monitoring program.

The ASME Code does not require volumetric examination of Class 1 small bore piping. The Small Bore Class 1 Piping Inspection is a one-time inspection that consists of volumetric examination of a representative sample of small bore piping locations that are susceptible to cracking. The inspection sample will include both socket welds and butt welds. The sample size and inspection locations are based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small bore piping locations. The guidelines of EPRI Report 1011955, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)" [Reference A.1-13], and the supplemental guidelines 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)" [Reference A.1-20] are considered in selecting the sample size and locations. Volumetric examinations (including qualified destructive or nondestructive techniques) are performed by qualified personnel following procedures that are consistent with Section XI of the ASME Code and 10 CFR 50, Appendix B.

If a qualified non-destructive volumetric examination technique does not become available for socket welds, an opportunistic destructive examination will be conducted. Opportunistic destructive examination is performed when a weld is removed from service for other considerations, such as plant modifications. If a socket weld does not become available on an opportunistic bases, one will be selected for destructive testing. This socket weld will be selected from a piping location that is susceptible to cracking.

A.1.38 STEAM GENERATOR TUBE INTEGRITY PROGRAM

The Steam Generator Tube Integrity Program is credited for aging management of cracking, denting, loss of material, and reduction in heat transfer of the steam generator tubes, as well as cracking of the tube plugs, tube sleeves, and tube support plates.

The Steam Generator Tube Integrity Program is a combination condition monitoring and mitigation program. The Steam Generator Tube Integrity Program is based on the Steam Generator Management program, which meets the intent of the guidance in NEI 97-06, "Steam Generator Program Guidelines" [Reference A.1-16] and the requirements of the Technical Specifications. The Steam Generator Tube Integrity Program also includes secondary-side examinations to assist in verification of tube integrity and the condition of the tube support plates. The program establishes a framework for prevention, inspection, evaluation, repair or removal from service, and leakage monitoring measures.

Primary-side and secondary-side water chemistry control and foreign material exclusion requirements inhibit degradation. Eddy current testing and visual inspections are used for the detection of flaws. Condition monitoring compares the inspection results against performance criteria, and an operational assessment ensures that the performance criteria will be met throughout the next operating cycle.

A.1.39 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program manages age-related degradation of plant structures and structural components within the scope of the program to ensure that each structure or structural component retains the ability to perform its intended function. Aging effects are detected by visual inspection of external surfaces prior to the loss of the structure's or component's intended function. The Structures Monitoring Program encompasses and implements the Water Control Structures Inspection and the Masonry Wall Inspection. The Structures Monitoring Program implements provisions of the Maintenance Rule, 10 CFR 50.65, that relate to structures, masonry walls, and water control structures. Concrete, masonry walls, and other structural components that perform a fire barrier intended function are also managed by the Fire Protection Program.

A.1.40 WATER CONTROL STRUCTURES INSPECTION

The Water Control Structures Inspection, implemented as part of the Structures Monitoring Program, consists of inspection activities to detect age-related degradation. The Water Control Structures Inspection ensures the structural integrity and operational adequacy of the Intake Structure, Forebay, Service Water Discharge Structure, and in-scope structural components within the structures.

A.1.41 REFERENCES

A.1-1 NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995

A.1-2 NEI 94-01, "Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J," Revision 0

A.1-3 EPRI Report TR-104213, "Bolted Joint Maintenance and Applications Guide," December 1995

A.1-4 EPRI Report TR-111472, "Assembling Bolted Connections Using Spiral Wound Gaskets," August 1999

A.1-5 ANSI B30.2, "Overhead and Gantry Cranes," 1976

A.1-6 ANSI B30.11, "Monorail Systems and Underhung Cranes," 1980

A.1-7 ANSI B30.16, "Overhead Hoists (Underhung)," 1981

A.1-8 NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," February 1995

A.1-9 EPRI Report MRP-47, "Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application," Revision 1, September 2005

A.1-10 NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998

A.1-11 NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999

A.1-12 EPRI Report 1011838, "Recommendations for An Effective Flow Accelerated Corrosion Program (NSAC-202L-R3)," May 2006

A.1-13 EPRI Report 1011955, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)," June 2005

A.1-14 EPRI Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," December 2008

A.1-15 NEI 03-08, "Guideline for the Management of Materials Issues," May 2003

A.1-16 NEI 97-06, "Steam Generator Program Guidelines," Revision 2

A.1-17 NRC Generic Letter 89-08, "Erosion/Corrosion, – Induced Pipe Wall Thinning," May 1989

A.1-18 NFPA 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems," 2002

A.1-19 NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," July 1989

A.1-20 EPRI Report 1018330, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines – Supplemental Guidance (MRP-146S)," December 2008

A.1-21 NRC Regulatory Guide 1.65, "Material and Inspection for Reactor Vessel Closure Studs," October 1973

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A.2 EVALUATION SUMMARIES OF TIME-LIMITED AGING ANALYSES

In accordance with 10 CFR 54.21(c), an application for a renewed operating license requires an evaluation of time-limited aging analyses (TLAAs) for the period of extended operation. The following TLAAs have been identified and evaluated to meet this requirement.

A.2.1 INTRODUCTION

Time-limited aging analyses are defined in 10 CFR 54.3(a) as those 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.*

The TLAAs (i.e., each calculation or analysis) that meet all six aspects above, are evaluated in accordance with 10 CFR 54.21(c)(1) to demonstrate that:

(i) *The analyses remain valid for the period of extended operation, or*

(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.*

This section provides a summary of the TLAAs identified in the Davis-Besse License Renewal Application, and includes the following topics:

- Reactor Vessel Neutron Embrittlement (Section A.2.2)
- Metal Fatigue (Section A.2.3)
- Environmental Qualification of Electrical Equipment (Section A.2.4)
- Containment Fatigue (Section A.2.5)
- Inservice Inspection – Fracture Mechanics Analyses (Section A.2.6)
- Other Plant-Specific Time-Limited Aging Analyses (Section A.2.7)

A.2.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

Neutron embrittlement is the term used to describe changes in mechanical properties of reactor vessel materials that result from exposure to fast neutron flux, energy greater than 1.0 mega-electron volts (E>1.0 MeV), within the vicinity of the reactor core called the beltline region. The most pronounced material change is a reduction in fracture toughness. As fracture toughness decreases with cumulative fast neutron exposure, the material's resistance to crack propagation decreases. The rate of neutron exposure is neutron flux ($n/cm^2/sec$) and the cumulative neutron exposure over time is neutron fluence (n/cm^2).

Fracture toughness is also dependent on temperature. The reference temperature for nil-ductility transition (RT_{NDT}) is the temperature above which the material behaves in a ductile manner and below which the material behaves in a brittle manner. As fluence increases, RT_{NDT} increases. This means higher temperatures are required for the material to continue to act in a ductile manner. Determining the projected reduction in fracture toughness as a function of fluence affects several analyses used to support the operation of Davis-Besse:

- Neutron Fluence (Section A.2.2.1)
- Upper Shelf Energy (Section A.2.2.2)
- Pressurized Thermal Shock (Section A.2.2.3)
- Pressure-Temperature Limits (Section A.2.2.4)
- Low-Temperature Overpressure Protection Limits (Section A.2.2.5)
- Intergranular Separation – Underclad Cracking (Section A.2.2.6)
- Reduction in Fracture Toughness of Reactor Vessel Internals (Section A.2.2.7)

Requirements associated with fracture toughness and pressure-temperature limits for the reactor coolant pressure boundary are contained in Appendices G and H of 10 CFR 50.

A.2.2.1 Neutron Fluence

Neutron fluence is not a TLAA, it is a time-limited assumption used in the evaluation of neutron embrittlement TLAAs.

<u>Fluence Projection</u>

The fluence analysis methodology from BAW-2241P-A [Reference A.2-14] was used to calculate the fast neutron fluence (E > 1.0 MeV) of the reactor vessel welds and forgings of interest. The fast neutron fluence at each location was calculated in accordance with the requirements of NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Vessel Neutron Fluence," [Reference A.2-17].

Fluence results were calculated for Cycles 13-14 irradiation using a computer model that extends from below the core to the vessel mating surface. The sum of the End-of-Cycle (EOC) 12 and Cycles 13-14 fluence results in the EOC 14 cumulative fluence. This data was benchmarked against cavity dosimetry data for Cycles 13-14. To extrapolate the fluence values to end of life, Cycle 15 design information was utilized to develop flux projections at each location. These Cycle 15 flux values were used to extrapolate the EOC 14 fluence to 52 effective full power years (EFPY) assuming 100% power at 2,817 MWt and a partial low leakage core design whereby High Thermal Performance fuel assemblies (a total of 12) were introduced on the periphery.

Beltline Evaluation

10 CFR 50.61 defines the reactor vessel beltline as the region of the reactor vessel (shell materials including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most controlling material with regard to radiation damage.

The Davis-Besse beltline for the first 40 years of operation includes the nozzle belt forging (ADB 203), the nozzle belt forging to upper shell forging circumferential weld (WF-232/233), the upper shell forging (AKJ 233), the upper shell forging to lower shell forging circumferential weld (WF-182-1), and the lower shell forging (BCC 241).

For the period of extended operation, the beltline will include all items with 52 EFPY surface fluence greater than 1.0E+17 n/cm^2 (E >1 MeV). The limiting weld with regard to USE, adjusted reference temperature (ART), and reference temperature for pressurized thermal shock (RT_{PTS}) is the upper shell to lower shell weld WF-182-1, as is the case for the first 40 years of operation. The limiting forging with regard to ART and RT_{PTS} is the lower shell forging BCC 241, as is the case at 40 years. Both of these materials are included in the Reactor Vessel Surveillance Program and no additional materials are required for irradiation and testing.

A.2.2.2 Upper-Shelf Energy

10 CFR 50 Appendix G requires the USE for the reactor vessel beltline materials to be no less than 50 ft-lb at all times during plant operation, including the effects of neutron radiation. If USE cannot be shown to remain above this limit, then an equivalent margin analysis (EMA) must be performed to show that the margins of safety against fracture are equivalent to those required by Appendix G of ASME Section XI. Initial (unirradiated) USE values for the Davis-Besse reactor vessel are recorded in USAR Table 5.2-15. As no initial USE is available for the beltline welds (Linde80 welds), operation for 32 EFPY was justified based on an equivalent margins analysis (fracture mechanics analysis).

For license renewal, the initial USE values are projected to 52 EFPY using Regulatory Guide 1.99, Revision 2, Position 1.2. Position 2.2, use of surveillance data, was also used for weld WF-182-1 and lower shell forging BCC 241. All locations are above 50 ft-lb with the exception of weld WF-182-1.

The limiting reactor vessel beltline weld WF-182-1 is the only 60-year (52 EFPY) beltline location with a projected Charpy impact energy level below 50 ft-lbs. The fracture mechanics evaluation of weld WF-182-1 was extended from 40 years (32 EFPY) to 60 years (52 EFPY) based on the projected 52 EFPY neutron fluence values. The analysis demonstrates that the limiting reactor vessel beltline weld satisfies the ASME Code requirements of Appendix K for ductile flaw extensions and tensile stability using projected upper-shelf Charpy impact energy levels for the weld material at 52 EFPY.

Reactor vessel USE and the equivalent margin analyses have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.3 Pressurized Thermal Shock

10 CFR 50.61(a)(2) defines pressurized thermal shock (PTS) as an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. 10 CFR 50.61(b)(2) defines screening criteria for embrittlement of reactor vessel materials in PWRs, and required actions if the screening criteria are exceeded. The screening criteria are based on the RT_{PTS}. The screening criterion for circumferential welds is 300°F maximum and the screening criterion for forgings is 270°F maximum. If the projected RT_{PTS} values remain below the applicable screening temperature, then no corrective actions are required.

For license renewal, a 52 EFPY RT_{PTS} evaluation was performed for the reactor vessel beltline materials. In accordance with 10 CFR 50.61, the RT_{PTS} values were calculated by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT} including a margin term to cover the uncertainties, as prescribed by Regulatory Guide 1.99 Revision 2. The predicted radiation induced ΔRT_{NDT} was calculated using the 52 EFPY neutron fluence at the clad-low alloy steel interface. Initial RT_{NDT} and margins for welds WF-182-1 and WF-232 (Nozzle Belt Forging to Upper Shell Forging Circumferential Weld) were obtained from BAW-2308, Revision 1-A.

All RT_{PTS} values are below the screening criteria at 60 years. The upper to lower shell circumferential weld (WF-182-1) is the limiting material with respect to RT_{PTS}.

Reactor vessel RT_{PTS} has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.4 Pressure-Temperature Limits

10 CFR 50 Appendix G requires the establishment of pressure-temperature (P-T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials. 10 CFR 50, Appendix G requires the use of the ASME Section III, Appendix G to determine the stresses and fracture toughness at locations within the reactor coolant pressure boundary.

The current P-T limits, generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99 Revision 2, are valid until 21 EFPY. A revised pressure and temperature limits report (PTLR) will be submitted to the NRC, in accordance with Technical Specification 5.6.4, before Davis-Besse operates beyond 21 EFPY, in accordance with the requirements of 10 CFR 50, Appendix G. The P-T limit curves, as contained in the PTLR, will be updated as necessary through the period of extended operation as part of the Reactor Vessel Surveillance Program.

Reactor vessel P-T limits will be managed, as part of the Reactor Vessel Surveillance Program, for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.2.5 Low-Temperature Overpressure Protection Limits

Appendix G of ASME Section XI establishes procedures and limits for Reactor Coolant System pressure and temperature primarily for low temperature conditions to provide protection against non-ductile failure of the reactor vessel.

Low-temperature overpressure protection (LTOP) is provided in two ways at Davis-Besse.

1. Administrative controls are used to assure protection within the existing pressure-temperature limits when the pressurizer power-operated relief valve and the safety valves are no longer providing overpressure protection.
2. A relief valve in the Decay Heat Removal System suction piping is placed into service when the Reactor Coolant System temperature is below 280°F.

The current technical specifications for LTOP are valid through 21 EFPY. These technical specifications used an improved methodology to calculate LTOP limits in accordance with generically approved topical report BAW-10046A [Reference A.2-16]. Maintaining the LTOP limits in accordance with Appendix G of ASME Section XI, as required by Appendix G of 10 CFR 50, assures that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

LTOP limits will be managed, as part of the Reactor Vessel Surveillance Program, for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.2.6 Intergranular Separation – Underclad Cracking

Underclad cracking refers to intergranular separations in the heat affected zones of low alloy base metal under austenitic stainless steel cladding in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice, and clad by high-heat-input submerged arc processes. BAW-10013-A [Reference A.2-15] contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to design fatigue cycles. The flaw growth analysis was performed for a 40 year cyclic loading, and an end-of-life assessment of radiation embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report concluded that the intergranular separations found in B&W vessels would not lead to vessel failure. This report was accepted by the Atomic Energy Commission.

Evaluation of intergranular separations for the Davis-Besse SA-508 Class 2 forgings was performed for 60 years using the current fracture toughness information, applied stress intensity factor solutions, and fatigue crack growth correlations for SA-508 Class 2 material. The analysis was applied to two relevant regions of the reactor vessel: the beltline and the nozzle belt. Both axial and circumferential oriented flaws were considered in the evaluation; however, the detailed flaw evaluation was only performed for the bounding axially oriented flaws. The fatigue crack growth analysis considered the normal and upset condition transients with the associated 60-year projected cycles for the period of extended operation. The analysis determined that the postulated underclad cracks in the reactor vessel are acceptable through the period of extended operation.

[Proposed text for this section, pending closure of Confirmatory Action Letter CAL No. 3-10-001 commitments related to the replacement of the Davis-Besse closure head in 2011.] The closure head/head flange was replaced in the Fall of year 2011. This replacement head was fabricated using SA-508 Class 3 material, which is not susceptible to the subject intergranular separations. Therefore, this replacement closure head/head flange is not considered in the underclad cracking evaluation.

Reactor vessel underclad cracking TLAAs have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.7 Reduction in Fracture Toughness of Reactor Vessel Internals

Reduction in fracture toughness of (stainless steel) reactor vessel internals is an aging effect caused by exposure to neutron irradiation. Prolonged exposure to high-energy neutrons results in changes to the mechanical properties, such as an increase in tensile and yield strength, and decreases in ductility and fracture toughness. The extent of reduction in fracture toughness is a function of the material, irradiation temperature, and neutron fluence.

USAR Appendix 4A describes the detailed stress analysis of the reactor vessel internals under accident conditions for the current term of operation. The results of this analysis show that although there is some deflection of the internals, the reactor vessel internals will not fail because the stresses are within established limits.

Evaluation of the impact of the measurement uncertainty recapture (MUR) power uprate on the structural integrity of the reactor vessel internals components concluded that the temperature changes due to the power uprate are bounded by those used in the existing analyses. As part of MUR uprate, FirstEnergy Nuclear Operating Company (FENOC) provided the following commitment:

> "As appropriate, FENOC commits to incorporate recommendations from EPRI's MRP inspection guidelines into the reactor vessel internals program at Davis-Besse Nuclear Power Station, Unit, No. 1."

Integrity of reactor vessel internals will be managed, as part of the PWR Reactor Vessel Internals Program, for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3 METAL FATIGUE

The following sections summarize the analyses associated with metal fatigue of fluid systems:

- Class 1 Code Fatigue Requirements (Section A.2.3.1)
- Class 1 Fatigue Evaluations (Section A.2.3.2)
- Non-Class 1 Fatigue Evaluations (Section A.2.3.3)
- Generic Industry Issues on Fatigue (Section A.2.3.4)

A.2.3.1 Class 1 Code Fatigue Requirements

The ASME Class 1 components for Davis-Besse include the reactor vessel, reactor coolant pressure boundary components, and the once through steam generators. The specific codes and standards to which systems, structures, and components were designed are listed in USAR Table 3.2-2.

Cumulative usage factors for the Class 1 components are calculated based on normal and upset design transient definitions contained in the component design specifications. The design transients used to generate cumulative usage factors for Class 1 components are reported in USAR Table 5.1-8. In accordance with Davis-Besse Technical Specification 5.5.5, provides controls to track the USAR Section 5 cyclic and transient occurrences to ensure that components are maintained within design limits.

Fatigue of Class 1 components is managed by the Fatigue Monitoring Program. This program tracks the occurrence of plant transients that affect fatigue. The number of design cycles originally considered in the design fatigue analyses is not a design limit. The design limit for fatigue is the ASME Code allowable cumulative usage factor of 1.0. The fatigue usage for a component is normally the result of several different thermal transients, coupled with mechanical loads. Exceeding the design cycles for one or more transients does not necessarily imply that fatigue usage will exceed the allowable limit.

A.2.3.1.1 ASME Section III

The primary code governing design and construction of the Class 1 systems and components is the ASME Boiler and Pressure Vessel Code, Section III. The ASME Code requires evaluation of transient thermal and mechanical load cycles and determination of fatigue usage for Class 1 components.

A.2.3.1.2 B31.7 Piping Code

The Davis-Besse reactor coolant system piping, as well as reactor coolant pressure boundary piping in other systems, was designed to American National Standards Institute (ANSI) B31.7 Draft, February 1968 with Errata, June 1968 and also meets the design requirements of ANSI B31.7, 1969 Edition. The ANSI B31.7 Piping Code requires evaluation of transient thermal and mechanical load cycles and determination of fatigue usage for Class 1 piping. The reactor head vent and other piping designated as quality group A, B, or C is designed to ASME Section III, 1971 Edition, Class 1, 2 or 3 respectively. Davis-Besse has no Class 1 piping designed to ANSI B31.1.

A.2.3.1.3 Design Cycles

ASME Class 1 components are designed to withstand the effects of cyclic loads due to temperature and pressure changes in the reactor system. These cyclic loads are introduced by normal unit load transients, reactor trips, startup and shutdown operations, and earthquakes. The 14 original design transients for the Reactor Coolant System (RCS) are found in USAR Table 5.1-8. Over the life of the plant, additional transients have been identified, including analyzed transients for new components and non-RCS components. The design cycles that are significant contributors to fatigue usage are included in the Fatigue Monitoring Program.

A.2.3.1.4 Reactor Coolant Piping

The reactor coolant piping connects the major components of the Reactor Coolant System, including the reactor vessel, the steam generators and the reactor coolant pumps. The reactor coolant piping has welded connections for pressure taps, temperature elements, vents, drains, decay heat removal, and emergency core cooling high-pressure injection water.

A thermal sleeve is provided in the high-pressure injection connection to the reactor coolant inlet piping. The analysis of the high-pressure injection nozzles determined that high-pressure injection flow tests had negligible effect on the high-pressure injection nozzles, but a significant effect on the normal makeup nozzle. The cumulative usage factor (CUF) for the normal makeup nozzle was calculated to be 0.558 after 40 flow tests; 0.513 usage due to the 40 flow tests and 0.045 usage due to all other transients. Projections of cycles for 60 years implies that the 40 design cycles will be reached in year 51, with 48 cycles occurring by year 60. Projecting the CUF to a 60-year number with 50 tests, gives a CUF of 0.686 (0.045 + 50/40 * 0.513), which implies the nozzles will still be acceptable. However, Davis-Besse monitors these cycles and will ensure action is taken before the analyzed number of cycles is reached. Because these nozzles may be reanalyzed for other reasons such as the planned modification to replace the nozzle safe ends and thermal sleeves, Davis-Besse will manage fatigue of these nozzles for the period of extended operation rather than reanalyze for the possible additional cycles at this time. Davis-Besse has committed (Table A-1, item 23) to replace the nozzle safe ends and thermal sleeves prior to reaching the period of extended operation.

The effects of fatigue on the reactor coolant piping will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.1.5 Steam Generator Remote Welded Plugs

Each remote welded plug installed in the once-through steam generators is limited to 33 cycles of heatup and cooldown. The 60-year cycle projection for some of these plugs exceeds the analyzed number of cycles. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

The effects of fatigue on the steam generator remote welded plugs will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2 Class I Fatigue Evaluations

The Fatigue Monitoring Program monitors the number of plant transient cycles to ensure that action is taken before the number of design cycles is exceeded. As such, the effects of aging due to fatigue are managed for the period of extended operation for the Class 1 piping and components. The effects of fatigue on the high energy line break analyses are also managed by the Fatigue Monitoring Program.

Specific evaluations for Class 1 components are discussed below.

A.2.3.2.1 Reactor Vessel Internals Bolts

Although the reactor vessel internals are designed to meet the stress requirements of ASME Section III, they are not code components. Consequently, a fatigue analysis of the reactor vessel internals was not required and was not performed as part of the original design.

FENOC has replaced the majority of the stainless steel, Alloy A-286, bolts for the reactor vessel internals with Alloy X-750 HTH bolts at Davis-Besse. The replacement bolts were designed to ASME Section III, and are provided with fatigue analyses. FENOC has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts at Davis-Besse. Design cumulative usage factors for the reactor vessel internals bolts are based on design cycles.

The effects of fatigue on the reactor vessel internals bolts will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2.2 Reactor Vessel Internals Flow Induced Vibration

The reactor vessel internals were analyzed for flow induced vibration. The classic endurance limit approach to design of components subject to flow-induced vibration was used, except for the incore instrumentation nozzles and the re-designed surveillance capsule holder tubes. The classic endurance limit approach is based on the observation that a fatigue curve becomes approximately asymptotic to a given value of stress (the endurance limit) for large numbers of cycles. A component can be designed for infinite life by maintaining the actual peak stresses below the endurance limit.

For the Davis-Besse reactor vessel internals, the ASME Code fatigue curve was extended to 1E+12 cycles (the upper bound on the number of cycles for a 40-year design life). The resulting stress value of 20,400 psi was reduced to 18,000 psi as the endurance limit. For 60-years of operation, it follows that 1.5E+12 would bound the expected loading cycles. The extrapolated fatigue curve at 1.5E+12 cycles is approximately 20,200 psi, still above the 18,000 psi that was used as the endurance limit. As such, the 18,000 psi endurance limit used for the flow induced vibration analyses of the reactor vessel internals remains valid for the period of extended operation. Therefore, the endurance limit for flow induced vibration of the reactor vessel internals remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The effects of fatigue due to flow induced vibration were analyzed for the incore instrument nozzles and re-designed surveillance capsule holder tubes for 40 years of operation. The resulting cumulative usage factors have been projected to remain below the limit of 1.0 for 60 years of operation.

The flow induced vibration analyses of the incore instrument nozzles and re-designed surveillance capsule holder tubes have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.3.2.3 Control Rod Drive Housings

The control rod drive housings are designed to ASME Section III and are analyzed for fatigue. The fatigue analyses for the control rod drive housings are based on the design transients, and the resulting cumulative usage factors are all less than 1.0.

The effects of fatigue on the control rod drive housings will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2.4 Reactor Coolant Pump Casings Fatigue

The reactor coolant pump casings are designed to ASME Section III and are analyzed for fatigue. The fatigue analyses for the reactor coolant pump casings are based on design transients, and the resulting cumulative usage factors are all less than 1.0.

The effects of fatigue on the reactor coolant pump casings will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2.5 Pressurizer Fatigue

The pressurizer is designed to ASME Section III and is analyzed for fatigue. Design cumulative usage factors for the limiting pressurizer locations, including the surge nozzle, were analyzed based on design transients and are all less than 1.0.

The effects of fatigue on the pressurizer will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2.6 Steam Generator Tube Sleeves Fatigue

USAR Section 5.5.2.3 indicates that steam generator tubes that are found to be leaking may be plugged or repaired by mechanical (rolled) sleeving.

Section III of the ASME Code does not provide design rules for mechanically roll expanded attachments, and theoretical stress analyses are inadequate. In such cases, Appendix II of ASME Section III permits the use of experimental stress analysis to substantiate the critical or governing stress. The structural adequacy of the sleeve attachment to withstand cyclic loadings was demonstrated by a fatigue test with the sleeve loading transients based on the design transients. The pressure cycling portion of the fatigue test for the steam generator tube sleeves is based on 360 startup cycles to bound all Babcock & Wilcox 177 fuel assembly plants. Davis-Besse has only

240 startup cycles allowed in USAR Table 5.1-8, and only 128 startup cycles projected for 60 years of operation.

The fatigue testing of the once-through steam generator tube sleeves remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.3.2.7 Auxiliary Feedwater Header Modification

The original auxiliary feedwater (AFW) headers internal to the steam generators were found damaged during the 1982 refueling outage. The repair was to install an external header on each steam generator, including some rerouting of piping and supports. Since this 1982 modification, the new design has been included in the steam generator stress report and included in the steam generator fatigue analyses.

The auxiliary feedwater thermal sleeve stresses were also analyzed according to the ASME Code for Class I components. The analysis provided a basis for demonstrating that the AFW thermal sleeve is capable of withstanding 300 cycles of auxiliary feedwater injection transients.

In addition, the riser flange attachment to the steam generator shell was analyzed per ASME Code requirements. However, it was necessary to limit the design life to 875 cycles of auxiliary feedwater initiation.

Flow induced vibration of the steam generator tubes with the new feedwater header design was reviewed. It was concluded that the stress and deflection with the external headers was significantly less than the stress and deflection with the original internal headers; consequently flow induced vibration was not reanalyzed for this modification. Section A.2.3.2.8 discusses the flow induced vibration analyses of the steam generator tubes.

Auxiliary feedwater initiations are projected to a maximum of 442 cycles through the period of extended operation. This projection exceeds the 300 cycles analyzed for the thermal sleeve but is less than the 875 cycles analyzed for the riser flange. The number of occurrences of design transients is tracked by the Fatigue Monitoring Program to ensure that action is taken before the design cycles are reached. As such, the effects of aging due to fatigue are managed for the period of extended operation.

The effects of fatigue on the auxiliary feedwater header modification will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2.8 Steam Generator Tubes and Tube Stabilizers Flow Induced Vibration

Flow induced vibration of the steam generator tubes has been analyzed for 40 years of operation. The analysis for an un-repaired tube has been projected to remain below 1.0 for 60 years of operation in accordance with 10 CFR 54.21(c)(1)(ii).

The CUF for the 3/8 inch tube stabilizers is calculated using both high cycle (flow induced vibration) and low cycle (transients) fatigue. The CUFs for the tube stabilizers have been projected to remain below 1.0 for 60 years of operation.

The analyses associated with the effects of flow induced vibration on the steam generator tubes and tube stabilizers have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.3.3 Non-Class 1 Fatigue Evaluations

The specific codes and standards to which systems and components important to safety were designed are listed in USAR Table 3.2-2.

The non-Class 1 mechanical components susceptible to fatigue fit into the two major categories:

1. Piping and in-line components (tubing, piping, thermowells, valve bodies, etc.)

   Non-class 1 components that are quality group B or C are largely designed and constructed to the ASME Code, but certain components are built to other codes including ANSI B31.1. The design of ASME Section III Code Class 2 and 3 piping systems incorporates a stress range reduction factor for determining acceptability of piping design with respect to thermal stresses. Piping systems designed to ANSI B31.1 also incorporate stress range reduction factors based upon the number of thermal cycles. In general, a stress range reduction factor of 1.0 in the stress analyses applies for up to 7,000 thermal cycles. The allowable stress range is reduced by the stress range reduction factor if the number of thermal cycles exceeds 7,000. If fewer than 7,000 cycles are expected through the period of extended operation, then the fatigue analysis (stress range reduction factor) of record will remain valid through the period of extended operation.

2. Non-piping components (Major Components)

   Fatigue need not be addressed for non-Class 1 vessels, heat exchangers, storage tanks, and pumps, unless these components were designed to ASME Section VIII Division 2 or ASME Section III, Subsection NC-3200.

Each of these categories is addressed below.

A.2.3.3.1 Non-Class 1 Piping and In-Line Components

Thermal cycles have been projected through 60 years of plant operation. These projections, applied to the non-Class 1 piping and in-line components, indicate that 7,000 thermal cycles will not be exceeded during 60 years of operation.

The analyses associated with fatigue of non-Class 1 piping and in-line components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.3.3.2 Non-Class 1 Major Components

For those non-Class 1 non-piping components identified as possibly subject to fatigue, a review of component design codes was conducted to determine if fatigue analyses of the components were required. If no fatigue analysis was required, then no TLAA for fatigue exists.

While most Class 1 components are designed in accordance with ASME Section III, non-Class 1 pressure vessels, heat exchangers, tanks, and pumps are often designed in accordance with other industry codes and standards, reactor designer specifications, and architect engineer specifications. ASME Section III, Subsection NC-3200 and ASME Section VIII, Division 2 include fatigue design requirements, and include provisions for "exemption from fatigue," which is actually a simplified fatigue evaluation based on materials, configuration, temperature, and cycles. If cyclic loading and fatigue usage for a component could be significant, then ASME Section III, Subsection NC-3200 or ASME Section VIII, Division 2 are specified.

Due to conservatism in ASME Section III, Subsections NC-3100 and ND-3000 and ASME Section VIII, Division 1, detailed fatigue analyses are not required. Also, fatigue analyses are not required for ASME Section III, Subsection NC and ND pumps and storage tanks (less than 15 psig), or for other design codes (e.g., ASME Section VIII, Division 1, American Water Works Association, Manufacturer's Standardization Society, National Electrical Manufacturers Association).

There are no fatigue analyses, and therefore no TLAA, associated with the non-Class 1 non-piping components.

A.2.3.4 Generic Industry Issues on Fatigue

This section addresses the Davis-Besse fatigue TLAAs associated with NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," and with the effects of the primary coolant environment on fatigue life.

A.2.3.4.1 Pressurizer Surge Line Thermal Stratification

NRC Bulletin 88-11 required the re-evaluation of the cyclic fatigue of the pressurizer surge line. As part of the re-evaluation, the Davis-Besse plant heatup and cooldown transients were redefined. Other transients were modified to include thermal stratification and striping. The surge line piping and nozzles were analyzed for license renewal, considering the effects of the reactor coolant environment. See Section A.2.3.4.2 for a discussion of the effects of the reactor coolant environment on fatigue.

A.2.3.4.2 Effects of the Reactor Coolant Environment on Fatigue

Industry test data indicates that certain environmental effects (such as temperature and dissolved oxygen content) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. The ASME design fatigue curves were based on laboratory tests in air and at low temperatures. Although the failure curves derived from laboratory tests were adjusted to account for effects such as data scatter, size effect, and surface finish, these adjustments may not be sufficient to account for actual plant operating environments.

No immediate NRC staff or licensee action is necessary to deal with the environmentally assisted fatigue issue. However, because metal fatigue effects increase with service life, environmentally assisted fatigue is evaluated for license renewal.

NUREG/CR-6260 [Reference A.2-5] identifies locations of interest for consideration of environmental effects in several types of nuclear plants. Section 5.3 of NUREG/CR-6260 reviewed the following locations for Babcock & Wilcox pressurized water reactors.

1. Reactor vessel shell and lower head; including the instrumentation nozzles
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including pressurizer surge nozzle and hot leg surge nozzle)
4. High pressure injection/makeup nozzle
5. Reactor vessel core flood nozzle
6. Decay heat removal Class 1 piping

Evaluations performed for the period of extended operation indicate that 40-year cumulative usage factors will not exceed 1.0; however an environmentally assisted fatigue adjustment is not applied for the initial 40 years of operation, consistent with the closure of Generic Safety Issue (GSI) 190, “Fatigue Evaluation of Metal Components for 60-year Plant Life.”

The effect of the reactor coolant environment on fatigue usage has been evaluated for the six locations identified in NUREG/CR-6260. The results for those six locations show that most locations have an environmentally assisted fatigue adjusted cumulative usage factor of less than 1.0. However, high pressure injection/makeup (HPI/MU) nozzle stainless steel safe end and associated Alloy 82/182 weld have environmentally adjusted CUFs greater than 1.0. FENOC has committed (see Table A-1, Item 23) to replace the HPI/MU nozzle safe end and associated Alloy 82/182 weld prior to entering the period of extended operation.

The effects of environmentally assisted fatigue for each NUREG/CR-6260 location will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.4 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The Environmental Qualification (EQ) of Electrical Components 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, components in the EQ program that are not qualified for the full current license term (40 years) are required to be refurbished, replaced, or have their qualification extended prior to reaching the limits established in the evaluation. The EQ program ensures that the environmentally qualified components are maintained in accordance with their qualification bases. Equipment qualification evaluations for components in the EQ program that specify a qualification of at least 40 years are TLAAs for license renewal.

Environmental qualification of electrical equipment will be managed by the Environmental Qualification (EQ) of Electrical Components Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.5 CONTAINMENT FATIGUE ANALYSES

Additional potential TLAAs associated with the Containment structure were reviewed and are summarized in the following sections:

- Containment Vessel (Section A.2.5.1)
- Containment Penetrations (Section A.2.5.2)
- Permanent Canal Seal Plate (Section A.2.5.3)

A.2.5.1 Containment Vessel

The containment vessel is a Class B vessel as defined in the ASME Section III, Paragraph N-132, 1968 Edition through Summer Addenda 1969. The containment vessel meets the requirements for Paragraph N-415.1 of ASME Section III, thereby justifying the exclusion of cyclic or fatigue analyses in the design of the containment vessel, as discussed in USAR Section 3.8.2.1.5. The containment vessel has been analyzed for 400 pressure cycles (from -25 psi to 120 psi) and 400 temperature cycles (from 30°F to 120°F). The containment vessel has not seen any pressure cycles in the defined range (through 2009). The values of 400 pressure cycles and 400 temperature cycles used to exclude fatigue analyses will not be exceeded for 60 years of operation.

The TLAA associated with exclusion of the containment vessel from fatigue analyses per ASME Section III, Paragraph N-415.1 remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.5.2 Containment Penetrations

There are no fatigue analyses, and hence no TLAA, associated with the containment vessel penetration assemblies.

A.2.5.3 Permanent Canal Seal Plate

The permanent canal seal plate (also known as permanent reactor cavity seal plate) spans the gap between the reactor vessel and the fuel transfer canal floor, and retains water in the canal when the canal is flooded. The permanent canal seal plate is made up of a support structure that rests on the shield plate and reactor vessel seal ledge and a seal membrane that covers the support structure and is welded to the shield plate and reactor vessel seal ledge.

The fatigue analysis of the permanent canal seal plate seal membrane installed in 2004 shows that the maximum fatigue cumulative usage factor location is the inner leg to the reactor vessel seal ledge weld. A limit of 50 zero-to-full power cycles is recommended to meet the ASME Code requirement of maintaining the cumulative usage factor less than 1.0. The permanent canal seal plate is projected to experience 51 heatup and cooldown cycles from the date of installation (2004) through the end of the period of extended operation. However, the number of occurrences of permanent canal seal plate heatup and cooldown is tracked by the Fatigue Monitoring Program to assure that action is taken before the analyzed number of transients is reached.

The effects of fatigue of the permanent canal seal plate will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.6 INSERVICE INSPECTION – FRACTURE MECHANICS ANALYSES

10 CFR 50.55a(g) requires an Inservice Inspection program to verify the integrity of the reactor coolant pressure boundary. Flaws detected during examination are compared to acceptance standards established in ASME Section XI. Unacceptable flaw require detailed analyses, repair, or replacement.

Acceptance via fracture mechanics analysis requires a prediction of flaw growth considering a chosen evaluation period, i.e., no shorter than the time until the next inspection following discovery of the flaw or as long as the remaining service life of the plant. Flaw indications that are determined not to grow beyond acceptance limits during the evaluation period are justified for continued operation. Fracture mechanics analyses performed for the life of the plant are TLAAs that typically involve the same design transient cycle assumptions considered in the current licensing basis.

A.2.6.1 Reactor Coolant System Loop 1 Cold Leg Drain Line Weld Overlay Repair

A full structural overlay repair was performed for an axial indication found on the Reactor Coolant System Loop 1 cold leg drain line during the Cycle 14 refueling outage. The structural weld overlay of the cold leg drain nozzle was designed consistent with the requirements of ASME Section XI; Code Case N-504-2; Non-mandatory Appendix Q; and was supplemented by additional design considerations specific to the cold leg drain nozzle-to-elbow weld.

The overlay is designed as a full structural overlay that assumes the as-found flaw propagates to a 100% through-wall 360-degree crack rather than performing a crack growth analysis of the as-found flaw. Thus there is no time dependency in the weld overlay design.

The fatigue analysis estimated cycles for 60 years based on the original design cycles. Because this analysis is based on a specific number of cycles, it is considered a TLAA. All cumulative usage factors for the reactor coolant pump drain line weld overlay are less than 1.0.

The effects of fatigue on the reactor coolant pump drain line weld overlay repair will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.6.2 OTSG 1-2 Flaw Evaluations

During the Cycle 5 refueling outage (year 1988) a number of flaw indications were detected in steam generator 1-2, both in the shell near the steam outlet nozzle and in the shell welds near the lower tubesheet-to-shell juncture. Two of the indications in the shell near the steam outlet nozzle were evaluated according to ASME Section XI, with the remaining shell indications bounded by those evaluated. Five of the indications in the shell welds near the lower tubesheet-to-shell juncture were evaluated, with the remaining shell weld indications bounded by those evaluated.

Simplified evaluation of fatigue crack growth, based on 240 heatup and cooldown cycles concluded that there would be only slight crack growth, and the indications were found to be acceptable by ASME Section XI, IWB-3612 standards. Because these analyses are based on a specific number of cycles, they are TLAAs.

The effects of fatigue on the steam generator flaw evaluations will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

The TLAAs that do not fit into any of the previous major categories are evaluated below.

A.2.7.1 Leak-Before-Break

The leak-before-break (LBB) concept relies on the plant's ability to detect leakage from a through-wall flaw and then take appropriate action before that flaw grows to the point of pipe failure. Analyses showed that postulated flaws producing detectable leakage exhibit stable growth, and thus, allow a controlled plant shutdown before any potential exists for catastrophic piping failure.

The LBB analyses were updated to include the Alloy 52 weld overlays that were installed on the reactor coolant pump suction and discharge nozzles for PWSCC mitigation. These analyses considered fatigue flaw growth, and PWSCC. Because these analysis considerations could be influenced by time, LBB analyses are considered to be TLAAs. Fatigue flaw growth, thermal aging, and PWSCC are addressed separately below.

Fatigue Flaw Growth

Surface flaws were postulated at the piping system locations with the highest stress coincident with the lower bound of material properties for the base metal and welds. The leak-before break analysis for the reactor coolant pump suction and discharge weld overlays is based on 1.5 times the design cycles.

The effects of fatigue flaw growth on piping approved for LBB will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

Thermal Aging

The only stainless steel materials addressed in the LBB analysis are the safe ends welded to the reactor coolant pump casings and the casings themselves; with the pump casings being the only cast stainless steel.

The updated LBB analysis was based on saturated embrittlement of the cast austenitic stainless steel (CASS) casings such that there is no embrittlement TLAA.

Aging management review of the RCS determined reduction of fracture toughness due to thermal embrittlement of CASS components to be an aging effect requiring management for the reactor coolant pump casings. The acceptability of a 10-year inspection interval for these weld overlays was demonstrated in the updated LBB. This analysis does not justify operation of the weld overlays for the life of the plant, but for the 10 years between inspections. Therefore, the effects of thermal aging on CASS components in the approved LBB piping will be managed by the Inservice Inspection Program for the period of extended operation.

The effects of thermal aging on CASS components in the approved LBB piping are not a TLAA.

PWSCC

FENOC received relief to install weld overlays on certain Alloy 600 components and Alloy 82/182 dissimilar metal welds for mitigation of PWSCC, including Alloy 82/182 welds in piping approved for LBB. FENOC updated the original leak-before-break calculations for Davis-Besse with an evaluation demonstrating that the weld overlays resolve the concerns for original welds susceptibility to primary water stress corrosion cracking. Critical crack sizes and leakage rates with the weld overlay in place were evaluated to demonstrate that margins exist for detection of leakage, i.e., the conclusions of the existing leak-before-break analysis remain valid.

Aging management review of the RCS, including the nickel alloy weld locations, identified cracking due to PWSCC as an aging effect requiring management for the period of extended operation. Cracking due to PWSCC is managed by the Inservice Inspection Program and the Nickel-Alloy Management Program.

The analyses associated with the effects of PWSCC of Alloy 600/82/182 materials on the LBB analysis are not a TLAA.

A.2.7.2 Metal Corrosion Allowance for Pressurizer Instrument Nozzles

USAR Section 5.2.3.2 indicates that pressurizer nozzle repairs and replacements have resulted in a portion of the carbon steel pressurizer nozzle bore being exposed to reactor coolant. This resulted in an increase of the general corrosion rate of the pressurizer shell base metal in the nozzle bores from zero to 1.42 thousandths of an inch (mils) per year. Over the 9 years from the installation of this modification to the end of the original licensed period, this will result in a loss of 13 mils of the pressurizer carbon steel shell in the nozzle annular regions. The allowable radial corrosion limit, calculated per ASME Section III, is 293 mils for the level instrument nozzles, 493 mils for the sample nozzle and 495 mils for the vent and thermowell nozzles. This corrosion analysis is a TLAA.

Loss of material in the annular region of the repaired pressurizer nozzles has been projected through the end of the period of extended operation and remains below the allowable radial corrosion limit, to meet ASME Section III, Class 1 Code design for the nozzles.

The corrosion allowance TLAA for the pressurizer nozzle annular regions has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(ii).

A.2.7.3 Reactor Vessel Thermal Shock due to Borated Water Storage Tank Water Injection

USAR Section 5.2 addresses integrity of the reactor coolant pressure boundary and the analysis to demonstrate that the reactor vessel can safely accommodate the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at the end of the vessel's design life. The analysis documents the reactor vessel integrity during a small steam line break, which creates a pressurized thermal shock condition. This transient generates the greatest level of stress in the reactor vessel. Technical Specifications allow the borated water storage tank (BWST) water temperature to be as low as 35°F. The analysis was revised for license renewal to use reactor vessel embrittlement values that bound the period of extended operation.

The revised fracture mechanics analysis evaluated the integrity of the reactor vessel against pressurized thermal shock (PTS) for 52 EFPY considering the 35° F minimum temperature for the BWST. Several locations in the reactor vessel were analyzed for PTS, and all locations have demonstrated service life greater than 52.0 EFPY. Flaws do not initiate for any of the postulated flaw depths. The minimum critical margin to applied pressure margin is 2.21 at the nozzle belt forging.

The reactor vessel integrity analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.7.4 High Pressure Injection / Makeup Nozzle Thermal Sleeves

During the Cycle 5 refueling outage, Davis-Besse discovered a failed thermal sleeve for HPI/MU nozzle A-1. Corrective actions included assessment and preservation of the structural integrity of the nozzle, which had experienced thermal cycling due to the thermal sleeve failure. The makeup flow path was re-routed from nozzle A-1 to nozzle A-2 during the Cycle 6 refueling outage (1990) as one of the corrective actions. Fracture mechanics analysis of thermal sleeve life under various makeup flow cycling conditions predicted a thermal sleeve lifetime exceeding 20 eighteen-month operating cycles under current makeup flow control conditions.

Since that analysis, Davis-Besse had an extended (approximately two year) Cycle 13 refueling outage, converted to a 24-month fuel cycle, and performed a measurement uncertainty recapture power uprate. The corresponding predicted end-of-life for the HPI/MU nozzle thermal sleeve is approximately 2022, based on the predicted number of makeup thermal cycles. The current operating license for Davis-Besse will expire in April of 2017. However, FENOC has committed (see Table A-1, Item 23) to replace the

HPI/MU nozzle safe end and associated Alloy 82/182 weld prior to entering the period of extended operation.

The TLAA associated with cracking of the high pressure injection / makeup nozzle thermal sleeves will be managed by the Fatigue Monitoring Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.8 APPENDIX A.2 REFERENCES

A.2-1 Allen, Barry S. (Davis-Besse), Letter to NRC, L-09-072, "License Amendment Request to Incorporate the Use of Alternate Methodologies for the Development of Reactor Pressure Vessel Pressure-Temperature Limit Curves, and Request for Exemption from Certain Requirements Contained in 10 CFR 50.61 and 10 CFR 50, Appendix G," April 15, 2009.

A.2-2 NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," May 1994.

A.2-3 Wengert, Thomas J. (NRC), Letter to Barry S. Allen (Davis-Besse), R-08-153, "Davis-Besse Nuclear Power Station, Unit 1-Request for Additional Information Related to Improved Technical Specifications Conversion (MD6398)," June 18, 2008.

A.2-4 Allen, Barry S. (Davis-Besse), Letter to NRC, L-08-224, "Response to Request for Additional Information Regarding License Amendment Request: Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS) and Copy of Two Questions from the U.S. Regulatory Commission and Davis-Besse Nuclear Power Station Improved Technical Specifications Conversion Website (TAC No. MD6398)," September 3, 2008

A.2-5 NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," February 1995.

A.2-6 Collins, Daniel S. (NRC), Letter to Bezilla, Mark B. (Davis-Besse), Log 6459, "Davis-Besse Nuclear Power Station, Unit 1- Evaluation of Request for Relief RE: Full Structural Weld Overlay (TAC No. MD0683)," October 19, 2006

A.2-7 Wengert, Thomas J. (NRC), Letter to Barry S. Allen (Davis-Besse), R-08-162, "Davis-Besse Nuclear Power Station, Unit No.1 – Issuance of Amendment RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MD8326), June 30, 2008

A.2-8 Bezilla, Mark B. (Davis-Besse), Letter to NRC, L-08-034, "Summary of Design and Analyses of the Weld Overlays for Pressurizer and Hot Leg Nozzle Large Bore Dissimilar Metal Welds for Alloy 600 Mitigation (TAC No. MD4452)," February 8, 2008

A.2-9 NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2

A.2-10 DOR Guidelines, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979

A.2-11 NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1

A.2-12 NRC Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants," Revision 1

A.2-13 NRC Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3

A.2-14 AREVA NP Document BAW-2241P-A, "Fluence and Uncertainty Methodologies," April 1999 (NRC Safety Evaluation Report included)

A.2-15 AREVA NP Document BAW-10013-A, "Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones under Austenitic Stainless Steel Weld Cladding," Last Revised February 15, 1972

A.2-16 AREVA NP Document BAW-10046A, "Method of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," Revision 4

A.2-17 NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Vessel Neutron Fluence," March 2001

A.3 LICENSE RENEWAL COMMITMENT LIST

Table A-1 identifies those actions committed to by FENOC for Davis-Besse in the Davis-Besse LRA. These regulatory commitments will be tracked within the FENOC regulatory commitment management program. Any other actions discussed in the LRA represent intended or planned actions by FENOC. These other actions are described only as information and are not regulatory commitments. This list will be revised as necessary in subsequent amendments to reflect changes resulting from NRC audit questions and Davis-Besse responses to NRC requests for additional information.

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Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
1	Enhance the Aboveground Steel Tanks Inspection Program to: • Include a volumetric examination of the tank bottoms to detect evidence of loss of material due to crevice, general, or pitting corrosion, or to confirm a lack thereof. Establish the examination technique, the inspection locations, and the acceptance criteria for the examination of the tank bottoms. Require that unacceptable inspection results be entered into the FENOC Corrective Action Program.	April 22, 2017	LRA	A.1.2 B.2.2
2	Implement the Boral® Monitoring Program as described in LRA Section B.2.5.	April 22, 2017	LRA	A.1.5 B.2.5
3	Enhance the Buried Piping and Tanks Inspection Program to: • Add 1) bolting for buried Fire Protection System piping and 2) the emergency diesel fuel oil storage tanks (DB-T153-1, DB-T153-2) to the scope of the program. [continued]	April 22, 2017	LRA	A.1.7 B.2.7

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
3 [continued]	• Require that an inspection of coated and wrapped buried piping or tank be performed within the 10-year period prior to entering the period of extended operation (i.e., between year 30 and year 40). Specify that if an opportunistic inspection has not occurred between year 30 and year 38, then an excavation of a section of coated and wrapped buried piping for the purpose of inspection will be performed before year 40. • Require that an additional inspection of coated and wrapped buried piping or tank be performed within 10 years after entering the period of extended operation (i.e., between year 40 and year 50). Specify that if an opportunistic inspection has not occurred between year 40 and year 48, then an excavation of a section of coated and wrapped buried piping for the purpose of inspection will be performed before year 50. • Require that an inspection of uncoated cast iron buried piping be performed within the 10-year period prior to entering the period of extended operation (i.e., between year 30 and year 40). Specify that if an opportunistic inspection has not occurred between year 30 and year 38, then an excavation of a section of uncoated cast iron buried piping for the purpose of inspection will be performed before year 40. [continued]	April 22, 2017	LRA	A.1.7 B.2.7

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
3 [continued]	• Require that an additional inspection of uncoated cast iron buried piping be performed within 10 years after entering the period of extended operation (i.e., between year 40 and year 50). Specify that if an opportunistic inspection has not occurred between year 40 and year 48, then an excavation of a section of uncoated cast iron buried piping for the purpose of inspection will be performed before year 50. • Require that an inspection of buried Fire Protection System bolting will be performed when the bolting becomes accessible during opportunistic or focused inspections. • Require that the inspections of buried piping be conducted using visual (VT-3 or equivalent) inspection methods. Excavation shall be of approximately 10 linear feet of piping, with all surfaces of the pipe exposed.	April 22, 2017	LRA	A.1.7 B.2.7
4	Implement the Collection, Drainage, and Treatment Components Inspection Program as described in LRA Section B.2.9.	April 22, 2017	LRA	A.1.9 B.2.9
5	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection as described in LRA Section B.2.11.	April 22, 2017	LRA	A.1.11 B.2.11
6	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.12.	April 22, 2017	LRA	A.1.12 B.2.12

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
7	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program as described in LRA Section B.2.13.	April 22, 2017	LRA	A.1.13 B.2.13
8	Enhance the External Surfaces Monitoring Program to: • Add systems which credit the program for license renewal but do not have Maintenance Rule intended functions to the scope of the program. • Perform opportunistic inspections of surfaces that are inaccessible or not readily visible during normal plant operations or refueling outages, such as surfaces that are insulated. • Perform, in conjunction with the One-Time Inspection, inspection and surveillance of elastomers and polymers exposed to air-indoor uncontrolled or air-outdoor environments, but not replaced on a set frequency or interval (i.e., are long-lived), for evidence of cracking and change in material properties (hardening and loss of strength). Specify acceptance criteria of no unacceptable visual indications of cracks or discoloration that would lead to loss of function prior to the next inspection. [continued]	April 22, 2017	LRA	A.1.15 B.2.15

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
8 [continued]	• Perform inspection of the control room emergency ventilation system air-cooled condensing unit cooling coil tubes and fins and the station blackout diesel generator radiator tubes and fins for visible evidence of external surface conditions that could result in a reduction in heat transfer. Specify acceptance criteria of no unacceptable visual indications of fouling (build up of dirt or other foreign material) that would lead to loss of function prior to the next scheduled inspection.	April 22, 2017	LRA	A.1.15 B.2.15
9	Enhance the Fatigue Monitoring Program to: • For locations, including NUREG/CR-6260 locations, projected to exceed a cumulative usage factor (CUF) of 1.0, the program will implement one or more of the following: (1) Refine the fatigue analyses to determine valid CUFs less than 1.0 using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case), (2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC), (3) Repair or replacement of the affected locations. • Monitor any transient where the 60-year projected cycles were used in an environmentally-assisted fatigue evaluation and establish an administrative limit that is equal to or less than the 60-year projected cycles.	April 22, 2017	LRA	A.1.16 B.2.16

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
10	Enhance the Fire Water Program to: • Perform periodic ultrasonic testing for wall thickness of representative above-ground water suppression piping that is not periodically flow tested but contains, or has contained, stagnant water. The ultrasonic testing will be performed prior to the period of extended operation and at appropriate intervals thereafter, based on engineering evaluation of the initial results. • Perform at least one opportunistic or focused visual inspection of the internal surface of buried fire water piping and of similar above-ground fire water piping, within the five-year period prior to the period of extended operation, to confirm whether conditions on the internal surface of above-ground fire water piping can be extrapolated to be indicative of conditions on the internal surface of buried fire water piping. • Perform representative sprinkler head sampling (laboratory field service testing) or replacement prior to 50 years in-service (installed), and at 10-year intervals thereafter, in accordance with NFPA 25, or until there are no untested sprinkler heads that will see 50 years of service through the end of the period of extended operation. [continued]	April 22, 2017	LRA	A.1.18 B.2.18

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
10 [continued]	• Perform opportunistic fire water supply and water-based suppression system internal inspections each time a fire water supply or water-based suppression system (including fire pumps) is breached for repair or maintenance. These internal visual inspections must be demonstrated to be: 1) representative of water supply and water-based suppression locations, 2) performed on a reasonable basis (frequency), and 3) capable of evaluating wall thickness and flow capability. If the internal inspections cannot be completed of a representative sample, then ultrasonic testing inspections will be used to complete the representative sample.	April 22, 2017	LRA	A.1.18 B.2.18
11	Implement the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.21.	April 22, 2017	LRA	A.1.21 B.2.21

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
12	Enhance the Masonry Wall Inspection to: • Include and list the structures within the scope of license renewal that credit the program for aging management. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Specify that for each masonry wall, the extent of observed masonry cracking or degradation of steel edge supports or bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking or steel degradation is sufficient to invalidate the evaluation basis. An option is to develop a new evaluation basis that accounts for the degraded condition of the wall (i.e., acceptance by further evaluation).	April 22, 2017	LRA	A.1.27 B.2.27

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
13	Implement the One-Time Inspection as described in LRA Section B.2.30. Enhance the One-Time Inspection to: • Include visual inspections to detect and characterize the material condition of aluminum, copper alloy (including copper alloy greater than 15% zinc), stainless steel, and steel (including gray cast iron) components exposed to condensation or diesel exhaust to provide direct evidence as to whether, and to what extent, cracking, loss of material, or reduction in heat transfer has occurred. • Include visual and physical examination, such as manipulation and prodding, of elastomers (flexible connections) to supplement the External Surfaces Monitoring Program and provide direct evidence as to whether, and to what extent, hardening and loss of strength due to thermal exposure, ultraviolet exposure, and ionizing radiation of elastomers has occurred.	April 22, 2017	LRA	A.1.30 B.2.30
14	Implement the PWR Reactor Vessel Internals Program as described in LRA Section B.2.32.	April 22, 2017	LRA	A.1.32 B.2.32
15	The PWR Reactor Vessel Internals Program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A.	Following NRC approval of MRP-227 and re-issuance of the guidelines as MRP-227-A.	LRA	A.1.32 B.2.32

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
16	Enhance the Reactor Head Closure Studs Program as follows: • Select an alternate stable lubricant that is compatible with the fastener material and the environment. A specific precaution against the use of compounds containing sulfur (sulfide), including molybdenum disulfide (MoS_2), as a lubricant for the reactor head closure stud assemblies will be included in the program.	April 22, 2017	LRA	A.1.34
17	Enhance the Reactor Vessel Surveillance Program as follows: • The Capsule Insertion and Withdrawal Schedule for Davis-Besse will be revised to schedule testing of the TE1-C capsule.	April 22, 2017	LRA	A.1.35 B.2.35
18	Implement the Selective Leaching Inspection as described in LRA Section B.2.36.	April 22, 2017	LRA	A.1.36 B.2.36
19	Implement the Small Bore Class 1 Piping Inspection as described in LRA Section B.2.37.	April 22, 2017.	LRA	A.1.37 B.2.37

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
20	Enhance the Structures Monitoring Program to: • Include and list the structures within the scope of license renewal that credit the program for aging management. • Include aging effect terminology (e.g., loss of material, cracking, change in material properties, and loss of form). • List ACI 349.3R-96 and ANSI/ASCE 11-90 as references and indicate that they provide guidance for the selection of parameters monitored or inspected. • Clarify that a "structural component" for inspection includes each of the component types identified within the scope of license renewal as requiring aging management. • Require the responsible engineer to review site raw water pH, chlorides, and sulfates test results prior to the inspection to take into account the raw water chemistry for any unusual trends during the period of extended operation. Raw water chemistry data shall be collected at least once every five years. Data collection dates shall be staggered from year to year (summer-winter-summer) to account for seasonal variation. [continued]	April 22, 2017	LRA	A.1.39 B.2.39

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
20 [continued]	• Include a special provision to monitor below-grade inaccessible concrete components before and during the period of extended operation. Perform a below-grade examination of concrete below elevation 570 (groundwater elevation) of an in-scope structure prior to the period of extended operation. The inspection will include concrete examination using acceptance criteria from NUREG-1801 XI.S6 Program element 6. The examination of concrete below elevation 570 feet may be conducted during maintenance activities. Any degradation found that exceeds the acceptance criteria will be trended and processed through the FENOC Corrective Action Program. • Specify that, upon notification that a below-grade structural wall or other in-scope concrete structural component will become accessible through excavation, a follow-up action is initiated to the responsible engineer to inspect the exposed surfaces for age-related degradation. Such inspections will include concrete examination using acceptance criteria from NUREG-1801 XI.S6 Program element 6. Any degradation found that exceeds the acceptance criteria will be trended and processed through the FENOC Corrective Action Program. [continued]	April 22, 2017	LRA	A.1.39 B.2.39

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
20 [continued]	• List ACI 349.3R-96, ANSI/ASCE 11-90, and EPRI Report 1007933 as references and indicate that they provide guidance for detection of aging effects. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • Indicate that ACI 349.3R-96 provides acceptable guidelines which will be considered in developing acceptance criteria for concrete structural elements, steel liners, joints, coatings, and waterproofing membranes.	April 22, 2017	LRA	A.1.39 B.2.39

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
21	Enhance the Water Control Structures Inspection to: • Include the Service Water Discharge Structure which is within the scope of license renewal. • Include parameters monitored and inspected for water control structures, including the Service Water Discharge Structure, in accordance with applicable inspection elements listed in Section C.2 of Regulatory Guide 1.127 Revision 1. Descriptions of concrete conditions will conform with the appendix to the American Concrete Institute (ACI) publication, ACI 201. The use of photographs for comparison of previous and present conditions will be included as a part of the inspection program. • Specify that water control structure periodic inspections are to be performed at least once every five years. • Add an action to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management. • List ACI 349.3R-96 as a reference and indicate that it will be considered in developing acceptance criteria for inspection of water control structures.	April 22, 2017	LRA	A.1.40 B.2.40
22	FENOC commits to enclose or otherwise protect the safety-related station ventilation radiation monitors located in the Turbine Building such that leakage and spray from surrounding piping systems does not adversely impact the intended function of the radiation monitors.	April 22, 2017	N/A	N/A

Table A-1
Davis-Besse License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
23	In association with the TLAA for cracking of the high pressure injection / makeup nozzle thermal sleeves, FENOC commits to replace all four high pressure injection / makeup nozzle thermal sleeves and safe ends prior to the period of extended operation. In addition, FENOC commits to evaluate the environmental effects on the replacement HPI nozzle safe ends and associated welds in accordance with NUREG/CR-6260 and the guidance of EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." Any nickel-based alloy locations will be evaluated in accordance with NUREG/CR-6909.	April 22, 2017	LRA	A.2.7.4
24	The elements of corrective actions, confirmation process, and administrative controls in the Quality Assurance Program Manual will be applied to the credited aging management programs and activities for the structures and components determined to require aging management for the period of extended operation.	April 22, 2017	LRA	A.1
25	FENOC commits to create a preventive maintenance task to periodically replace the letdown coolers (DB-E21-1 & 2) at a set frequency.	April 22, 2017	LRA	2.3.3.18

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APPENDIX B

AGING MANAGEMENT PROGRAMS

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APPENDIX B

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APPENDIX B

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B.0 AGING MANAGEMENT PROGRAMS

B.1 INTRODUCTION

B.1.1 OVERVIEW

License renewal aging management program (AMP) descriptions are provided in this appendix for each program credited for managing aging effects based upon the aging management review results provided in Sections 3.1 through 3.6.

Each aging management program described in this appendix is evaluated on the basis of 10 program elements in accordance with the guidance in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

B.1.2 METHOD OF DISCUSSION

For those existing AMPs that are comparable to the programs described in Sections X and XI of NUREG-1801, the "Generic Aging Lessons Learned (GALL) Report," the program evaluation is presented in the following summary format:

- **Aging Management Program Description** – An abstract of the overall program is provided.
- **NUREG-1801 Consistency** – A statement is made regarding consistency between the Davis-Besse program and the corresponding NUREG-1801 program.
- **Exceptions to NUREG-1801** – Exceptions to NUREG-1801 programs are identified when elements of the Davis-Besse program are different from the NUREG-1801 program elements or when elements of the NUREG-1801 program are not applicable to Davis-Besse. Each exception is listed along with the affected element. A justification is provided for each exception.
- **Enhancements** – Enhancements to existing programs necessary to ensure consistency with NUREG-1801 or to expand the scope of the program for license renewal are identified. Each enhancement is listed along with the affected program element and a proposed schedule for completion of the enhancement.
- **Operating Experience** – Discussion of operating experience information specific to the program is provided.
- **Conclusion** – A conclusion section provides a statement of reasonable assurance that the program is effective, or will be effective, once enhanced or developed.

For those programs that are either new or plant-specific, the above format is generally followed along with the additional provision of a discussion of each of the 10 elements associated with the program.

B.1.3 QUALITY ASSURANCE PROGRAM AND ADMINISTRATIVE CONTROLS

Three elements of an effective aging management program that are common to each of the aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the FirstEnergy Nuclear Operating Company (FENOC) Quality Assurance Program Manual (QAPM), which implements the requirements of 10 CFR 50 Appendix B. The QAPM is incorporated by reference in the Updated Safety Analysis Report (USAR) Section 17.

Prior to the period of extended operation, the elements of corrective actions, confirmation process, and administrative controls in the QAPM will be applied to required aging management programs for both safety-related and nonsafety-related structures and components determined to require aging management during the period of extended operation. The corrective actions, confirmation process, and administrative controls in the QAPM, to be applied to the credited aging management programs and activities for the structures and components determined to require aging management, are consistent with the related discussions in the Appendix on Quality Assurance for Aging Management Programs in NUREG-1801, Volume 2.

The elements of corrective actions, confirmation process, and administrative controls of the QAPM are described in the sections below, including a general comparison to the associated elements of the corresponding NUREG-1801 aging management programs.

Corrective Actions:

Corrective actions are implemented through the FENOC Corrective Action Program that satisfies the requirements of 10 CFR 50, Appendix B, Criterion XVI. Conditions adverse to quality, an all inclusive term used in reference to failures, malfunctions, deficiencies, defective items, and non-conformances are identified, reported to management, and corrected. In the case of significant conditions adverse to quality, measures are implemented to ensure that the root cause is determined and that corrective actions are taken to preclude recurrence.

The Corrective Action Program is the subject of periodic NRC examination and Davis-Besse self-assessment and audit. In general, problems are effectively identified, evaluated, and prioritized, and effective corrective actions implemented for conditions adverse to quality. Some program shortfalls have been identified, but corresponding process improvements have been developed and implemented. The current program is, therefore, adequate for aging management considerations.

Confirmation Process:

The focus of the confirmation process is on the follow-up actions taken to verify effective implementation of corrective actions and to preclude repetition of significant conditions adverse to quality. The Corrective Action Program includes the requirement that measures be taken to preclude repetition of significant conditions adverse to quality. These measures include actions to verify effective implementation of proposed corrective actions. The confirmation process is part of the Corrective Action Program and, for significant conditions adverse to quality, includes:

- reviews to assure proposed actions are adequate,
- tracking and reporting of open corrective actions,
- root cause, and
- reviews of corrective action effectiveness.

Effectiveness reviews are conducted as part of the Corrective Action Program to ensure that corrective actions have been completed and to identify any repetition of events. The Corrective Action Program is also monitored for potentially adverse trends. The existence of an adverse trend due to recurring or repetitive adverse conditions will result in the initiation of follow-up actions in the Corrective Action Program.

Administrative Controls:

Administrative controls that govern aging management activities are established within the document control procedures that implement: (1) industry standards related to administrative controls and quality assurance for the operational phase of nuclear power plants, and (2) the requirements of 10 CFR 50, Appendix B, Criterion VI.

Plant policies, directives, and procedures are written and controlled to specify and manage various activities, particularly those related to compliance with 10 CFR 50, Appendix B. The phrase "administrative control" refers to the adherence to the policies, directives, and procedures, and includes the formal review and approval process that the plant policies, directives, and procedures undergo as they are issued (and subsequently revised). The individual documents (i.e., the plant policies, directives, and procedures), in conjunction with the plant's quality assurance program documents, provide the overall administrative framework to ensure regulatory requirements are met.

B.1.4 OPERATING EXPERIENCE

The operating experience review demonstrates the effectiveness of the plant programs and activities that are credited with aging management for the period of extended operation. Industry and plant-specific operating experience for existing and new programs and for components to be managed by new Davis-Besse plant programs and activities was reviewed as an input to the aging management program evaluations. Industry operating experience was incorporated into the license renewal process through the use of license renewal guidance documents that incorporated operating experience regarding aging effects requiring management. Industry operating experience applicable to Davis-Besse since issuance of the industry guidance documents (2005) was reviewed and evaluated. The search of industry operating experience (OE) was performed through a search of NRC generic communications (Bulletins, Information Notices, Generic Letters, Regulatory Issue Summaries, etc.), and a search of industry operating experience from the Institute for Nuclear Power Operations (INPO) and from the World Association of Nuclear Operators (WANO) as contained in the FENOC Corrective Action Program.

Plant procedures require that the discovery of conditions adverse to quality be documented in accordance with the FirstEnergy Nuclear Operating Company Corrective Action Program. A review of plant records from January 2001 and later was performed to identify examples of age-related degradation related to current plant operation. The scope of the review included reports generated under the Corrective Action Program and licensee event reports. These records provide documentation of situations where systems, structures, and components exhibit adverse conditions, including conditions adverse to quality and age-related degradation. Keywords related to aging and degradation were used to search the records.

The industry and plant-specific operating experience review provides the basis for the determination that existing programs are either effective or require enhancement; that one-time inspections are appropriate to verify that either aging is not occurring or that aging is being effectively managed by an existing program; or that a new program is required to be established to manage the effects of aging.

The operating experience review included consideration of the results of programmatic assessments performed by Davis-Besse and of those performed by outside agencies, including the NRC. Past corrective actions resulting in program enhancements are included in the evaluation of program effectiveness. Industry operating experience was considered for existing programs and for new programs. The operating experience review provides objective evidence that the effects of aging will be managed for the period of extended operation.

B.1.5 AGING MANAGEMENT PROGRAMS

Table B-1 provides a listing of the NUREG-1801 aging management programs and the corresponding aging management programs for Davis-Besse. Table B-2 provides a summary of the aging management programs for Davis-Besse with respect to consistency with NUREG-1801 aging management programs. Table B-2 also provides information on whether programs are existing or new, whether enhancements are required, and whether the programs are plant-specific. Each aging management program credited for license renewal is addressed in Section B.2.

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B.2 AGING MANAGEMENT PROGRAMS

The correlation between NUREG-1801 programs and Davis-Besse aging management programs is shown in the following table. The table is organized by the NUREG-1801 program number, first for Chapter X, then for Chapter XI, and finally for plant-specific programs.

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
NUREG-1801 Chapter X and XI		
X.E1	Environmental Qualification (EQ) of Electrical Components	Environmental Qualification (EQ) of Electrical Components Program See Section B.2.14.
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Fatigue Monitoring Program See Section B.2.16.
X.S1	Concrete Containment Tendon Prestress	Not applicable. Davis-Besse has a free-standing steel containment vessel with a reinforced concrete Shield Building that does not contain pre-stressed tendons, as described in USAR Section 3.8.2.
XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Inservice Inspection Program See Section B.2.24.
XI.M2	Water Chemistry	PWR Water Chemistry Program See Section B.2.33.
XI.M3	Reactor Head Closure Studs	Reactor Head Closure Studs Program See Section B.2.34.
XI.M4	BWR Vessel ID Attachment Welds	Not applicable. Davis-Besse is a PWR.
XI.M5	BWR Feedwater Nozzle	Not applicable. Davis-Besse is a PWR.
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not applicable. Davis-Besse is a PWR.
XI.M7	BWR Stress Corrosion Cracking	Not applicable. Davis-Besse is a PWR.
XI.M8	BWR Penetrations	Not applicable. Davis-Besse is a PWR.

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs (continued)

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
XI.M9	BWR Vessel Internals	Not applicable. Davis-Besse is a PWR.
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Program See Section B.2.6.
XI.M11	Nickel-Alloy Nozzles and Penetrations	Plant-specific aging management program is credited for aging management; Nickel-Alloy Management Program (See Section B.2.28).
XI.M11A	Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors	Nickel-Alloy Reactor Vessel Closure Head Nozzles Program See Section B.2.29.
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not credited for aging management. Davis-Besse has no CASS components other than pump casings and valve bodies subject to thermal embrittlement. As reduction of fracture toughness of these components is managed by the Inservice Inspection Program (See Section B.2.24), a program similar to XI.M12 is not required.
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not credited for aging management. The only CASS components subject to both thermal and radiation embrittlement are portions of the reactor vessel internals. As reduction of fracture toughness of these components is managed by the PWR Reactor Vessel Internals Program (See Section B.2.32), a program similar to XI.M13 is not required.
XI.M14	Loose Parts Monitoring	Not credited for aging management. This program is not credited for aging management of any line item in NUREG-1801 Section IV.
XI.M15	Neutron Noise Monitoring	Not credited for aging management. This program is not credited for aging management of any line item in NUREG-1801 Section IV.
XI.M16	PWR Vessel Internals	Plant-specific aging management program is credited for aging management; PWR Reactor Vessel Internals Program (See Section B.2.32).

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs (continued)

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
XI.M17	Flow-Accelerated Corrosion	Flow-Accelerated Corrosion (FAC) Program See Section B.2.19.
XI.M18	Bolting Integrity	Bolting Integrity Program See Section B.2.4.
XI.M19	Steam Generator Tube Integrity	Steam Generator Tube Integrity Program See Section B.2.38.
XI.M20	Open-Cycle Cooling Water System	Open-Cycle Cooling Water Program See Section B.2.31.
XI.M21	Closed-Cycle Cooling Water System	Closed Cooling Water Chemistry Program See Section B.2.8.
XI.M22	Boraflex Monitoring	Plant-specific aging management program is credited for aging management; Boral® Monitoring Program (See Section B.2.5). Spent fuel racks at Davis-Besse use Boral® as the neutron absorber (rather than Boraflex).
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Cranes and Hoists Inspection Program See Section B.2.10.
XI.M24	Compressed Air Monitoring	Not credited for aging management. Operating experience shows that the air and gas is dry except in certain locations for which the One-Time Inspection (See Section B.2.30) is credited. In addition, the plant-specific Air Quality Monitoring Program (See Section B.2.3) ensures that compressed air in the Instrument Air System is dry and free of contaminants.
XI.M25	BWR Reactor Water Cleanup System	Not applicable. Davis-Besse is a PWR.
XI.M26	Fire Protection	Fire Protection Program See Section B.2.17.
XI.M27	Fire Water System	Fire Water Program See Section B.2.18.

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs (continued)

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
XI.M28	Buried Piping and Tanks Surveillance	Not credited for aging management. NUREG-1801 XI.M34 is an acceptable option and is credited for aging management. See the Buried Piping and Tanks Inspection Program (See Section B.2.7).
XI.M29	Aboveground Steel Tanks	Aboveground Steel Tanks Inspection Program See Section B.2.2.
XI.M30	Fuel Oil Chemistry	Fuel Oil Chemistry Program See Section B.2.20.
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance Program See Section B.2.35.
XI.M32	One-Time Inspection	One-Time Inspection See Section B.2.30.
XI.M33	Selective Leaching of Materials	Selective Leaching Inspection See Section B.2.36.
XI.M34	Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection Program See Section B.2.7.
XI.M35	One-time Inspection of ASME Code Class 1 Small-Bore Piping	Small Bore Class 1 Piping Inspection See Section B.2.37.
XI.M36	External Surfaces Monitoring	External Surfaces Monitoring Program See Section B.2.15.
XI.M37	Flux Thimble Tube Inspection	Not credited for aging management. Davis-Besse is a Babcock & Wilcox design that does not have flux thimble tubes.

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs (continued)

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
XI.M38	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Not credited for aging management. The External Surfaces Monitoring Program (See Section B.2.15) is credited instead for aging management of internal surfaces where the internal and external environments are the same (e.g., air-indoor uncontrolled). Confirmation that aging is not occurring on internal surfaces that are not the same as the external environment (i.e., internal environments of air-outdoor or condensation) is provided by the One-Time Inspection (See Section B.2.30).
XI.M39	Lubricating Oil Analysis	Lubricating Oil Analysis Program See Section B.2.26.
XI.S1	ASME Section XI, Subsection IWE	Inservice Inspection (ISI) Program – IWE See Section B.2.22.
XI.S2	ASME Section XI, Subsection IWL	Not applicable. Davis-Besse has a free-standing steel containment vessel with a reinforced concrete Shield Building that does not contain pre-stressed tendons, as described in USAR Section 3.8.2.
XI.S3	ASME Section XI, Subsection IWF	Inservice Inspection (ISI) Program – IWF See Section B.2.23.
XI.S4	10 CFR Part 50, Appendix J	10 CFR Part 50, Appendix J Program See Section B.2.1.
XI.S5	Masonry Wall Program	Masonry Wall Inspection See Section B.2.27.
XI.S6	Structures Monitoring Program	Structures Monitoring Program See Section B.2.39.
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Water Control Structures Inspection See Section B.2.40.

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs (continued)

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
XI.S8	Protective Coating Monitoring and Maintenance Program	Not credited for aging management. Davis-Besse does not credit coatings inside the Containment to manage the effects of aging for structures and components or to ensure that the intended functions of coated structures and components are maintained. Therefore, these coatings do not have an intended function and do not require aging management for license renewal.
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 Environmental Qualification Requirements Program See Section B.2.12.
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 Environmental Qualification Requirements Used in Instrumentation Circuits Program See Section B.2.13.
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 Environmental Qualification Requirements Program See Section B.2.21.
XI.E4	Metal-Enclosed Bus	Not credited for aging management. Davis-Besse does not utilize metal-enclosed bus.
XI.E5	Fuse Holders	Not applicable. A review of Davis-Besse documents indicated that fuse holders utilizing metallic clamps are either part of an active electrical panel or are located in circuits that perform no license renewal intended function. Therefore, fuse holders with metallic clamps at Davis-Besse are not subject to aging management review.
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 Inspection See Section B.2.11.

Table B-1
Correlation of NUREG-1801 and Davis-Besse Aging Management Programs (continued)

Number	NUREG-1801 Program	Corresponding Davis-Besse AMP
Davis-Besse Plant-Specific Programs		
N/A	Plant-Specific Program	Air Quality Monitoring Program See Section B.2.3.
N/A	Plant-Specific Program	Boral® Monitoring Program See Section B.2.5.
N/A	Plant-Specific Program	Collection, Drainage, and Treatment Components Inspection Program See Section B.2.9.
N/A	Plant-Specific Program	Leak Chase Monitoring Program See Section B.2.25
N/A	Plant-Specific Program	Nickel-Alloy Management Program See Section B.2.28.
N/A	Plant-Specific Program	PWR Reactor Vessel Internals Program See Section B.2.32.

Table B-2
Consistency of Davis-Besse Aging Management Programs with NUREG-1801

Program Name	New / Existing	Consistent with NUREG-1801	Consistent with NUREG-1801 with Exceptions	Plant-Specific	Enhancement Required
10 CFR Part 50, Appendix J Program Section B.2.1	Existing	Yes	--	--	--
Aboveground Steel Tanks Inspection Program Section B.2.2	Existing	Yes	--	--	Yes
Air Quality Monitoring Program Section B.2.3	Existing	--	--	Yes	--
Bolting Integrity Program Section B.2.4	Existing	--	Yes	--	--
Boral® Monitoring Program Section B.2.5	New	--	--	Yes	--
Boric Acid Corrosion Program Section B.2.6	Existing	Yes	--	--	--
Buried Piping and Tanks Inspection Program Section B.2.7	Existing	Yes	--	--	Yes
Closed Cooling Water Chemistry Program Section B.2.8	Existing	--	Yes	--	--
Collection, Drainage, and Treatment Components Inspection Program Section B.2.9	New	--	--	Yes	--

Table B-2
Consistency of Davis-Besse Aging Management Programs with NUREG-1801 (continued)

Program Name	New / Existing	Consistent with NUREG-1801	Consistent with NUREG-1801 with Exceptions	Plant-Specific	Enhancement Required
Cranes and Hoists Inspection Program Section B.2.10	Existing	Yes	--	--	--
Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection Section B.2.11	New	Yes	--	--	--
Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program Section B.2.12	New	Yes	--	--	--
Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program Section B.2.13	New	Yes	--	--	--

Table B-2
Consistency of Davis-Besse Aging Management Programs with NUREG-1801 (continued)

Program Name	New / Existing	Consistent with NUREG-1801	Consistent with NUREG-1801 with Exceptions	Plant-Specific	Enhancement Required
Environmental Qualification (EQ) of Electrical Components Program Section B.2.14	Existing	Yes	--	--	--
External Surfaces Monitoring Program Section B.2.15	Existing	Yes	--	--	Yes
Fatigue Monitoring Program Section B.2.16	Existing	Yes	--	--	Yes
Fire Protection Program Section B.2.17	Existing	--	Yes	--	--
Fire Water Program Section B.2.18	Existing	Yes	--	--	Yes
Flow-Accelerated Corrosion (FAC) Program Section B.2.19	Existing	Yes	--	--	--
Fuel Oil Chemistry Program Section B.2.20	Existing	--	Yes	--	--
Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program Section B.2.21	New	Yes	--	--	--

Table B-2
Consistency of Davis-Besse Aging Management Programs with NUREG-1801 (continued)

Program Name	New / Existing	Consistent with NUREG-1801	Consistent with NUREG-1801 with Exceptions	Plant-Specific	Enhancement Required
Inservice Inspection (ISI) Program – IWE Section B.2.22	Existing	Yes	--	--	--
Inservice Inspection (ISI) Program – IWF Section B.2.23	Existing	Yes	--	--	--
Inservice Inspection Program Section B.2.24	Existing	Yes	--	--	--
Leak Chase Monitoring Program Section B.2.25	Existing	--	--	Yes	--
Lubricating Oil Analysis Program Section B.2.26	Existing	Yes	--	--	--
Masonry Wall Inspection Section B.2.27	Existing	Yes	--	--	Yes
Nickel-Alloy Management Program Section B.2.28	Existing	--	--	Yes	--
Nickel-Alloy Reactor Vessel Closure Head Nozzles Program Section B.2.29	Existing	Yes	--	--	--
One-Time Inspection Section B.2.30	New	Yes	--	--	Yes
Open-Cycle Cooling Water Program Section B.2.31	Existing	--	Yes	--	--

Table B-2
Consistency of Davis-Besse Aging Management Programs with NUREG-1801 (continued)

Program Name	New / Existing	Consistent with NUREG-1801	Consistent with NUREG-1801 with Exceptions	Plant-Specific	Enhancement Required
PWR Reactor Vessel Internals Program Section B.2.32	New	--	--	Yes	--
PWR Water Chemistry Program Section B.2.33	Existing	Yes	--	--	--
Reactor Head Closure Studs Program Section B.2.34	Existing	Yes	--	--	Yes
Reactor Vessel Surveillance Program Section B.2.35	Existing	Yes	--	--	Yes
Selective Leaching Inspection Section B.2.36	New	Yes	--	--	--
Small Bore Class 1 Piping Inspection Section B.2.37	New	Yes	--	--	--
Steam Generator Tube Integrity Program Section B.2.38	Existing	Yes	--	--	--
Structures Monitoring Program Section B.2.39	Existing	Yes	--	--	Yes
Water Control Structures Inspection Section B.2.40	Existing	--	Yes	--	Yes

B.2.1 10 CFR PART 50, APPENDIX J PROGRAM

Program Description

The 10 CFR Part 50, Appendix J Program monitors Containment leak rate. Containment leak rate tests are required to assure that: (a) leakage through primary Containment and systems and components penetrating primary Containment will not exceed allowable values specified in technical specifications, and (b) periodic surveillance of primary Containment penetrations and isolation valves is performed so that proper maintenance and repairs are made. Appendix J, Option B is utilized. The Containment leak rate tests are performed in accordance with the guidelines contained in NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program" (as modified by approved exceptions) and NEI 94-01, "Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J."

NUREG-1801 Consistency

The 10 CFR Part 50, Appendix J Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.S4, "10 CFR Part 50, Appendix J."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

For Davis-Besse, the integrated leakage rates for Type A tests, including any additions for B and C leakage rate test penalties or volume change corrections, have been less than the maximum allowable leakage rates specified in the Technical Specifications.

During the Cycle 15 refueling outage, approximately 47% of the local leak rate tests were performed to fulfill inservice inspection pressure testing requirements. Since the Cycle 15 refueling outage was the only refueling outage scheduled during the inspection interval period, all local leak rate tests performed to fulfill pressure testing requirements had to be completed. An electrical penetration assembly exceeded its individual component administrative leakage criteria but subsequently returned to within limits. A containment isolation valve exceeded its individual component administrative leakage criteria. The administrative limit for the isolation valve was temporarily raised. The isolation valve was reworked. The Minimum Pathway Leakage total is less than 25% of allowable for both Total Bypass and Combined Total.

The results of the most recent Type A test are shown below. No Type A tests have failed to meet their acceptance criteria at Davis-Besse. The NRC reviewed the last Type A test during the Cycle 13 refueling outage (March 2002 - 2004) and found it to have been performed successfully.

Test Results:

Date	Outage	Test Results Type A As-left	Acceptance Criteria	Performance Criteria
April 2003	Cycle 13	0.1671 wt.% / day	0.75 L_a (0.375 wt.% / day)	1.0 L_a (0.5 wt.% / day)

Conclusion

The 10 CFR Part 50, Appendix J Program has been demonstrated to be capable of detecting and managing aging effects for the Containment and systems and components penetrating primary Containment. The continued implementation of the 10 CFR Part 50, Appendix J Program provides reasonable assurance that the aging effects will be managed such that the Containment will continue to perform its intended function consistent with the current licensing basis for the period of extended operation.

B.2.2 ABOVEGROUND STEEL TANKS INSPECTION PROGRAM

Program Description

The Aboveground Steel Tanks Inspection Program manages the effects of corrosion on the external surfaces and inaccessible locations of the steel fire water storage tank and diesel oil storage tank. The Aboveground Steel Tanks Inspection Program is a condition monitoring program that consists of periodic visual inspections for loss of material, and a volumetric examination of the tank bottoms. This program includes an assessment of the condition of tank surfaces protected by a coating, although the paint is not credited to perform a preventive function. Performing inspection of the tank bottoms ensures that degradation or significant loss of material will not occur in inaccessible locations. The frequency of tank bottom volumetric inspection will be based on the findings of an inspection performed prior to the period of extended operation.

NUREG-1801 Consistency

The Aboveground Steel Tanks Inspection Program is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M29, "Aboveground Steel Tanks."

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Scope, Parameters Monitored or Inspected, Detection of Aging Effects, Monitoring and Trending, Acceptance Criteria**

  The program will be enhanced to include a volumetric examination of tank bottoms to detect evidence of loss of material due to crevice, general, or pitting corrosion, or to confirm a lack thereof. The enhancement will include establishing the examination technique, the inspection locations, and the acceptance criteria for the examination of the tank bottoms. Unacceptable inspection results will be entered into the Corrective Action Program. The volumetric examination of the tank bottoms will be performed prior to the period of extended operation.

Operating Experience

The Aboveground Steel Tanks Inspection Program is an ongoing program for which plant operating experience has shown the system walkdowns to effectively manage the effects of corrosion on the external surfaces of the fire water storage tank and the diesel oil storage tank. The visual inspection methods are consistent with accepted industry practices.

The system walkdown activities have identified numerous cases of paint degradation, including flaking, blistering, peeling, and chipping throughout the plant. This confirms that the visual inspections are capable of detecting the condition of painted surfaces. No cases of corrosion degradation specific to the tank exterior surfaces were identified.

In 2002, an inspection of the exterior of the diesel oil storage tank revealed rust and corrosion at the base flange of the tank and corroded bolts at the lower access plate at the base of the tank. The work order system was used to address painting and preservation of the corroded areas of the tank.

In 2008, an inspection of the exterior of the tank revealed minor paint blemishes (scratches and chipping) with no corrosion. The work order system was used to address cleaning and repainting of the affected areas.

Corrosion at the sand to metal interface on the bottom of the fire water storage tank is recognized as an area of interest. The tank design is such that it sits on a layer of oiled sand over compacted fill with the tank bottom six inches above grade. No cases of corrosion degradation specific to the bottom exterior surface of the tanks were identified. Inspection prior to the period of extended operation will determine the condition of the tank bottom. The timing and techniques for inspection of the tank bottom will consider industry operating experience with similar configurations. Industry operating experience is monitored by the site on an ongoing basis.

Conclusion

The Aboveground Steel Tanks Inspection Program has been demonstrated to be capable of managing loss of material for the accessible external surfaces of the fire water storage tank and the diesel oil storage tank. The continued implementation of the Aboveground Steel Tanks Inspection Program, with enhancement, provides reasonable assurance that the effects of aging will be managed such that the tanks will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

B.2.3 AIR QUALITY MONITORING PROGRAM

Program Description

The purpose of the Air Quality Monitoring Program is to ensure that the Instrument Air System remains dry and free of contaminants, to ensure that there are no aging effects requiring management. The program is based on existing commitments to NRC Generic Letter 88-14 and comprises periodic air quality sampling from the Instrument Air System. The Air Quality Monitoring Program is implemented via the work order system. The Air Quality Monitoring Program is a preventive program.

NUREG-1801 Consistency

The Air Quality Monitoring Program is an existing plant-specific program for Davis-Besse. While NUREG-1801 includes a Compressed Air Monitoring Program (XI.M24), the Air Quality Monitoring Program is considered plant-specific, and is therefore evaluated against the 10 elements described in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The scope of the Air Quality Monitoring Program includes periodic sampling of the air quality in the Instrument Air System piping and piping components to ensure that the compressed air environment remains dry and free of contaminants, thereby ensuring that there are no aging effects requiring management for this system. These components are exposed to compressed air during normal operation. The Air Quality Monitoring Program includes periodic sampling of system air quality, consistent with Generic Letter 88-14, and corresponding actions, if unacceptable moisture or contaminants are detected.

- Preventive Actions
  The Air Quality Monitoring Program includes periodic sampling of the air quality of components in the Instrument Air System, to ensure that the air remains dry and free of contaminants.

- Parameters Monitored or Inspected
  As described in the *Preventive Actions* element above, the Air Quality Monitoring Program periodically samples the compressed air within components of the Instrument Air System for hydrocarbons, dew point, and particulates to verify proper air quality and ensure that the intended function of the system is maintained.

- Detection of Aging Effects
  The Air Quality Monitoring Program does not directly inspect for or detect the effects of aging in the Instrument Air System. Rather, as described for the *Preventive Actions* element above, the presence of an environmental stressor (moisture), which could lead to corrosion of system components, is detected and moisture, if any, is removed to ensure air quality (and intended function) is maintained.

- Monitoring and Trending
  Air quality sampling of the Instrument Air System is performed periodically with a frequency dependent on the results of previous testing. Results are sent to the plant or system engineer and are available for trending analysis as necessary.

- Acceptance Criteria
  Acceptance criteria for compressed air are specified for particulates (< 2.0 milligrams per cubic meter for < 3 micron particles), hydrocarbons (< 1.0 parts per million), and dew point (1 of 3 readings must be ≤ -37°F dew point atmospheric) (as necessary) for sampling of the Instrument Air System. If specified acceptance criteria are not met, then the failure is entered into the Corrective Action Program which drives corrective actions to meet the acceptance criteria.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  As described in the Davis-Besse responses to Generic Letter 88-14, and confirmed by subsequent site operating experience, air quality monitoring continues to show that the instrument air is dry and contaminant free. There have been no failures or significant degradation of components in the Instrument Air System. Industry operating experience is also considered in the program.

Review of Davis-Besse operating experience did not reveal a loss of component intended function for components exposed to instrument air that could be attributed to an inadequacy of the Air Quality Monitoring Program. Abnormal air system conditions are promptly identified, evaluated, and corrected.

For example, in 2007, one out of nine air samples drawn for particulate testing exceeded the Preventive Maintenance established limit. This limit was established as a threshold for further investigation. The work order system was used to investigate and characterize the system piping that produced the high particulate loading.

Enhancements

None.

Conclusion

The Air Quality Monitoring Program has been demonstrated to be capable of ensuring that the Instrument Air System remains dry and free of contaminants, thereby ensuring that there are no aging effects requiring management for this system.

B.2.4 BOLTING INTEGRITY PROGRAM

Program Description

The Bolting Integrity Program is a condition monitoring program that consists of existing Davis-Besse activities that, in conjunction with other credited programs (identified below), address the management of aging for the bolting of subject mechanical components and structural connections within the scope of license renewal. The Bolting Integrity Program relies on manufacturer and vendor information, as well as on industry recommendations for a comprehensive bolting and bolting maintenance program that addresses proper selection, assembly, and maintenance of bolting for pressure-retaining closures and structural connections.

The Bolting Integrity Program includes periodic inspection of bolted closures and connections for indications of degradation such as leakage, loss of material due to corrosion, loss of preload, and cracking due to stress corrosion cracking. It also includes preventive measures to preclude or minimize loss of preload and cracking.

The program inspections are implemented through the following other aging management programs: Inservice Inspection Program; Inservice Inspection (ISI) Program – IWE; Inservice Inspection (ISI) Program – IWF; Structures Monitoring Program; and External Surfaces Monitoring Program.

NUREG-1801 Consistency

The Bolting Integrity Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801 Section XI.M18, "Bolting Integrity," with the following exceptions.

Exceptions to NUREG-1801

<u>Program Elements Affected:</u>

- **Scope, Preventive Actions, Corrective Actions**

  The Bolting Integrity Program does not explicitly address the guidelines outlined in EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," or those as further delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants" for safety-related bolting (the programs and activities outlined in these documents apply only to safety-related bolting, and primarily to nuclear steam supply system bolting). However, the Bolting Integrity Program does rely on the recommendations of the manufacturer, the vendor and the industry in general, as contained in EPRI documents TR-104213 and TR-111472, for bolting maintenance.

- **Monitoring and Trending**

  NUREG-1801 recommends weekly or biweekly follow up inspections of bolted connections that are reported to be leaking. Periodic inspection of bolting, other than of ASME Class 1, 2, 3 and MC bolting, is performed through the External Surfaces Monitoring Program or the Structures Monitoring Program.

  Leaks that are "conditions adverse to quality" (i.e., that could result in a challenge to a system or component function) are entered into the FENOC Corrective Action Program. The FENOC Corrective Action Program is relied upon to ensure evaluations are performed and appropriate corrective actions are applied. Depending on the magnitude and significance of the leak, corrective actions may include periodic monitoring and trending of leakage.

  Leaks that do not constitute a condition adverse to quality are documented and entered into the Work Management Process. Operators performing daily rounds, Maintenance personnel in the plant, System Engineers performing walkdowns, and other personnel passing through accessible plant areas provide additional resources to identify leaks that could result in a challenge to system or component intended functions.

  Davis-Besse operating experience has not shown a need for a pre-set inspection frequency (e.g., daily, weekly, or biweekly) applicable to all cases involving bolting of pressure-retaining components.

Enhancements

None.

Operating Experience

Review of site operating experience shows that the Bolting Integrity Program has been effective in managing the effects of aging on bolted closures. A few instances of failed or improper bolting (fasteners) have been identified and some corroded bolting or closure (facing) surfaces (e.g., from general corrosion or leakage) have been identified at Davis-Besse and corrected.

Leakage from borated water systems is a primary cause of bolting degradation. The related operating experience is addressed separately for the Boric Acid Corrosion Program, and is not discussed here.

Review of refueling and outage inspection reports since 2002 and a search of the Corrective Action Program revealed instances of bolting problems, both design and degradation related, being identified and corrected via the existing activities included in the Bolting Integrity Program. Examples include:

- The head of one of two bolts holding the emergency diesel generator jacket water elbow to the head of a cylinder was found to be loose. The head came off with minimal effort. No evidence of leakage was found around the affected area. It could not be readily determined if the bolt head was over-torqued during the previous assembly, corroded while in service, or damaged during the removal of the power pack. The bolt was replaced.
- During walkdowns on multiple systems in 2002 it was determined that nut-to-bolt thread engagement varied from bolt tip flush with the nut to one thread below the surface of the nut. As a result, calculations, specifications, and an instructional memo were developed (or updated) to address acceptable nut-to-bolt thread engagement. This acceptable thread engagement information has been incorporated into related site maintenance procedures.
- A corroded expansion anchor for a tubing support was found. The subject expansion anchor had been corroded by ground water leaking through an adjacent wall penetration. The leak was corrected and the anchor bolt was repaired.

Conclusion

The Bolting Integrity Program has been demonstrated to be capable of managing loss of material, loss of preload, and cracking for the bolting of pressure-retaining mechanical components. The Bolting Integrity Program will provide reasonable assurance that the aging effects will be managed such that bolting will continue to perform its intended functions consistent with the current licensing basis for the period of extended operation.

B.2.5 BORAL® MONITORING PROGRAM

Program Description

The Boral® Monitoring Program is a new plant-specific aging management program that will be implemented prior to the period of extended operation. The Boral® Monitoring Program will provide reasonable assurance that potentially detrimental aging effects will be adequately detected such that the neutron absorber intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

The Boral® neutron absorbers exposure to the spent fuel pool environment would be less than 40 years at the end of the period of extended operation.

Boral® monitoring is not required by the current licensing basis based on the NRC Safety Evaluation Report received for the spent fuel pool re-rack project and an NRC letter to Holtec (the rack vendor) stating that there was no current requirement for in-service surveillance on Boral® in spent fuel pool storage racks.

The Boral® Monitoring Program detects degradation of Boral® neutron absorbers in the spent fuel storage racks with in situ testing. From the monitoring data, the stability and integrity of Boral® in the storage cells are assessed. Periodic monitoring of Boral® permits early determination of aging degradation. Adverse conditions will be documented in the Corrective Action Program.

NUREG-1801 Consistency

The Boral® Monitoring Program is a new plant-specific program for Davis-Besse. There is no corresponding aging management program described in NUREG-1801. The program is evaluated against the 10 elements described in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The scope of the new Boral® Monitoring Program consists of in situ testing of the Boral® neutron absorbing material in the spent fuel storage racks at Davis-Besse.

  The Boral® Monitoring Program is credited for detecting loss of material aging effects of the Boral® neutron absorbers in the spent fuel racks.

- Preventive Actions
  The program is a condition monitoring program that does not include preventive actions. No actions are taken as part of the Boral® Monitoring Program to prevent aging effects or mitigate age-related degradation.

- Parameters Monitored or Inspected
  The Boral® Monitoring Program monitors changes that can lead to loss of material or change of physical form of the Boral® neutron absorbers in the spent fuel racks. The program monitors changes in physical properties of the Boral® by in situ testing.

  The program provides for additional, optional measurement parameters and actions, including radiography, destructive wet chemical analysis or inspection of the Boral® panels. These additional actions provide options for confirming or further investigating results of in situ testing.

- Detection of Aging Effects
  The Boral® Monitoring Program monitors the condition of the absorber material with in situ testing. Visual inspections and measurements, as appropriate, are used to determine and assess the extent of degradation in the Boral® before there is a loss of intended function.

- Monitoring and Trending
  In situ testing of Boral® will provide information on the radiological effects, thermal effects, and chemical effects of the spent fuel pool environment on the Boral® panels. Visual inspections determine the extent of loss of material. These inspections will be reported in a manner which allows trending of results.

- Acceptance Criteria
  The most significant measurements taken are for evaluation of thickness (to monitor for swelling). There is no evidence that neutron attenuation testing (to confirm the concentration of Boron-10 in the Boral®) will serve any useful purpose. Based on the monitoring methods used, acceptance criteria for measurements will be established prior to the period of extended operation. Changes in excess of the acceptance criteria will require investigation and engineering evaluation to identify whether further testing or corrective actions may be necessary.

  Other measurement parameters will also be examined for early indications of the potential onset of Boral® degradation that would suggest a need for further attention. These include:

  - Visual or photographic evidence of unusual geometric changes
  - The existence of areas of reduced boron density

- Corrective Actions
This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
The Boral® Monitoring Program is a new aging management program proposed for the period of extended operation. No similar program has existed, therefore no specific plant operating experience is available. A review of the Corrective Action Program did not identify instances of Boral® aging at Davis-Besse. Boral® monitoring is not required by the current licensing basis based on the Safety Evaluation Report received for the spent fuel pool re-rack project and an NRC letter to Holtec stating that there was no current requirement for in-service surveillance on Boral® in spent fuel pool storage racks. All available industry operating experience for Boral® shows that there has been no significant reduction in Boral® neutron attenuation capability.

  A search of industry experience revealed the same conclusions as described above except for a Boral® blistering issue. Boral® blistering has been observed in the industry. These cases were deemed to be 10 CFR Part 21 issues. Root cause analysis and additional testing concluded that blisters did not affect neutron attenuation and did not affect the structural integrity of Boral® canisters. While Boral® is subject to both generalized corrosion and local corrosion in spent fuel pools, the overall performance to date has been acceptable. This conclusion is based on the data from utility coupon surveillance programs that have shown no reduction in Boron-10 loading due to these effects. Similarly, in-pool blistering of Boral® has, to date, proved to be primarily an esthetic effect. However, potential effects on fuel assembly clearance and the reactivity state of racks have been described.

  FENOC re-racked the spent fuel pool in the Cycle 13 refueling outage (February 2002 to March 2004) with Boral® as the neutron absorber. As a result, the Boral® neutron absorbers exposure to the spent fuel pool environment would be less than

40 years at the end of the period of extended operation. The overall performance of the Boral® at Davis-Besse currently (less than 10 years) would be similar to the results evaluated by EPRI from industry coupon surveillance programs such that the Boral®'s neutron attenuation capability remains acceptable. An EPRI report on neutron absorber materials contains a compilation of data and operating experience for all neutron absorber materials used or proposed for spent fuel storage and transportation applications over the last 40 years.

The NRC Safety Evaluation Report issued for the Davis-Besse spent fuel pool re-rack project states that Boral® is the neutron absorbing material used in the new spent fuel pool rack arrays. Boral® is a hot-rolled ceramic-metal (cermet) of aluminum and boron carbide clad in 1100 alloy aluminum. Boron carbide has a high boron content and is physically stable and chemically inert. Boral® also provides a high cross-section for removing thermal neutrons. The 1100 alloy aluminum provides corrosion resistance through a hydrated aluminum oxide film that develops on the surface, within a few days, after exposure to the atmosphere or water. As this film forms, the corrosion layer penetrates the surface of the aluminum cladding only a few microns with no net loss of aluminum cladding. Hydrogen, a byproduct of the corrosion process, may cause deformation of the sheathing holding the Boral® panels attached to the racks resulting in deformation of the storage cells. To prevent this degradation from occurring, the Boral® is contained in a sheathing cavity attached to the racks with spot welding, allowing the gases to vent. The neutron absorbing capability of Boral® is not affected by this corrosion process. Based on the evaluation, the NRC staff concluded that the materials used in the fabrication of the spent fuel rack arrays are compatible with the spent fuel pool environment at Davis-Besse. The degradation of the sheathing holding the Boral® panels is prevented by venting the corrosion hydrogen byproduct. In addition, the corrosion process does not affect the neutron absorbing capability of Boral®. Therefore, the materials used in the new spent fuel rack arrays are acceptable to the NRC staff.

In October 2009, the NRC issued Information Notice (IN) 2009-26 which provides industry operating experience on the degradation of neutron absorbing materials in spent fuel pools. IN 2009-26 addressed issues of degradation of the Carborundum neutron-absorbing materials and the deformation of Boral® panels in spent fuel pools. The operating experience on degradation of Boral® is applicable to Davis-Besse. IN 2009-26 described Beaver Valley inspections in 2007 of the Boral® neutron absorber material coupons that identified numerous blisters of the aluminum cladding, while only a few small blisters had been identified in 2002. In region 1 fuel storage racks, blisters can displace water from the flux traps between storage cells and challenge dimensional assumptions used in the criticality analysis. Based on these inspections, FENOC determined that the Boral® aluminum cladding blistering was an aging effect and that it would credit the existing Boral® Surveillance Program with management of this aging effect at Beaver Valley. The other operating experience was at Susquehanna where the licensee had identified a significant bulge in a poison can wall. Although the licensee has not definitively determined the cause of the bulge, the licensee's letter states that it may be the result of hydrogen gas generation from either moisture contained in the Boral® at the time of manufacture or a leaking seal weld in the poison can. This bulge prevented the placement of a blade guide into the deformed cell. The spent fuel racks at Davis-Besse are vented to prevent this condition.

In May 2010, the NRC issued License Renewal Interim Staff Guidance LR-ISG-2009-01, "Aging Management of Spent Fuel Pool Neutron-Absorbing Materials other than Boraflex," providing guidance as to one acceptable approach for managing the effects of aging during the period of extended operation for neutron-absorber material in spent fuel pools within the scope of the License Renewal Rule. Recent operating experience has documented several instances of degradation and deformation of the neutron-absorber materials in the spent fuel pools of operating reactors, as described in IN 2009-26. LR-ISG-2009-01 highlighted that a plant-specific aging management program should be submitted that addresses neutron-absorber material in order to detect and mitigate the aging of the material that could impact the neutron-absorbing function during the period of extended operation. The applicant should consider both plant-specific and industry operating experience.

Enhancements

None.

Conclusion

The new plant-specific Boral® Monitoring Program will provide reasonable assurance that potentially detrimental aging effects will be adequately detected such that the Boral® neutron absorber intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

B.2.6 BORIC ACID CORROSION PROGRAM

Program Description

The Boric Acid Corrosion Program manages the effects of boric acid leakage on the external surfaces of structures and components potentially exposed to boric acid leakage. The Boric Acid Corrosion Program is a condition monitoring program consisting of visual inspections.

The Boric Acid Corrosion Program is an existing program that provides for management of loss of material due to boric acid corrosion. The program includes provisions to identify, inspect, examine and evaluate leakage, and initiate corrective action. The program relies in part on implementation of recommendations of NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Components in PWR Plants." The Boric Acid Corrosion Program ensures that the pressure boundary integrity and material condition of the subject structures and components are maintained consistent with the current licensing basis during the period of extended operation.

NUREG-1801 Consistency

The Boric Acid Corrosion Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M10, "Boric Acid Corrosion."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

As documented in Licensee Event Report (LER) 2002-02, significant degradation of the original Davis-Besse reactor vessel closure head was discovered. Performance deficiencies in the implementation of the boric acid corrosion control program and Corrective Action Program allowed the reactor coolant system pressure boundary leakage to occur undetected for a prolonged period of time resulting in the head degradation. Program compliance reviews were performed to ensure proper interface with supporting plant programs, proper consideration of industry experience, proper staffing, and timely resolution of identified weaknesses. Detailed reviews were performed to ensure the programs were conducted in accordance with the governing processes.

The current Boric Acid Corrosion Program incorporates the recommendations of Generic Letter (GL) 88-05 and additionally includes consideration of the systems outside Containment that contain boric acid.

Quarterly health reports are prepared for the Boric Acid Corrosion Program. The health reports evaluate the overall program and the specifics of program personnel, infrastructure, implementation, and equipment performance.

A self-assessment of the Boric Acid Corrosion Program was conducted in October 2008. The assessment identified a strength in conservatively obtaining management approval for temporary delay of an inspection for boric acid. Improvements included identifying acceptance criteria for pump seal leakage, ensuring that conclusion statements in the Corrective Action Program have sufficient level of detail to summarize the issue and resolution, monitoring the effectiveness of corrective actions for packing adjustments.

As documented in NRC inspection report 05000346/2008002, the NRC performed a review in 2008 of the boric acid corrosion control inspection activities against commitments made in response to GL 88-05. The inspection activities included plant walkdowns, review of procedures and records, and review of Corrective Action Program documentation, including corrective actions. The NRC report concluded that no findings of significance were identified.

NRC inspection report 05000346/2007003 documented that FENOC performed a detailed, systematic evaluation of the Boric Acid Corrosion Control program, and made comprehensive programmatic improvements to the program. The NRC found that the programmatic boric acid issues that resulted in LER 2002-02 were properly resolved.

A self-assessment of the Boric Acid Corrosion Program was conducted in November 2005. The assessment noted a strength in the use of computer based training to facilitate personnel qualifications. The program was found to be effectively implemented, meeting current industry requirements, and to have incorporated industry beneficial practices.

Conclusion

The Boric Acid Corrosion Program has been demonstrated to be capable of managing loss of material due to boric acid corrosion for susceptible structures and components. The continued implementation of the Boric Acid Corrosion Program provides reasonable assurance that the aging effects will be managed such that structures and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.7 BURIED PIPING AND TANKS INSPECTION PROGRAM

Program Description

The Buried Piping and Tanks Inspection Program manages the effects of corrosion on the external surfaces of piping, tanks and associated bolting exposed to a buried (soil) environment. Piping and tanks that are not in contact with a soil environment are not within the scope of this program. The Buried Piping and Tanks Inspection Program is a combination of a mitigation program (consisting of protective coatings) and a condition monitoring program (consisting of visual inspections). The Buried Piping and Tanks Inspection Program ensures that the intended function of the subject components will be maintained consistent with the current licensing basis during the period of extended operation.

The Buried Piping and Tanks Inspection Program manages loss of material for steel piping, tanks and associated bolting, which are provided with protective coatings. The program also manages loss of material due to corrosion for gray cast iron piping and piping components, which are not provided with protective coatings. Loss of material due to selective leaching of gray cast iron is managed by the Selective Leaching Inspection.

The buried piping and piping components within the scope of this program are in the following plant systems:

- Emergency Diesel Generators (EDG) System – fuel oil subsystem
- Fire Protection System
- Service Water System

The buried tanks within the scope of the program are the EDG Fuel Oil Storage Tanks (DB-T153-1, DB-T153-2).

The buried bolting within the scope of the program is associated with the Fire Protection System piping.

Degradation or leakage found during inspections is entered into the FENOC Corrective Action Program to ensure evaluations are performed and appropriate corrective actions are taken.

NUREG-1801 Consistency

The Buried Piping and Tanks Inspection Program is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M34, "Buried Piping and Tanks Inspection."

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Scope**

  Add the emergency diesel fuel oil storage tanks (DB-T153-1, DB-T153-2) to the scope of the program. The existing program scope includes only buried piping.

  Add bolting for buried Fire Protection System piping to the scope of the program.

- **Detection of Aging Effects**

  Add a requirement that an inspection of coated and wrapped buried piping or tank be performed within the 10-year period prior to entering the period of extended operation (i.e., between year 30 and year 40). Specify that if an opportunistic inspection has not occurred between year 30 and year 38, then an excavation of a section of coated and wrapped buried piping or tank for the purpose of inspection will be performed before year 40.

  Add a requirement that an additional inspection of coated and wrapped buried piping or tank be performed within 10 years after entering the period of extended operation (i.e., between year 40 and year 50). Specify that if an opportunistic inspection has not occurred between year 40 and year 48, then an excavation of a section of coated and wrapped buried piping for the purpose of inspection will be performed before year 50.

  Add a requirement that an inspection of uncoated cast iron buried piping be performed within the 10-year period prior to entering the period of extended operation (i.e., between year 30 and year 40). Specify that if an opportunistic inspection has not occurred between year 30 and year 38, then an excavation of a section of uncoated cast iron buried piping for the purpose of inspection will be performed before year 40.

  Add a requirement that an additional inspection of uncoated cast iron buried piping be performed within 10 years after entering the period of extended operation (i.e., between year 40 and year 50). Specify that if an opportunistic inspection has not occurred between year 40 and year 48, then an excavation of a section of uncoated cast iron buried piping for the purpose of inspection will be performed before year 50.

Add a requirement that an inspection of buried Fire Protection System bolting will be performed when the bolting becomes accessible during opportunistic or focused inspections.

Add a requirement that the inspections of buried piping be conducted using visual (VT-3 or equivalent) inspection methods. Also, to ensure that a sufficient inspection area of the buried component is exposed, the excavation shall be of approximately 10 linear feet of piping, with all surfaces of the pipe exposed.

Operating Experience

A search of Davis-Besse operating experience identified an Emergency Diesel Generator (EDG) underground fuel oil piping leak due to corrosion that appeared to be the result of damage to the piping coating and wrapping. The leak was first documented in the Corrective Action Program in 1995 and the piping system was repaired in 1997. Later evaluations of the fuel oil piping conditions concluded that a more robust cathodic protection system could further mitigate piping damage due to coating and wrapping deficiencies. A new cathodic protection system was installed in 2008 for the fuel oil piping.

An assessment of the condition of the external surfaces of buried piping was also performed in 2002. The assessment found no signs of significant degradation of the buried piping. One holiday on the coatings for the emergency diesel fuel oil supply piping was identified and repaired. Another assessment was recommended.

The second assessment of the condition of the external surfaces of buried piping was performed in 2008. UT inspection to determine the condition of the external surfaces of buried circulating water piping was performed in January 2008. The UT was performed from the inside due to the depth of the buried piping. The inspections determined the piping to be in good condition. The Corrective Action Program documents (August 2008) damaged coatings (holidays) on three sections of buried emergency diesel fuel oil lines with instances of pitting and minor corrosion. Two areas of coating damaged were thought to be the result of probe strikes in an earlier effort to locate the buried piping. A UT examination was performed on the areas where pitting was identified. The wall thickness was found to be greater than the nominal thickness for the pipe and was determined acceptable. The defects were considered to be minor and the overall condition of the pipe was noted to be very good.

The Corrective Action Program documents (October 2008) a leak in buried carbon steel piping associated with a three-inch condensate demineralizer backwash line. A corrective action was the establishment of a buried piping integrity program for Davis-Besse. The root cause of the piping leak was identified as general corrosion due to coating damage and a non-functioning cathodic protection system. The degraded section of piping was replaced with polyethylene plastic piping. A second item in the Corrective Action Program documents (also October 2008) damaged coating on buried Circulating Water System blowdown piping expected to have resulted from excavation associated with repair of the condensate demineralizer backwash line. Prior to repairing the damaged coating, UT of the piping determined the wall thickness to be acceptable.

The industry has issued EPRI TR-1016456, "Recommendations for an Effective Program to Control the Degradation of Buried Pipe," which includes a six step process to have an effective buried piping program. FENOC has implemented the program, which has identified all systems and components potentially susceptible to the buried piping conditions and their risk of degradation through a Systems Susceptibility Risk Ranking Criteria. The criteria include radiological process fluid, EPA concern, safety related, Limiting Condition for Operation risk, and others.

Davis-Besse operating experience demonstrates that the coating of buried steel piping and tanks is now effective in managing the effects of aging. Plant design considerations addressed the potential for degradation of buried steel piping and tanks through the application of protective coatings. Review of site operating experience demonstrates that the uncoated cast iron piping is resistant to corrosion in the buried environment by virtue of no identified instances of noted degradation or failures. Industry operating experience has been addressed in the implementation of the EPRI buried piping program, and will continue to be addressed as industry operating experience is gained.

Conclusion

The Buried Piping and Tanks Inspection Program has been demonstrated to be capable of managing loss of material due to corrosion for piping in buried (soil) environments. The continued implementation of the Buried Piping and Tanks Inspection Program, with enhancement, provides reasonable assurance that the effects of aging on buried piping, tanks and bolting will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.8 CLOSED COOLING WATER CHEMISTRY PROGRAM

Program Description

The purpose of the Closed Cooling Water Chemistry Program is to mitigate damage due to loss of material, cracking, and reduction in heat transfer of plant components within the scope of license renewal that contain treated water in a closed cooling water system or component (e.g., heat exchanger) served by or connected to a closed cooling water system. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material, cracking, or reduction in heat transfer through proper monitoring and control of corrosion inhibitor concentrations consistent with current EPRI water chemistry guidelines. The Closed Cooling Water Chemistry Program is a condition monitoring and mitigation program.

The Closed Cooling Water Chemistry Program also includes corrosion rate measurement at selected locations in the closed cooling water systems. In addition, the Closed Cooling Water Chemistry Program is supplemented by the One-Time Inspection, which provides verification of the effectiveness of the program in managing the effects of aging.

NUREG-1801 Consistency

The Closed Cooling Water Chemistry Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M21, "Closed-Cycle Cooling Water System," with the following exceptions.

Exceptions to NUREG-1801

Program Elements Affected:

- **Parameters Monitored or Inspected, Detection of Aging Effects, Monitoring and Trending, and Acceptance Criteria**

  The program does not include performance or functional testing for aging management. Based on Davis-Besse operating experience, the Closed Cooling Water Chemistry Program has been determined to be effective in maintaining the intended functions of subject components in closed cooling water systems without the use of performance monitoring or functional testing. However, it does include measurement of corrosion rates in select locations, via corrosion coupons, and inspections of opportunity when systems are open for maintenance. The corrosion coupons are periodically replaced and evaluated to provide information on the effectiveness of the chemical treatment program and corrosion rate data.

In addition, to confirm adequate condition monitoring and mitigation of loss of material and cracking in low flow and stagnant areas and adequate mitigation of reduction in heat transfer, the program is supplemented by the One-Time Inspection, which includes closed cooling water system locations and heat exchangers served by closed cooling water systems.

Enhancements

None.

Operating Experience

The Closed Cooling Water Chemistry Program is an ongoing program that incorporates EPRI closed cooling water guidelines as well as "lessons learned" from operating experience. The program is subject to assessment of its ability to manage the relevant conditions that could lead to or are indicative of a loss of material, cracking, or reduction in heat transfer of components.

A recent internal assessment was performed to assess the programs for the primary, secondary, and auxiliary closed cooling water systems. The assessment found that for the auxiliary systems, which include component cooling water and emergency diesel generator jacket water, the chemistry parameters are being sampled and analyzed in accordance with the chemistry procedures. The major conclusion developed was that the action levels and responses in the procedure are generally consistent with those provided in the EPRI Closed Cooling Water Chemistry Guidelines. However, enhancements were recommended for frequency gaps and action level responses. The assessment resulted in improvements to the program to ensure consistency with the EPRI Closed Cooling Water Chemistry Guidelines.

During the data review for the fourth quarter 2008 Closed Cooling Water Chemistry Quarterly Report it was determined that the Davis-Besse typical closed cooling water sulfate concentration has historically been above the current EPRI guideline specification of 150 ppb for hydrazine-treated systems. All other closed cooling water chemistry parameters were found to be within the current EPRI guideline values. A review of corrosion coupon corrosion rate trends since 2000 determined consistent rates of less than 0.1 millimeters per year for all metals which is an indication of "excellent" corrosion control in the system. Sulfate monitoring frequency was increased from monthly to weekly until the value was returned to less than 150 parts per billion in April 2009.

Review of Corrective Action Program documents indicates that abnormal chemistry conditions are identified, evaluated, and corresponding adjustments made, through the corrective action process, to correct the chemistry conditions before a loss of component intended function, and that industry operating experience is considered for impact to the program.

For example, in 2008, an evaluation of nitrite levels was performed in the EDG Jacket Water System that were outside the station specification levels but less than the EPRI action level for high nitrite. The controlling chemistry procedure was enhanced to ensure actions are included when exceeding the station upper limit, including evaluation of microbiological activity trend and other control parameters.

In 2004, an event at McGuire was evaluated in which their CCW System experienced a buildup of nitrogen gas due to naturally occurring bacteria in the water that produces nitrogen as a byproduct. Sodium Nitrite, the corrosion inhibitor at the time, is a nutrient source for bacteria which enable them to proliferate and thereby produce nitrogen gas. This OE was screened out for Davis-Besse because a different corrosion inhibitor is used and biocide additions are made as needed.

Review of Davis-Besse operating experience did not reveal a loss of component intended function of subject components exposed to closed cooling water that could be attributed to an inadequacy of the Closed Cooling Water Chemistry Program.

Conclusion

The Closed Cooling Water Chemistry Program has been demonstrated to be capable of managing loss of material, cracking, and reduction in heat transfer for susceptible components through monitoring and control of the corrosion inhibitor concentrations and relevant parameters in closed cooling water systems and the components that are connected to or served by those systems. The Closed Cooling Water Chemistry Program, as supplemented by the One-Time Inspection, provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.9 COLLECTION, DRAINAGE, AND TREATMENT COMPONENTS INSPECTION PROGRAM

Program Description

The Collection, Drainage, and Treatment Components Inspection Program is a new plant-specific program for Davis-Besse. The program will consist of visual inspections of steel or other metal components exposed to raw (untreated) water, that are not covered by other aging management programs, for evidence of loss of material, as well as cracking or reduction in heat transfer. Opportunistic inspections, when surfaces are accessible during maintenance, repair, or surveillance, will ensure that the existing environmental conditions in collection, drainage, and treatment service are not causing material degradation that could result in a loss of component intended function during the period of extended operation. If an opportunistic inspection has not been conducted prior to the period of extended operation, a focused inspection will be conducted prior to entering the period of extended operation. The Collection, Drainage, and Treatment Components Inspection Program is a condition monitoring program.

NUREG-1801 Consistency

The Collection, Drainage, and Treatment Components Inspection Program is a new plant-specific program for Davis-Besse. There is no corresponding aging management program described in NUREG-1801. The program is evaluated against the 10 elements described in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The scope of the Collection, Drainage, and Treatment Components Inspection Program includes visual inspections of the internal surfaces of copper alloy (including copper alloy greater than 15% zinc), gray cast iron, stainless steel (including cast austenitic stainless steel), and steel components exposed to untreated water, in collection, drainage, or treatment service, that are not covered by other aging management programs. These inspections will ensure that the existing environmental conditions are not causing cracking, loss of material, or reduction in heat transfer that could result in a loss of component intended functions.

  The environmental conditions vary depending on the service, from water maintained by the PWR Water Chemistry Program up to the point of drainage, potable water for treatment of Control Room air, raw fire protection water or diesel fire pump coolant or makeup water, to miscellaneous collection, plumbing, or drainage water.

The piping and components (filter bodies, flexible connections, heat exchanger shell and tubes, humidifier tubing, orifices, pump casings (including bolting), rupture discs, strainer bodies, tanks, tubing, and valve bodies) in the scope of this program are in the following systems:

- Auxiliary Building Heating, Ventilation and Air Conditioning (HVAC) – Control Room Normal Ventilation System
- Fire Protection System (including diesel fire pump)
- Gaseous Radwaste System
- Makeup and Purification System
- Makeup Water Treatment System
- Miscellaneous Liquid Radwaste System
- Reactor Coolant Vent and Drain System
- Spent Fuel Cooling and Cleanup System
- Station Plumbing, Drains, and Sumps System

Loss of material due to selective leaching of gray cast iron or copper alloy greater than 15% zinc components in the raw (untreated) water environment will be managed separately by the Selective Leaching Inspection.

- Preventive Actions
  The Collection, Drainage, and Treatment Components Inspection Program does not include any actions to prevent or mitigate the effects of aging. It is a condition monitoring program.

- Parameters Monitored or Inspected
  Inspections of the surfaces of collection, drainage, treatment, and other miscellaneous components that are exposed to raw (untreated) water, but are not addressed by other aging management programs, will be performed during maintenance and surveillance activities, when the surfaces are accessible for inspection.

  If opportunities for inspection do not arise, then a focused inspection will be performed as described for the *Detection of Aging Effect*s element below.

  Parameters monitored or inspected are directly related to degradation of the components under review and include visible evidence of material degradation due to, loss of material (corrosion), as well as due to cracking, of susceptible materials, or reduction in heat transfer (fouling) for susceptible components.

- Detection of Aging Effects
  The Collection, Drainage, and Treatment Components Inspection Program provides for detection of aging effects prior to the loss of component intended function. These inspections will be opportunistic visual inspections performed when component surfaces are accessible during maintenance, repair, and surveillance activities.

  The program will be implemented after the issuance of the renewed license and prior to the end of the current operating license for Davis-Besse. If opportunistic inspections have not occurred in this time-period, then a focused inspection, inclusive of each material in the scope of the program, will be performed prior to entering the period of extended operation.

  The inspections will be conducted using visual (VT-3 or equivalent) inspection methods performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50, Appendix B. Any evidence of degradation that could lead to a loss of component intended function will be documented and evaluated through the Corrective Action Program to determine the need for subsequent inspections, expansion, and for monitoring and trending the results.

  Visual inspection by qualified personnel will detect a loss of material or fouling of surfaces exposed to raw (untreated) water prior to a loss of component function. In addition, visual inspection combined with evaluation of conditions by qualified personnel will also detect cracking of susceptible materials exposed to raw (untreated) water, at temperatures above 140°F or with ammonia or ammonium compounds present, prior to a loss of component function. These visual inspections will be supplemented by enhanced visual inspection of components susceptible to cracking.

- Monitoring and Trending
  Inspection findings will be evaluated by assigned engineering personnel. Inspection findings not meeting the acceptance criteria will be evaluated and tracked through the Corrective Action Program. The Corrective Action Program will be used to identify the corrective actions including additional inspections or expansion. Degradation of surfaces exposed to raw (untreated) water will be evaluated to determine other potentially susceptible locations. The susceptible locations will be monitored or inspected based on engineering evaluation. Trending the results of previous inspections may be used as a qualitative tool for identifying susceptible locations that may require additional examinations.

- Acceptance Criteria
  Indications or relevant conditions of degradation detected during the inspections will be compared to pre-determined acceptance criteria. If the acceptance criteria are

not met, then the indications and conditions will be evaluated under the Corrective Action Program to assess the material condition and determine whether the component intended function is affected.

Unacceptable inspection findings will include visible evidence of cracking, loss of material, or reduction in heat transfer due to fouling that could lead to loss of component intended function during the period of extended operation.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  The operating experience confirms that periodic surveillance and maintenance activities, and as-needed repairs, are conducted for components exposed to raw (untreated) water.

  For example, a review of operating experience for the diesel fire pump radiator and cooling circuit identified instances of venting and filling the radiator with coolant and monitoring the performance of the radiator and cooling circuit, but did not identify any degradation of the components or inspection of component surfaces.

  Similarly, minor corrosion was identified and evaluated on the interior surface of the fire water storage tank (FWST) in 2004. It was determined to be acceptable and not to impact component intended function.

  Review of Davis-Besse operating experience did not identify degradation that could be attributed to exposure to the raw water in the Makeup Water Treatment System, or to the water that is periodically drained from the Makeup and Purification and Spent Fuel Pool Cooling and Cleanup demineralizers.

In 2005, an evaluation and repair of a leak between the boric acid mix tank (BAMT) and miscellaneous waste drain tank (MWDT) was performed, but did not include indication of component internal condition or of the need for future inspections.

Review of Davis-Besse operating experience did not identify other failures that could be attributed to frequent or prolonged exposure to raw (untreated) component drainage water, station plumbing (domestic) water, to gaseous radwaste moisture accumulation (condensation), or miscellaneous liquid radwaste collection water.

The elements that comprise the Collection, Drainage, and Treatment Components Inspection Program inspections (i.e., the scope of the inspections and inspection techniques) will be consistent with industry practice. Industry and plant-specific operating experience will be considered in the development and implementation of this program. As additional operating experience is obtained, lessons learned will be incorporated, as appropriate.

Enhancements

None.

Conclusion

Implementation of the Collection, Drainage, and Treatment Components Inspection Program will provide reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.10 CRANES AND HOISTS INSPECTION PROGRAM

Program Description

The Cranes and Hoists Inspection Program is credited with managing loss of material for the structural components of cranes (including bridge, trolley, rails, and girders), monorails, and hoists within the scope of license renewal. The cranes, monorails and hoists within the scope of license renewal are those defined by NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and light load handling systems related to refueling.

The Cranes and Hoists Inspection Program is a condition monitoring program that is based on guidance contained in American National Standards Institute (ANSI) B30.2 for overhead and gantry cranes, ANSI B30.11 for monorail systems and underhung cranes, and ANSI B30.16 for overhead hoists. The inspections monitor structural members for signs of corrosion and wear. The inspections are performed periodically for installed cranes and hoists.

NUREG-1801 Consistency

The Cranes and Hoists Inspection Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

A review of crane and hoist inspections previously conducted at Davis-Besse, program and system health reports, the FENOC Corrective Action Program, and industry operating experience confirms the reasonableness and acceptability of the inspections and their frequency, in that degradation of cranes (including bridge, trolley, rails, and girders), monorails, and hoists was detected prior to loss of function. Related crane and hoist inspections have found isolated minor age-related degradation such as minor corrosion and paint chipping due to mechanical damage.

For example, one issue identified in the Corrective Action Program in 2009 indicated age-related degradation found while performing Intake Gantry Crane preventive maintenance. The Intake Gantry Crane is exposed to weather. The Corrective Action Program document noted that parts of the crane structure have areas of missing paint and corrosion. In areas around the bridge drive gear, bolts were degraded from corrosion. The grout on the crane bridge rails was cracked. No loose grout was noted, but the grout was considered to be susceptible to freeze thaw damage. The work order system was used to address the identified issues.

Review of select completed work orders from 2005 through 2008 and a review of plant-specific operating experience through a search of Corrective Action Program documentation from 2000 and later revealed minor issues of flaking paint and loss of material due to corrosion (e.g., polar crane handrail - 2003). A 2004 Corrective Action Program item described action taken from industry operating experience, in that several metal filings were found on the rail of a Fuel Building Overhead Crane at another nuclear plant. A follow-up communication to the crane engineer at the plant revealed that the shavings were determined to be "flaking" from the crane rails and were not metal filings from wear of the bridge wheels or rails. Corrective action taken at Davis-Besse was to add an inspection step to look for wear products on the rails, bridge wheels and trolley wheels for fuel handling and spent fuel pool cask cranes. The remaining adverse conditions identified in the Corrective Action Program dealt with issues unrelated to aging, including issues such as active components not properly working, procedural issues, rigging issues, operator qualification, clearance tagging, and human-related events.

Conclusion

The Cranes and Hoists Inspection Program has been demonstrated to be capable of detecting and managing loss of material for cranes (including bridge, trolley, rails, and girders), monorails, and hoists within the scope of license renewal. The continued implementation of the Cranes and Hoists Inspection Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.11 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS INSPECTION

Program Description

The purpose of the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection is to detect and identify aging effects for the metallic parts of electrical cable connections that are not required to be environmentally qualified but are within the scope of license renewal.

This inspection is a new activity that will address external cable connections that are used to connect cable conductors to other cables or electrical end devices, such as motor terminations, switchgear, motor control centers, bus connections, transformer connections, and passive electrical boxes such as fuse cabinets. The most common types of connections used in nuclear power plants are splices (butt splices or bolted splices), crimp-type ring lugs, connectors, and terminal blocks. Most connections involve insulating material and metallic parts. The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection will focus primarily on bolted connections. This aging management inspection will account for aging stressors such as thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation of the metallic parts. Implementation of this inspection will provide added assurance that the electrical connections in the plant have electrical continuity and are not overheating due to increased resistance (from a loosened or degraded connection). The inspection will be performed via the use of thermography, with the optional use of contact resistance testing as a supplement.

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection is a new aging management activity (a one-time inspection) that will be conducted prior to the period of extended operation.

NUREG-1801 Consistency

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection is a new one-time inspection that will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.E6, “Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” as clarified by LR-ISG-2007-02.

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The metallic parts of electrical cable connections, not subject to 10 CFR 50.49, and associated with cables that are within the scope of license renewal, are part of this activity, regardless of their association with active or passive devices. This includes external cable connections terminating at active or passive devices associated with cables that are within the license renewal scope. Wiring connections internal to an active assembly are considered part of the active assembly and are therefore not within the scope of this activity.

  The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection is applicable to non-environmentally qualified electrical cable connections that are within the scope of license renewal.

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

- Parameters Monitored or Inspected
  This one-time inspection will focus on the metallic parts of electrical cable connections. The inspection will include detection of loosened bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation. A representative sample of electrical cable connections will be inspected. The following factors will be considered for sampling: connection type (e.g., bolted splices, bolted terminations, lug terminations, bolted cable terminations), circuit application (medium, or low voltage), circuit loading (high load), and physical location (e.g., high temperature, high humidity, vibration) with respect to connection stressors. The technical basis for the sample selected will be documented. If an unacceptable condition or situation is identified in the sample, a determination is made as to whether the same condition or situation is applicable to other connections not tested. The inspection will confirm that the loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, chemical contamination, corrosion, vibration, or oxidation is not an aging effect that requires a periodic aging management program.

- Detection of Aging Effects
  A representative sample of the metallic electrical cable connections not subject to 10 CFR 50.49 environmental qualification requirements and within the scope of

license renewal will receive a one-time inspection via thermography (augmented with optional contact resistance testing) prior to the period of extended operation. Thermography is a proven test method for detecting loose connections and degraded connections (i.e., chemical contamination, corrosion, oxidation) leading to increased resistance, and will be used to test a sample of electrical connections at a variety of plant locations. Thermography can detect aging effects due to thermal cycling, ohmic heating, vibration, and electrical transients. Thermography is an effective tool for inspecting connections that are covered by close fitting electrical tape, insulating boots or covers, heat-shrink material, and sleeving. The optional use of contact resistance testing of a sample of motor termination connections and other connections will also be utilized, as applicable. The one-time inspection provides additional confirmation that the electrical connections in the plant have not experienced general or repeated failures and that existing installation and maintenance practices are effective.

- Monitoring and Trending
  No actions are taken as part of the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection to monitor or trend inspection results. This is a one-time inspection activity used to determine if, and to what extent, further actions, including monitoring and trending, may be required.

  Sample size will be determined by engineering evaluation, as described for the *Detection of Aging Effects* element above. Results of the inspection activities that require further evaluation or resolution (e.g., if degradation is detected), if any, will be evaluated using the Corrective Action Program, including expansion of the sample size and inspection locations to determine the extent of the degradation.

- Acceptance Criteria
  The acceptance criteria will be based on the acceptance criteria already used for the thermography process at Davis-Besse; the acceptance criteria for any contact resistance tests will be defined in the implementing procedure.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

  For the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection, an engineering evaluation is 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. Such an evaluation is to 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. When an unacceptable condition or situation is identified, a determination is made on whether the same condition or situation is applicable to other in-scope cable connections not tested.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Operating experience has shown that loosening of connections and corrosion of connections are aging mechanisms that, if left unmanaged, could lead to a loss of electrical continuity and potential arcing or fire. Industry operating experience that forms the basis for this program is included in the operating experience element of the corresponding NUREG-1801, aging management program description.

  Based on review of plant-specific and industry operating experience, the identified aging effects will require inspection to determine the presence (and extent) of any degradation with the non-environmentally qualified electrical cable connections.

  Plant operating experience has shown that the Corrective Action Program has addressed issues related to degraded cable connections (primarily terminations) in recent years. For example, the use of routine thermography has identified terminations at circuit breakers with elevated temperatures, typically caused by increased resistance at phase terminations. A hot spot was found on a disconnect switch in the plant switchyard, due to a misaligned phase arm on the switch. Motor control center terminations have been identified with higher temperatures (via thermography), indicating increased resistance at the termination points. The use of thermography has been effective in identifying degraded cable connections. Industry operating experience will be considered in development of this activity.

Conclusion

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection will detect and identify aging issues related to the metallic parts of non-environmentally qualified electrical cable connections. The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Inspection will provide reasonable assurance that aging effects will be identified (and addressed) such that the non-environmentally qualified electrical cable connections within the scope of this program will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

B.2.12 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Program Description

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging of electrical cables and connections that are not required to be environmentally qualified but are within the scope of license renewal that are subject to adverse localized environments. The program provides for the periodic visual inspection of accessible, non-environmentally qualified electrical cables and connections, in order to determine if age-related degradation is occurring. Accessible electrical cables and connections installed in adverse localized environments will be visually inspected for signs of accelerated age-related degradation such as embrittlement, discoloration, cracking, or surface contamination. The program will provide reasonable assurance that the electrical components will continue to perform their intended functions for the period of extended operation.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new aging management program that will be implemented prior to the period of extended operation. The visual inspections will be performed on a 10-year interval, with the first inspection taking place prior to the end of the current operating license.

NUREG-1801 Consistency

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements."

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is credited with managing aging effects from adverse localized environments in non-environmentally qualified cables and connections.

  The program inspections will be prioritized based on location rather than component identification or function.

  Particular attention will be given to the identification of adverse localized environments. The inspection program will define these areas through a review of plant engineering data (e.g., environmental qualification records, environmental surveys) and also via performance of plant walkdowns to identify adverse localized environments. An adverse localized environment is defined as a condition in a limited plant area that is significantly more severe than the specified design or bounding plant environment for the general area. Adverse localized environments are addressed in EPRI report TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments."

- Preventive Actions
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is an inspection program; no actions are taken to prevent or mitigate aging degradation. The program is based on visual observation (and detection) only.

- Parameters Monitored or Inspected
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will provide for the visual inspection of accessible cables and connections located in adverse localized environments. Adverse localized environments will be determined based upon temperature, radiation levels, and moisture levels that are significantly more severe than the specified environments for the cables and connections.

  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Program focuses on a visual inspection of accessible cables and connections. Accessible is defined as easily viewed from the ground or from a platform (including scaffolding, if utilized). The cables and connections will not be touched during the inspection (either lifted, separated, felt, or handled in any way). The inspection merely records the visible condition of the cable jacket or the visible condition of the connection (e.g., splice, terminal block, fuse block).

For inspection of connections (i.e., fuse holders), it may be necessary to open an electrical box to view the passive components. This is an acceptable practice with respect to the definition of accessible, for electrical boxes at a floor level.

Inspection of the visible portions of cables and connections (the cable jackets and the insulating bases) is reflective of the condition of the insulation.

- Detection of Aging Effects
  As described above in *Parameters Monitored or Inspected*, the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program provides for a visual inspection of a representative sample of accessible electrical cables and connections located in adverse localized environments. The visual inspections will be performed on a 10-year interval, with the first inspection taking place within the 10-year period prior to the end of the current operating license. The program will inspect the accessible cables and connections for aging effects due to adverse localized environments caused by heat, radiation, or moisture, in the presence of oxygen. The visible effects of aging are embrittlement, discoloration, cracking, and surface contamination. The visible evidence of aging (on the cable jackets and the connection insulating bases) is considered representative of aging to the cable insulation and the connection insulation.

- Monitoring and Trending
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will not include trending actions. If anomalies are found during the visual inspection process, they will be addressed through the Corrective Action Program.

- Acceptance Criteria
  The inspections of accessible cables and connections will identify visual indications of surface anomalies, such as embrittlement, cracking, discoloration, crazing, crumbling, melting, and any other distinct visual evidence of oxidation, material deterioration, or other visible degradation. If the acceptance criteria are not met, then the anomalies will be evaluated under the Corrective Action Program to determine whether they could result in a loss of component intended function during the period of extended operation.

  The implementing documents for the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will provide specific guidance on the identification of surface degradation.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

  In addition, for the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Requirements Program, all unacceptable visual indications of cable and connection jacket surface anomalies are subject to an engineering evaluation. The evaluation will consider the age and operating experience of the component, as well as the severity of the anomaly and whether the anomaly has previously been correlated to degradation of the conductor insulation or connections. Corrective actions may include, but are not limited to, testing, shielding, or otherwise changing the environment, or the 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.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Based on review of plant-specific and industry operating experience, the identified aging effects require management for the period of extended operation.

  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 adverse localized environments caused by excessive localized overheating. Examples documented in the Corrective Action Program include a cracked feeder cable for a condensate pump, and brittle and cracked thermocouple wiring for containment air cooler motor instruments. Industry operating experience will be considered in the development of this program.

Conclusion

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be capable of managing aging effects due to heat and radiation in the presence of oxygen, for non-environmentally qualified cables and connections. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will provide reasonable assurance that the aging effects will be managed such that the non-environmentally qualified cables and connections within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.13 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS PROGRAM

Program Description

The purpose of this aging management program is to manage the age-related degradation associated with high voltage, low current instrumentation cables and connections that are not required to be environmentally qualified but are within the scope of license renewal. This program addresses a subset of the overall in-scope, non-environmentally qualified cable and connection population at Davis-Besse, which is primarily addressed by the program guidelines of NUREG-1801, Section XI.E1, via visual inspection.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a condition monitoring program that applies to in-scope, non-environmentally qualified electrical cables and connections used in neutron monitoring and radiation monitoring circuits with sensitive, low current signals. The sensitive nature of these circuits is such that visual inspection alone may not detect degradation to the insulation resistance function of the conductor insulation. This program will manage the aging of the low current instrumentation cables and connections that are not required to be environmentally qualified but are within the license renewal scope.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a new aging management program that will be implemented prior to the period of extended operation.

NUREG-1801 Consistency

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a new Davis-Besse program that will be consistent with the 10 elements of an effective aging management program, as described in NUREG-1801, Section XI.E2, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits."

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is credited with identifying aging effects for sensitive, high voltage, low current signal applications that are in-scope for license renewal at Davis-Besse. These sensitive circuits are potentially subject to reduction in insulation resistance (IR) when found in adverse localized environments.

  The program is applicable to non-environmentally qualified in-scope neutron monitoring and radiation monitoring circuits.

- Preventive Actions
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a testing program designed to identify cable and connection degradation; no actions are taken to prevent or mitigate aging degradation.

- Parameters Monitored or Inspected
  The parameters monitored (tested) by the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program are either instrumentation system performance or insulation condition. In addition, the program retains the ability to utilize the NUREG-1801 (XI.E2) option of performing a calibration records review for selected circuits.

- Detection of Aging Effects
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will perform testing of the cable systems of sensitive, high voltage, low current instrumentation circuits to identify reduction in IR. The testing methodology will utilize a proven test to detect degradation of the insulation. The test methodology will be specified prior to the first test, which will occur during the 10-year period prior to the end of the current operating license. Subsequent testing will be conducted at least once every 10 years, with the frequency to be determined by engineering evaluation. Selected circuits may be evaluated via the NUREG-1801 (XI.E2) option of a calibration records review.

- Monitoring and Trending
  The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will not include trending actions. If anomalies are found during the testing process, they will be addressed at that time through the Corrective Action Program. The records of the testing will be retained so that any negative trends may be noted.

- Acceptance Criteria
  The acceptance criteria for the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will be provided by the implementing documents for the program. The test results will be evaluated against the acceptance criteria. Results outside the acceptance criteria will be evaluated in accordance with the Corrective Action Program.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

  Corrective actions such as recalibration and circuit trouble-shooting are implemented when calibration or surveillance results or findings of surveillances do not meet the acceptance criteria. An engineering evaluation is 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. Such an evaluation is to 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.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Based on review of plant-specific and industry operating experience, the identified aging effects require management for the period of extended operation.

Plant operating experience has shown that the Corrective Action Program has addressed issues of neutron detector and connection degradation in recent years. For example, in 2003, the NI-5 power range detector was experiencing intermittent connection problems on the center conductor internal to the detector. In 2002 and 2003, the source range NI-1 and NI-2 instrumentation was found to experience circuit noise due to shielding problems in the cables. While not aging related, these problems highlighted the sensitive nature of the low current instrumentation circuits. Likewise, the Corrective Action Program has identified issues with radiation monitor and connection degradation. In 2005, the radiation detector associated with RE 1413 (for Component Cooling Water) was found to be degraded due to aging. In 2009, an intermittent connection failure was noted for RE 4597BB (for the connection between the detector and the pre-amplifier). Industry operating experience will be considered in development of this program.

Conclusion

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will manage reduction in insulation resistance for non-environmentally qualified cables and connections used in sensitive, high voltage, low current circuits. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program will provide reasonable assurance that the aging effects will be managed such that the non-environmentally qualified cables and connections used in sensitive, high voltage, low current circuits, that are within the scope of this program, will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.14 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRICAL COMPONENTS PROGRAM

Program Description

The Environmental Qualification (EQ) of Electrical Components Program is an existing program that implements the requirements of 10 CFR 50.49 (as further defined and clarified by the Division of Operating Reactors (DOR) Guidelines, NUREG-0588, Regulatory Guide (RG) 1.89, Revision 1 and RG 1.97, Revision 3). The program has been established to demonstrate that certain electrical components located in harsh plant environments are qualified to perform their safety functions in those harsh environments, consistent with 10 CFR 50.49 requirements. The program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations. The program requires action to be taken before individual components in the scope of the program exceed their qualified life. Actions taken include replacement on a specified time interval of piece parts or complete components to maintain qualification and reanalysis.

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. Some aging evaluations for EQ components specify a qualification of at least 40 years and are considered time-limited aging analyses for license renewal. The program ensures that these EQ components are maintained within the bounds of their qualification bases.

Reanalysis of an aging evaluation to extend the qualifications of components is performed on a routine basis as part of the program. Important attributes for the reanalysis of an aging evaluation include analytical models, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met). These attributes are discussed below.

Analytical Models

The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years). Use of actual plant operating history to re-evaluate and establish the normal integrated radiation dose for the 60-year period may also be used. The 60-year normal radiation dose is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection and Reduction Methods

Reducing excess conservatism in the component service conditions (for example, temperature, radiation and cycles) used in the prior aging evaluation is frequently employed for a reanalysis. Temperature data used in an aging evaluation is to be conservative and based on plant design temperatures or on actual plant temperature data. When used, actual plant temperature data can be obtained in several ways, including monitors used for compliance with Technical Specifications, other installed monitors, measurements made by plant operators during rounds and temperature sensors on large motors (while the motor is not running). When used, a representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are justified on a case-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations may be used for radiation and cyclical aging.

Underlying Assumptions

EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Actions

The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is maintained, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid. The reanalysis is to be performed in a timely manner (that is, sufficient time is available to refurbish, replace, or re-qualify the component if the reanalysis is unsuccessful).

NUREG-1801 Consistency

The Environmental Qualification (EQ) of Electrical Components Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section X.E1, "Environmental Qualification (EQ) of Electrical Components."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The elements that comprise the Environmental Qualification (EQ) of Electrical Components Program are consistent with industry practice and have proven effective in maintaining the material condition of Davis-Besse plant systems and components.

The Davis-Besse EQ program includes consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended functions during accident conditions after experiencing the effects of in-service aging.

The EQ program health report (1st quarter 2009) shows the program has a "Green" status, highest ranking available, for overall program performance. In addition, the EQ program has been and continues to be subject to periodic internal and external assessments that effect continuous improvement. Administrative controls require periodic formal assessments of the EQ program by knowledgeable personnel from outside the site EQ group.

In the year 2005, a site focused self assessment was performed to evaluate the effectiveness of the Davis-Besse EQ program. The scope of the assessment was to compare the Davis-Besse EQ program documentation against the INPO Engineering Good Practice Guide, for Environmental Qualification of Electrical Equipment. Interfacing procedures and maintenance and engineering procedures which implement EQ requirements were also reviewed. The Davis-Besse EQ program was found to be effective in establishing and maintaining the environmentally qualified status of electrical equipment important to safety located in an EQ harsh environment. The assessment determined that maintenance procedures reflect EQ requirements and preventive maintenance activities are in place to perform the activities necessary to maintain EQ equipment status.

Conclusion

The Environmental Qualification (EQ) of Electrical Components Program has been demonstrated capable of managing component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations. The Environmental Qualification (EQ) of Electrical Components Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.15 External Surfaces Monitoring Program

Program Description

The External Surfaces Monitoring Program manages the aging of external surfaces, and internal surfaces in cases where environment is the same, of mechanical components within the scope of license renewal.

The External Surfaces Monitoring Program is a condition monitoring program that consists of periodic visual inspections and surveillance activities of component external surfaces to manage loss of material. The program includes components located in plant systems within the scope of license renewal that are constructed of copper alloy (copper, brass, bronze, and copper-nickel), stainless steel (including cast austenitic stainless steel), and steel (carbon and low-alloy steel and cast iron) materials. Loss of material from the external surfaces of these metals will be evidenced by surface irregularities or localized discoloration and be detectable prior to loss of intended function.

The External Surfaces Monitoring Program, supplemented by the One-Time Inspection, includes inspection and surveillance of elastomers and polymers that are exposed to air-indoor uncontrolled and air-outdoor environments, but are not replaced on a set frequency or interval (i.e., are long-lived), for evidence of cracking and change in material properties (hardening and loss of strength).

In addition, the External Surfaces Monitoring Program consists of plant-specific inspection of the following components (exposed to an air-outdoor environment) for conditions that could result in a reduction in heat transfer, evidenced by visible dirt or other foreign material buildup on tube and fin external surfaces:

- Control room emergency ventilation system air-cooled condensing unit cooling coil tubes and fins
- Station blackout diesel generator radiator tubes and fins

NUREG-1801 Consistency

The External Surfaces Monitoring Program is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M36, "External Surfaces Monitoring."

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Scope**

  Systems that credit the External Surfaces Monitoring Program for license renewal, but which do not have Maintenance Rule intended functions, will be added to the scope of the program.

- **Scope, Detection of Aging Effects**

  Surfaces that are inaccessible or not readily visible during either plant operations or refueling outages, such as surfaces that are insulated, will be inspected opportunistically during the period of extended operation.

- **Scope, Parameters Monitored/Inspected, Detection of Aging Effects, Acceptance Criteria**

  The External Surfaces Monitoring Program, supplemented by the One-Time Inspection, will perform inspection and surveillance of elastomers and polymers exposed to air-indoor uncontrolled or air-outdoor environments, but not replaced on a set frequency or interval (i.e., are long-lived), for evidence of cracking and change in material properties (hardening and loss of strength). Acceptance criteria for these components will consist of no unacceptable visual indications of cracks or discoloration that would lead to loss of function prior to the next scheduled inspection.

  The External Surfaces Monitoring Program will perform inspection and surveillance of the control room emergency ventilation system air-cooled condensing unit cooling coil tubes and fins and the station blackout diesel generator radiator tubes and fins for visible evidence of external surface conditions that could result in a reduction in heat transfer. Acceptance criteria for these components will consist of no unacceptable visual indications of fouling (build up of dirt or other foreign material) that would lead to loss of function prior to the next scheduled inspection.

Operating Experience

The elements that comprise the External Surfaces Monitoring Program are consistent with industry practice and have proven effective in maintaining the material condition of Davis-Besse plant systems and components.

A review of recent (from 2002 and later) plant-specific operating experience, through a search of plant Corrective Action Program documents, revealed that component leakage, damage, and degradation are routinely identified by the inspections and surveillance activities which comprise the External Surfaces Monitoring Program, with subsequent corrective actions taken in a timely manner; and that no loss of pressure boundary integrity has occurred that was, or could have been, attributed to the applicable aging effects that are in the scope of the program. Various Corrective Action Program items address the finding and correction of minor rust and leakage identified during station walkdown inspections, or of deficiencies that are not related to aging of passive components (but would have identified age-related degradation, if any). In addition, system health and condition is reported quarterly in plant health reports.

Conclusion

The External Surfaces Monitoring Program has been demonstrated to be capable of detecting and managing loss of material for metallic components. The continued implementation of the External Surfaces Monitoring Program, with enhancement, provides reasonable assurance that the effects of aging on both metallic and non-metallic components will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.16 FATIGUE MONITORING PROGRAM

Program Description

The Fatigue Monitoring Program manages fatigue of select primary and secondary components, including the reactor vessel, reactor internals, pressurizer, and steam generators by tracking thermal cycles as required by Technical Specification 5.5.5, "Component Cyclic or Transient Limit."

The Fatigue Monitoring Program uses the systematic counting of plant transient cycles to ensure that the design cycles are not exceeded, thereby ensuring that component fatigue usage limits are not exceeded.

The Fatigue Monitoring Program acceptance criteria are to maintain the number of counted transient cycles below the design cycles for each transient. The program periodically updates the cycle counts. When the accumulated cycles approach the design cycles, corrective action is taken to ensure the analyzed number of cycles is not exceeded. Corrective action may include update of the fatigue usage calculation. Any re-analysis will use an NRC-approved version of the ASME Code or an NRC-approved alternative (e.g., NRC-approved code case) to determine a valid cumulative usage factor.

For license renewal, the effects of the reactor coolant environment on component fatigue life have been addressed by assessing the impact of the environment on a sample of critical components as identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." Environmental effects were evaluated in accordance with NUREG/CR-6260 and the guidance of EPRI Technical Report MRP-47, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." Components identified in NUREG/CR-6260 were evaluated using material specific guidance presented 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."

NUREG-1801 Consistency

The Fatigue Monitoring Program is an existing program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary."

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Preventive Actions, Monitoring and Trending, Acceptance Criteria**

  For locations, including NUREG/CR-6260 locations, projected to exceed a cumulative usage factor (CUF) of 1.0, the program will implement one or more of the following:

  (1) Refine the fatigue analyses to determine valid CUFs less than 1.0. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

  (2) Manage the effects of aging due to fatigue at the affected locations by an inspection program that will be 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 acceptable to the NRC).

  (3) Repair or replacement of the affected locations.

- **Parameters Monitored or Inspected**

  The Fatigue Monitoring Program will be enhanced to monitor any transient where the 60-year projected cycles were used in an environmentally-assisted fatigue evaluation and to establish an administrative limit that is equal to or less than the 60-year projected cycles.

Operating Experience

Industry operating experience has been factored into the Fatigue Monitoring Program through consideration of NRC documents (information notices, bulletins, regulatory issue summaries, and regulatory guides), vendor notices, industry documents (NEI, INPO, and EPRI), and other utility license renewal applications. Specific examples of that experience showing how the Davis-Besse program has remained responsive to emerging issues and concerns, are provided below. Continued program improvements based on industry experience provide evidence that the program will remain effective for managing cumulative fatigue damage for passive components.

NRC document RIS 2008-30 deals with the use of single dimension stress factors in on-line fatigue analyses. Davis-Besse reviewed RIS 2008-30 and determined that no changes were required to the Fatigue Monitoring Program. Davis-Besse has no on-line fatigue analyses. Davis-Besse's fatigue analyses of record evaluate multi-dimensional stresses and analyze the dimensions appropriate to each component.

NRC and vendor information caused Davis-Besse to assess thermal stratification of the pressurizer surge line. This resulted in changes to the fatigue analyses of record and to the cycles being counted.

Ongoing review of industry operating experience will be used to ensure that the program continues to be effective in managing the identified aging effects.

During the program review phase of the Cycle 13 refueling outage (ended March 2004) restart effort it was discovered that the Fatigue Monitoring Program (Allowable Operating Transient Cycles program) had not been updated or reviewed for a period of approximately four years. The Corrective Action Program was used to document deficiencies in various aspects of the Fatigue Monitoring Program. This item in the Corrective Action Program was processed as a significant condition adverse to quality, with a root cause analysis performed in order to provide the appropriate level of attention to the Fatigue Monitoring Program deficiencies. As a result of the root cause analysis, several program changes were made including the addition of a requirement to perform periodic self-assessments. Other corrective actions included evaluation of monitored transients against the Babcock & Wilcox functional specification to verify the cycle limit and basis, update of transient cycle counts, comparison of accrued cycles to allowable cycles (none of the allowable cycles were exceeded), preparation of a job familiarization guide to address program owner qualification requirements, and performance of a program self-assessment.

The self-assessment report was completed in October 2005. The purpose of this assessment was to determine the effectiveness of the changes made to the Allowable Operating Transient Cycles program due to implementation of the Corrective Action Program corrective actions. In summary, the assessment determined that the procedure changes have been effective in driving the collection, documentation, and evaluation of the required transient data. The programmatic changes have been shown to be effective in providing management involvement in the program through oversight and qualification of the program owner. Updates to the allowable operating transient cycles status and event log were evident and submittals to records management were within the allowable time period.

Conclusion

The Fatigue Monitoring Program uses the systematic counting of plant transient cycles to ensure that the numbers of design cycles are not exceeded, thereby ensuring that component fatigue usage limits are not exceeded. When the accumulated cycles approach the design cycles, corrective action is taken to ensure the design cycles are not exceeded. The Fatigue Monitoring Program provides reasonable assurance that the aging effect of cracking due to fatigue, will be adequately managed and that components will continue to perform their intended functions for the period of extended operation.

B.2.17 FIRE PROTECTION PROGRAM

Program Description

The Fire Protection Program is an existing program that manages the aging effects for components in the scope of license renewal that have a fire barrier function; including fire damper framing, fire-rated penetration seals, fire wraps, fire proofing, fire doors and fire barrier walls, ceilings, and floors. In addition, the Fire Protection Program supplements the Fuel Oil Chemistry Program through performance monitoring of the diesel fire pump. The Fire Protection Program is a combination condition and performance monitoring program, comprised of tests and inspections in accordance with the applicable National Fire Protection Association (NFPA) recommendations.

NUREG-1801 Consistency

The Fire Protection Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M26, "Fire Protection," with the following exceptions.

Exceptions to NUREG-1801

Program Elements Affected:

- **Scope, Parameters Monitored or Inspected, Detection of Aging Effects, Monitoring and Trending, and Acceptance Criteria**

  Fixed Halon or carbon dioxide suppression systems are not installed within the protected area at Davis-Besse, as described in the Fire Hazards Analysis Report and corresponding safety evaluation reports. Therefore, the associated portions of the NUREG-1801, XI.M26 program are not applicable to the Fire Protection Program for Davis-Besse.

- **Acceptance Criteria**

  The Fire Protection Program does not include specific confirmation of "no corrosion in the fuel oil supply line for the diesel fire pump." Rather, the Fire Protection Program includes periodic performance testing of the diesel fire pump. Degradation noted for the fuel oil supply line during these periodic tests, if any, is evaluated prior to loss of intended function. In addition, the One-Time Inspection characterizes the internal surface condition of the fuel oil supply line (tubing) for confirmation of the effectiveness of the Fuel Oil Chemistry Program.

Enhancements

None.

Operating Experience

A review of fire barrier, fire rated penetration seal, fire wrap, fire door, and diesel fire pump system inspections previously conducted at Davis-Besse confirms the reasonableness and acceptability of the inspections and their frequency in that degradation of the subject components was detected prior to loss of function.

The NRC presently conducts triennial fire protection team inspections at the Davis-Besse site to assess whether an adequate fire protection program has been implemented and maintained at Davis-Besse. The most recent of these inspections was conducted in April of 2007. The inspection team verified that the fire protection-related issues are entered into the Corrective Action Program at an appropriate threshold for evaluation. The inspection team also reviewed the program for implementing compensatory measures in place for out-of-service, degraded, or inoperable fire protection, with no findings identified. The inspection team evaluated the adequacy of fire area barriers, penetration seals, fire doors, fire wrap, and fire rated electrical cables. The team observed the material condition and configuration of the installed barriers, seals, doors, and cables. In addition, the team reviewed Davis-Besse documentation, such as NRC safety evaluation reports, and deviations from NRC regulations and the NFPA codes to verify that fire protection features met license commitments. No findings of significance were found. In addition, a past triennial NRC inspection of the Davis-Besse Fire Protection Program, conducted in October of 2004, identified one non-significant, non-cited violation. No findings of significance were found. The violation found that previously submitted licensing correspondence, regarding the basis for not protecting ventilation system cables, was no longer accurate. This issue is not related to the portions of the program credited with aging management.

A review of recent audits, health reports, and self-assessments revealed no NRC or Davis-Besse management concerns with respect to inspection, testing, or maintenance of the Fire Protection System. These documents found the program to be effectively implemented with good performance.

A review of recent plant-specific operating experience, such as that included in Corrective Action Program documents, demonstrates that the Fire Protection Program is an effective program, consistent with industry practices. When conditions were found that required correction, they were repaired and evaluated using the work order system and the Corrective Action Program. Examples include degraded penetration seals and fire barriers that were found during periodic surveillance activities and repaired.

Conclusion

The Fire Protection Program has been demonstrated to be capable of detecting and managing the effects of aging for components in the scope of license renewal that have fire barrier intended functions. The periodic Fire Protection Program inspections and tests of the diesel fire pump fuel supply line supplement the aging management provided by the Fuel Oil Chemistry Program. The Fire Protection Program provides reasonable assurance that aging effects will be managed such that structures and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.18 FIRE WATER PROGRAM

Program Description

The Fire Water Program (a sub-program of the overall Fire Protection Program) is an existing program that applies to the fire water supply and water-based suppression systems, which include sprinkler heads (spray nozzles), fittings, valve bodies, hydrants, hose stations, standpipes, a water storage tank, and aboveground and underground piping and components. The Fire Water Program is a condition monitoring program that comprises tests and inspections in accordance with applicable NFPA recommendations.

The program is credited with managing loss of material, as well as cracking of susceptible materials, for fire water supply and water-based fire suppression components in the scope of license renewal. The periodic inspection and testing activities include hydrant and hose station inspections, fire main (and hydrant) flushes, flow tests, tank inspections, and sprinkler system inspections. Such inspection and testing assures functionality of the fire water supply and water-based suppression systems. Also, the portions of the fire water supply and water-based suppression systems that are normally maintained at required operating pressure are monitored such that leakage resulting in loss of system pressure is promptly detected and corrective actions initiated.

In addition, all sprinkler heads in the scope of license renewal will either be replaced or a sample population field service tested, prior to seeing 50 years of service (in-place) using the guidance of NFPA 25, “Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems,” 2002 edition. Sprinkler head testing, if selected, will occur at 10-year intervals following this baseline inspection, until such time as there are no untested sprinkler heads that will see 50 years of service through the end of the period of extended operation.

For fire water supply and water-based suppression systems that are not flow tested, per NFPA 25, the Fire Water Program also includes wall thickness evaluations (i.e., ultrasonic testing or internal visual inspection). These wall thickness examinations of representative fire water supply and water-based suppression piping locations that are not periodically flow tested but contain, or have contained, stagnant water are performed prior to the period of extended operation and at appropriate intervals thereafter, based on engineering evaluation of the results.

NUREG-1801 Consistency

The Fire Water Program is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M27, “Fire Water System.”

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Parameters Monitored or Inspected, Detection of Aging Effects**

  Add a program requirement to perform periodic ultrasonic testing for wall thickness of representative above-ground water suppression piping that is not periodically flow tested but contains, or has contained, stagnant water. The ultrasonic testing will be performed prior to the period of extended operation and at appropriate intervals thereafter, based on engineering evaluation of the initial results.

- **Detection of Aging Effects**

  Add a program requirement to perform at least one opportunistic or focused visual inspection of the internal surface of buried fire water piping and of similar above-ground fire water piping, within the five-year period prior to the period of extended operation, to confirm whether conditions on the internal surface of above-ground fire water piping can be extrapolated to be indicative of conditions on the internal surface of buried fire water piping.

  Add a program requirement to perform representative sprinkler head sampling (laboratory field service testing) or replacement prior to 50 years in-service (installed), and at 10-year intervals thereafter, in accordance with NFPA 25, or until there are no untested sprinkler heads that will see 50 years of service through the end of the period of extended operation.

  Add a program requirement, if certain conditions are met, to perform opportunistic fire water supply and water-based suppression system internal inspections each time a fire water supply or water-based suppression system (including fire pumps) is breached for repair or maintenance. To be considered acceptable, these internal visual inspections must be demonstrated to be: 1) representative of water supply and water-based suppression locations, 2) performed on a reasonable basis (frequency), and 3) capable of evaluating wall thickness and flow capability. If the internal inspections cannot be completed of a representative sample, then ultrasonic testing inspections will be used to complete the representative sample.

Operating Experience

Water-suppression portions (subsystems) of the Fire Protection System are inspected, tested, and maintained following NFPA recommendations and at the intervals recommended by the corresponding NFPA standards, or as evaluated and adjusted by FENOC.

The NRC presently conducts triennial fire protection team inspections at the Davis-Besse site to assess whether an adequate fire protection program has been implemented and maintained. The most recent of these inspections was conducted in March-April of 2007 and is documented in Inspection Report (IR) 2007-006. FENOC intends to adopt the NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition." Therefore, the 2007 triennial inspection was conducted in accordance with the NRC inspection procedure for the NFPA 805 transition period. There were no findings of significance during this inspection. The inspectors evaluated the adequacy of fire suppression and detection systems in select areas, including observation of material condition and configuration of the installed fire detection and suppression systems. The inspection verified that fire suppression and detection systems met license commitments. In addition, the inspectors reviewed the Corrective Action Program procedures and samples of corrective action documents and verified that FENOC was identifying issues related to the fire protection program at an appropriate threshold and entering them in the Corrective Action Program.

Another past triennial NRC inspection of the Fire Protection Program (including the Fire Water Program) was conducted in October of 2004 and documented in IR 2004-009. A single Non-Cited Violation was identified during this inspection. The Non-Cited Violation was related to licensing and the basis for an exemption being changed via modification. It was entered in the Corrective Action Program and was not related to the Fire Water Program. Otherwise, the conclusions of the 2004 inspection were similar to the results of the 2007 inspection.

No NRC concerns or Davis-Besse management concerns (through periodic audits, self-assessments, and heath reports) were identified with respect to inspection, testing, and maintenance of fire water supply or water-based suppression portions of the Fire Protection System.

A review was performed for the purposes of license renewal of Corrective Action Program documentation related to the Fire Protection System, with respect to aging effects in the fire water suppression systems. This review concluded that when conditions were found that required correction they were evaluated and corrected as necessary using the FENOC Corrective Action Program, for example, the fire water storage tank was replaced in 1998 as a result of corrosion of the internal surfaces.

Areas for improvement were also identified and implemented through the Corrective Action Program, as appropriate. In addition, for license renewal purposes, a sampling of the results of the credited surveillance and test procedures were reviewed for recent monthly, semiannual, annual, and refueling interval inspections, flushes and flow tests. Any deviations from the acceptance criteria were evaluated and corrected in accordance with the Corrective Action Program.

Conclusion

The Fire Water Program has been demonstrated to be capable of detecting and managing loss of material, as well as fouling, for susceptible components. The Fire Water Program, with enhancement, will provide reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.19 FLOW-ACCELERATED CORROSION (FAC) PROGRAM

Program Description

The Flow-Accelerated Corrosion (FAC) Program manages loss of material for steel piping and other components of systems that are susceptible to flow-accelerated corrosion, also called erosion-corrosion, when exposed to single-phase water above 190°F or two phase steam at any temperature.

The Flow-Accelerated Corrosion (FAC) Program is a condition monitoring program that implements the recommendations of NRC Generic Letter 89-08, "Erosion/Corrosion Induced Pipe Wall Thinning," and follows the guidance and recommendations of EPRI NSAC-202L, R3, "Recommendations for an Effective Flow-Accelerated Corrosion Program," to ensure that the integrity of piping systems susceptible to flow-accelerated corrosion is maintained. The program combines: a) predictive analysis, b) baseline inspections to determine the extent of thinning, and c) follow-up inspections to confirm predictions or initiate repair or replacement of components as necessary.

NUREG-1801 Consistency

The Flow-Accelerated Corrosion (FAC) Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M17, "Flow-Accelerated Corrosion."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Flow-Accelerated Corrosion (FAC) Program is a mature, well-structured program at Davis-Besse. The program implements the recommended actions of NRC Generic Letter 89-08, and is effective in managing flow-accelerated corrosion in steel piping and components containing high-energy fluids. The program has been the subject of internal assessments (with industry participation), and improvements, as well as of fleet-wide assessments (including comparison to corresponding industry peer programs). It includes the evaluation of industry operating experience for impact to the program.

A recent assessment in late 2005 found the program to be comprehensive and to meet the requirements of EPRI NSAC-202L. In the same time frame (following the Cycle 14 refueling outage), the program was enhanced, based on industry benchmarking, to implement EPRI CHECWORKS Steam Feedwater Application version 2.1, to include alloy (chrome) testing as appropriate as a tool to fine tune the flow-accelerated corrosion model and to preclude further ultrasonic testing of chrome bearing components, and to enter component data for select large and small bore not-modeled lines into CHECWORKS (which manages ultrasonic testing thickness data) to facilitate future inspections.

In 2006, a steam leak was discovered on the moisture separator reheater 1 first stage reheat drain line that should have been detected by the Flow-Accelerated Corrosion (FAC) Program but resulted in a power reduction to facilitate repairs. The program was enhanced at that time to improve the documentation on quality of the software model and to include a second level of verification for entering data into CHECWORKS.

Results of inspections and evaluations are compiled into an outage flow-accelerated corrosion report for each cycle. Flow-accelerated corrosion inspections at 95 locations were conducted during the recent Cycle 15 refueling outage. This was the first outage utilizing CHECWORKS Steam Feedwater Application version 2.1. No significant issues were noted using the updated software. Approximately 120 feet of eight-inch piping in the stage reheat drains system and approximately 160 feet of 18-inch feedwater piping were replaced with 2.25% chrome piping during the Cycle 15 refueling outage. An eight-foot section of 18-inch pipe downstream of the feedwater common section was also replaced as scheduled. An additional six-foot section of pipe downstream of a tee near the feedwater common section was replaced after ultrasonic testing thickness readings showed that the component would most likely not reach the Cycle 16 refueling outage without exceeding the minimum allowable thickness. The examination was extended beyond the thinned area. The Cycle 14 refueling outage inspection included 90 segments and one additional baseline inspection. In addition, four 18-inch locations downstream of plate type flow elements in condensate and feedwater (single-phase lines) were inspected. All piping segments inspected had a minimum allowable wall thickness calculated by design engineering prior to the start of the inspections. A planned replacement of approximately 60 feet of large bore feedwater piping was accomplished in the Cycle 14 refueling outage with 2.25% chrome (flow-accelerated corrosion resistant) material.

Conclusion

The Flow-Accelerated Corrosion (FAC) Program has been demonstrated to be capable of detecting and managing loss of material due to flow-accelerated corrosion for susceptible components. The Flow-Accelerated Corrosion (FAC) Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.20 FUEL OIL CHEMISTRY PROGRAM

Program Description

The Fuel Oil Chemistry Program verifies and maintains the quality of the fuel oil consumed in the emergency diesel generators, diesel fire pump, and station black out diesel generator in order to mitigate the effects of aging for the storage tanks and associated piping and components containing fuel oil that are within the scope of license renewal. The program manages the presence of contaminants, such as water or microbiological organisms, which could lead to the onset and propagation of loss of material or cracking through proper monitoring and control of fuel oil consistent with plant Technical Specifications and ASTM standards for fuel oil. Exposure to these contaminants is minimized by a) verifying the quality of new fuel oil before it enters the storage tanks, b) periodic sampling of tank contents to ensure the fuel oil is free of water and particulates, and c) periodic cleaning and inspection of tanks containing fuel oil. The Fuel Oil Chemistry Program is a mitigation program.

The effectiveness of the Fuel Oil Chemistry Program is verified by the One-Time Inspection. The One-Time Inspection will include ultrasonic thickness measurement of a sample of fuel oil tank bottoms to ensure that significant degradation is not occurring.

NUREG-1801 Consistency

The Fuel Oil Chemistry Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M30, "Fuel Oil Chemistry," with the following exceptions.

Exceptions to NUREG-1801

Program Elements Affected:

- **Scope, Acceptance Criteria**

  Davis-Besse does not explicitly use ASTM D6217. Davis-Besse uses ASTM D2276 versus ASTM D6217 for guidance on the determination of particulate contamination. ASTM D2276 is used, with an acceptance criterion of a total particulate contamination of less than 10 milligrams per liter.

- **Scope, Parameters Monitored or Inspected, Acceptance Criteria**

  Davis-Besse does not explicitly use ASTM D1796, and uses D4176 or D2709. ASTM D1796 provides guidance for water and sediment determination in No. 4D diesel fuel, which is not used at Davis-Besse. Davis-Besse uses ASTM D4176 for guidance on the determination of (grade 2D) fuel oil appearance or ASTM D2709 for guidance on determination of water and sediment contamination.

ASTM D4176 or ASTM D2709 is used, with acceptance criterion of clear and bright appearance with proper color, or water and sediment contamination less than 0.05% by volume, respectively.

- **Parameters Monitored or Inspected**

  A filter with a pore size of 3.0 microns is not used when testing fuel oil for particulates. Instead, a filter with 0.8 micron pore size is used, as recommended by ASTM D2276. The use of a filter with a smaller pore size results in a larger sample of particulates because smaller particles are retained. Thus, use of a 0.8 micron filter is more conservative than use of a 3.0 micron filter.

- **Detection of Aging Effects**

  Multilevel sampling is not performed. Composite samples are from three separate locations in the lower portion of the emergency diesel generator fuel oil storage tanks, where contaminants may collect.

- **Preventive Actions**

  Preventive actions do not include the routine addition of biocides, stabilizers, or corrosion inhibitors to the fuel oil. The combination of ensuring the specified physical and chemical properties of new fuel oil are within specified limits and periodic cleaning and draining of the tanks has been shown to mitigate corrosion inside the tanks and fuel oil degradation. If necessary, fuel oil additive may be used at the program owner's discretion.

Enhancements

None.

Operating Experience

The Fuel Oil Chemistry Program is an ongoing program that utilizes sampling and analysis to ensure that adequate diesel fuel quality is maintained to minimize degradation (prevent loss of material and fouling) in the various in-scope fuel oil systems. Exposure of fuel oil to contaminants such as water and particulates is also minimized by periodic draining of accumulated water, tank interior cleaning, and by verifying the quality of new oil before its introduction into the storage tanks. Furthermore, no instances of fuel oil system component failure due to instances of contamination have been identified at Davis-Besse.

Water has occasionally been discovered in various Davis-Besse diesel fuel oil storage tanks during sampling activities. In accordance with sampling and analysis procedures, any detected water is removed from the affected tank as part of the sampling process.

Abnormal fuel oil chemistry conditions, such as high particulate levels and suspended solids, are identified, evaluated, and corresponding adjustments made through the Corrective Action Program to correct the chemistry conditions well before a loss of function. Examples include:

- The monthly particulate and non particulate tests following cleaning of the fuel oil day tank for the station blackout diesel generator in 2007 were within specification; however an increase in the time to perform the particulate test for that tank was noted. Samples were reanalyzed for indications of microbiology and corrective actions taken to re-circulate tank contents through a filter.

- Higher than normal particulate levels were noted during sampling of one of the emergency diesel generator fuel oil day tanks in 2006. The tank was re-sampled with the results being more consistent with past values (and within specification). To minimize sludge/particulate transport to the diesel day tanks during preventive maintenance evolutions, corrective actions were implemented to blow excess fuel lines into the day tank using air, perform a longer purge of transport lines to remove old fuel that was in the transfer pipe, and a cautionary note added to sampling procedures.

- High particulate levels were identified in 2003 and determined to be the result of using high sulfur diesel fuel and not adding stabilizer to the fuel. After additional evaluation, it was determined that the use of low sulfur diesel would ensure the operational control limits will be more consistently met. The use of alternate fuel stabilizers to ensure the tank inventory did not degrade was recommended.

Cleaning and visual inspection of fuel oil tanks is also conducted on a regular basis. These inspections have revealed acceptable conditions for the tank internal surfaces; that is, no significant material loss or obvious changes to the condition of the tank. Some minor corrosion was noted at the top of one of the emergency diesel generator fuel oil storage tanks during scheduled cleaning of the tank in 2003. This fuel oil storage tank corrosion led to partial clogging of fuel filters and was evaluated for continued use, but did not reveal a loss of component function of subject components that contain fuel oil which could be attributed to an inadequacy of the Fuel Oil Chemistry Program. Also, regular cleaning of the diesel fire pump day tank was implemented in 2002 as a result of an evaluation of a clogged filter. The station blackout diesel generator fuel oil day tank was recently cleaned and inspected in 2006 with no issues.

An important element of fuel oil (or any other) analysis is operation of the testing laboratory. Fuel oil samples from Davis-Besse are sent to Beta Laboratory (a FENOC subsidiary) after an initial set of factors are measured at the Davis-Besse site. The laboratory completes the oil analysis.

A fleet oversight quality assurance audit was conducted to assess the operation practices and regulatory compliance of the Beta Laboratory facility. The principal tool for this assessment was the FENOC Quality Assurance Program Manual. The audit identified multiple areas for improvement and Corrective Action Program items were generated to document and track the recommended improvements.

Conclusion

The Fuel Oil Chemistry Program has been demonstrated to be capable of managing loss of material, as well as cracking, in fuel oil for susceptible components through monitoring and control of contaminants. The continued implementation of the Fuel Oil Chemistry Program, supplemented by the One-Time Inspection, provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.21 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Program Description

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will manage the aging of non-environmentally qualified inaccessible medium-voltage electrical cables susceptible to aging effects caused by moisture and voltage stress, such that there is reasonable assurance that the cables will perform their intended function in accordance with the current licensing basis during the period of extended operation.

In-scope, inaccessible medium-voltage cables exposed to significant moisture and significant voltage will be 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 to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed. Testing will be conducted at least once every 10 years, with initial testing to be completed prior to the period of extended operation.

In addition, manholes associated with inaccessible non-EQ medium-voltage cables will be inspected for water accumulation and the water removed, as necessary. These inspections for water collection will be conducted at least once every two years, with the initial inspection to be completed prior to the period of extended operation.

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new aging management program that will be implemented prior to the period of extended operation.

NUREG-1801 Consistency

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.E3, "Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements."

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program applies to inaccessible, non-environmentally qualified medium-voltage (2-kV to 35-kV) cables within the scope of license renewal that are exposed to significant moisture simultaneous with significant voltage exposure.

  The program defines significant moisture as periodic exposure to moisture that lasts more than a few days (e.g., cable in standing water). Periodic exposure to moisture, which lasts less than a few days (i.e., normal rain and drain) is not significant.

  The program defines significant voltage exposure as being subject to system voltage for more than 25% of the time.

  The program defines "inaccessible" cable as cable that is located in conduit, duct bank, cable trenches or troughs, underground vaults, or is direct buried.

- Preventive Actions
  The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will require periodic preventive actions to inspect for water collection in electrical manholes and for water removal, if necessary. Inspections will be conducted at least once every two years, with the initial inspection to be completed prior to the period of extended operation.

- Parameters Monitored or Inspected
  The specific type of test to be utilized in the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be determined prior to the initial test, and is to be a proven test (such as partial discharge, power factor, or other test that is state-or-the-art at the time the testing is to be performed) for detecting the deterioration of the insulation system due to wetting (and energization). Testing of in-scope, inaccessible medium-voltage cables exposed to significant moisture and significant voltage will provide an indication of the condition of the conductor insulation.

- Detection of Aging Effects
  The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will test in-scope medium-voltage cables at least once every 10 years, with the first tests completed prior to the period of extended operation.

  The program will also conduct inspections of the electrical manholes at least once every two years. The inspection frequency will be based on actual plant experience with water accumulation in the manhole, with the first inspection to be completed prior to the period of extended operation.

- Monitoring and Trending
  The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will not include trending actions. If anomalies are found during the testing, they will be addressed at that time via the Corrective Action Program.

- Acceptance Criteria
  The acceptance criteria for each test in the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be defined by the specific type of test to be performed and the specific cable tested.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

  In addition, for the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Program, an engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the electrical cables can be maintained consistent with the current licensing basis. Such an evaluation will consider the significance of the test results, the operability of the components, the reportability of the event, the extent of concern, the potential root causes, the corrective actions required, and the likelihood of recurrence. When an unacceptable condition or situation is identified, a determination will be made as to whether the same condition or situation is applicable to other in-scope medium-voltage cables.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Based on review of plant-specific and industry operating experience, the identified aging effects require management for the period of extended operation.

  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 and exposure to wetting. There have also been failures of medium-voltage cable at Davis-Besse, both cables within the license renewal scope and cables that are not in-scope.

  The Davis-Besse response to Generic Letter 2007-01 contains a listing of the inaccessible or underground power cable failures, a listing of degraded cables identified through testing (prior to failure), and a description of testing activities on the electrical power cables. The Generic Letter 2007-01 response is documented in FENOC Letter to NRC, Serial 3333, "Response to NRC Generic Letter 2007-01 (TAC No. MD4320)," dated May 8, 2007 and FENOC Letter to NRC, L-08-013, "Supplemental Information Regarding Response to Generic Letter 2007-01 (TAC No. MD4320)," dated January 18, 2008.

  For example, in 1999, component cooling water pump #2 tripped due to a cable fault caused by prolonged exposure to water. The cable was replaced. In 2002, the feed cables to makeup pump #1 were found to have low insulation resistance; they were replaced. In 2004, an underground feed cable associated with a 13.8-kV breaker failed, resulting in the loss of circulating water pump #1 and two nonsafety 4-kV substations.

  In addition, as part of the Maintenance Rule program, inspections have been performed on various electrical manholes at Davis-Besse (as part of a structural inspection). Evaluation worksheets were prepared for each manhole inspected, photographs were taken, and the as-found conditions were documented. There are also preventive maintenance orders for performing inspections of the in-scope electrical manholes, which address water intrusion, the wireway, the conduits, the manhole sump pumps, and the electrical supports.

  There are also regular preventive maintenance activities (inspections and repair, if necessary) performed on the electrical manholes. The work activity includes a visual check of the conduit and raceway supports in the manholes, and a functional check of installed sump pumps. If water is found, the manholes are pumped out. All of the in-scope manholes at Davis-Besse have been inspected in recent years (2005 through 2008), with some water intrusion noted (from an inch or so on the floor, up to three feet of water). The manholes with water were pumped out. No submergence of safety-related cables was noted.

The quarterly Plant Health Report includes a system health evaluation of the medium-voltage AC system. A large part of this evaluation involves underground medium-voltage cables. The evaluation addresses Davis-Besse and industry operating experience on medium-voltage cable issues, and also provides a listing of cables that are planned to be replaced in the near future.

Industry operating experience will be considered in development of this program, along with input from EPRI guidance documents.

Conclusion

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will manage degradation of conductor insulation for inaccessible, non-environmentally qualified medium-voltage cables, and will also provide for inspection of the electrical manholes (and draining of the manholes, if necessary). The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will provide reasonable assurance that the aging effects will be managed such that the inaccessible, non-environmentally qualified medium-voltage cables within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.22 INSERVICE INSPECTION (ISI) PROGRAM – IWE

Program Description

The Inservice Inspection (ISI) Program – IWE establishes responsibilities and requirements for conducting ASME Code Section XI, Subsection IWE inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWE includes examination and/or testing of accessible surface areas of the steel containment vessel; containment hatches and airlocks; seals, gaskets and moisture barriers; and containment pressure-retaining bolting. These examinations are in accordance with the requirements of the ASME Code, Section XI, 1995 Edition through the 1996 Addenda.

The inservice examinations conducted throughout the service life of Davis-Besse will continue to comply with the requirements of the ASME Code Section XI edition and addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC. This is consistent with NRC Statements Of Consideration associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

NUREG-1801 Consistency

The Inservice Inspection (ISI) Program – IWE is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.S1, "ASME Section XI, Subsection IWE."

The Code year (e.g., 1992 Edition through 2001 Edition including the 2002 and 2003 Addenda), as endorsed by the NRC in 10 CFR 50.55a, is specifically included in the NUREG-1801 XI.S1 aging management program. Consistent with provisions in 10 CFR 50.55a to use the ASME Code in effect twelve months prior to the start of the inspection interval, the applicable ASME Code for the current Third Ten-Year Inspection Interval for Davis-Besse is ASME Section XI, 1995 Edition, through the 1996 Addenda.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

Davis-Besse containment examinations and tests required by the Inservice Inspection program have been implemented in accordance with the established schedule.

There have been three conditions identified which have required engineering evaluation or repair or replacement activities.

1. Prior to the implementation date of IWE, the "sand pocket" in the annulus was found to hold moisture which resulted in scale on the containment vessel surface in this region. The sand and scale were removed from this area and the containment vessel in this area was recoated. When the scale was removed, pitting of the containment vessel was identified. Ultrasonic thickness measurements verified that the minimum recorded vessel thickness was greater than the minimum required wall thickness. An engineering evaluation determined that the pitting was not detrimental to the containment vessel. The cause of the moisture in the sand pocket region was plugged floor drains.

2. During the Cycle 12 refueling outage, seepage of water between the containment vessel and the floor of the sand pocket in the annulus was noted. Similar seepage was also noted during the Cycle 13 refueling outage and documented in the Corrective Action Program. A plant modification was implemented to add a moisture barrier to this region. The seepage wets the containment vessel at the interface between the containment vessel and the floor only. Access to perform examinations in this area is not available. Therefore, this area was addressed in the Cycle 12 and Cycle 13 refueling outages in accordance with 10 CFR 50.55a(b)(2)(ix)(A), which requires that when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to inaccessible areas, the following information be provided in the Inservice Inspection summary report required by ASME Section XI, IWA-6000:

   (1) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

   (2) An evaluation of each area, and the result of the evaluation, and;

   (3) A description of necessary corrective actions.

3. Corrective Action Program documentation identified that gaps had formed at two areas between the containment vessel and the concrete ledge on the inside of Containment at the 565-foot elevation. Although no actual degradation has been identified as a result of these gaps, the affected areas were designated as surface areas requiring augmented examination (examination category E-C) as required by ASME Section XI, IWE-1240. Access to these areas of the containment vessel is only available from one side in the annulus area. Ultrasonic thickness readings were taken in these areas from the annulus in Cycle 13 and Cycle 15 refueling outages in accordance with ASME Code Case N-605 requirements. The thicknesses in these areas

have remained essentially unchanged since the initial Cycle 13 refueling outage ultrasonic thickness readings.

All of the examinations scheduled since the third period of the second inspection interval have been completed. All of these examinations and tests performed to date have satisfied the acceptance standards contained within ASME Section XI, IWE-3000. Inservice inspection records are maintained in accordance with ASME Section XI, IWA 6000 in permanent plant file storage.

Conclusion

The Inservice Inspection (ISI) Program – IWE has been demonstrated to be capable of detecting and managing loss of material for steel surfaces of the containment. The continued implementation of Inservice Inspection (ISI) Program – IWE provides reasonable assurance that the aging effects will be managed such that the structures and components will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

B.2.23 INSERVICE INSPECTION (ISI) PROGRAM – IWF

Program Description

The Inservice Inspection (ISI) Program – IWF establishes responsibilities and requirements for conducting ASME Code Section XI, Subsection IWF inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWF includes visual examination for supports based on sampling of the total support population. The sample size varies depending on the ASME class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Classes 2 and 3). Discovery of support deficiencies during regularly scheduled inspections triggers an increase of the inspection scope, in order to ensure that the full extent of deficiencies is identified. The primary inspection method employed is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. These examinations are in accordance with the requirements of the ASME Code, Section XI, 1995 Edition through the 1996 Addenda.

The in-service examinations conducted throughout the service life of Davis-Besse will continue to comply with the requirements of the ASME Code Section XI edition and addenda incorporated by reference in 10 CFR 50.55a(b) twleve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC. This is consistent with NRC Statements Of Consideration associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

NUREG-1801 Consistency

The Inservice Inspection (ISI) Program – IWF is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.S3, "ASME Section XI, Subsection IWF."

The Code year (e.g., 1989 Edition through 2001 Edition including the 2002 and 2003 Addenda), as endorsed by the NRC in 10 CFR 50.55a, is specifically included in the NUREG-1801 XI.S3 aging management program. Consistent with provisions in 10 CFR 50.55a to use the ASME Code in effect twelve months prior to the start of the inspection interval, the applicable ASME Code for the current Third Ten-Year Inspection Interval for Davis-Besse is ASME Section XI, 1995 Edition, through the 1996 Addenda.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

Davis-Besse IWF examinations required by the Inservice Inspection program have been implemented in accordance with the established schedule.

Review of Cycle 15, 14, and 13 refueling outage Inservice Inspection summary reports and plant operating experience did not reveal age-related issues that impaired intended functions with regards to ASME Class 1, 2, or 3 supports pertaining to ASME Section XI, Subsection IWF. There have been no conditions identified which have required engineering evaluation, repair, or replacement activities.

1. During the Cycle 14 refueling outage while performing an ISI examination of hangers SW-41-HBC-47-H7 and SW-41-HBC-46-H3 rusted areas were recorded on the I-beams supporting the service water (SW) piping. The rust and rust streaks appeared to be from the humidity condensing on the SW pipe and dripping onto the support I-beams. No evidence of material wastage was noted. These conditions were documented in the Corrective Action Program and evaluated. No corrective action was required.

2. In 2006, while performing a visual examination of sway strut CC-36-HBC-2-H7 for the ISI program, proper thread engagement of the strut paddle bolts could not be verified through the sight hole in the sway strut barrel. This was applicable for both the north and south struts, top strut paddle bolts. These conditions were documented in the Corrective Action Program. A review of the "as-found" condition of the sway strut upper pinned connections determined that the sway strut had been capable of performing its design function even with reduced thread engagement on one of the four threaded connections.

3. In 2005 corrosion was noted during visual inspection of snubber DB-SNC488 on pipe support AF-M1155/H5. The corrosion was noted on the snubber extension eyelet at the pipe clamp and its associated pin. This condition was documented in the Corrective Action Program. Design engineering classified the rust on the extension piece and its snubber as rust staining with areas of minor surface rust. The corrosion appeared to have been caused by age and exposure to a humid environment in containment. There was no loss of material due to this corrosion. The corrosion at the bracket and pin was minor and did not affect the rotation ability of the snubber. Corrective action was taken to inspect this snubber during the Cycle 14 refueling outage. The subject snubber was replaced as routine maintenance and not due to failure

All of the examinations scheduled since the third period of the second inspection interval have been completed. All of the examinations and tests performed to date have satisfied the acceptance standards contained within ASME Section XI, IWF-3000. Inservice inspection records are maintained in accordance with ASME Section XI, IWA 6000 and are in permanent plant file storage.

Conclusion

The Inservice Inspection (ISI) Program – IWF has been demonstrated to be capable of detecting and managing ASME Class 1, 2, and 3 piping supports and supports other than piping supports (Class 1, 2, and 3). The continued implementation of Inservice Inspection (ISI) Program – IWF provides reasonable assurance that aging effects will be managed such that applicable structures and components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.24 INSERVICE INSPECTION PROGRAM

Program Description

The Inservice Inspection Program manages cracking of reactor coolant pressure boundary components and once-through steam generator secondary side components. The Inservice Inspection Program, in conjunction with the PWR Water Chemistry Program, manages loss of material for once-through steam generator secondary side components. The Inservice Inspection Program also manages reduction in fracture toughness for cast austenitic stainless steel pump casings and valve bodies. The Inservice Inspection Program is a condition monitoring program that meets the inservice inspection requirements specified by the ASME Code, Section XI, as modified by 10 CFR 50.55a.

The Inservice Inspection Program includes periodic visual, surface, or volumetric examination and leakage (pressure) testing of ASME Class 1, 2, or 3 components, and their integral attachments, as well as repair, modification, or replacement of same. The inservice examinations (and pressure tests) conducted throughout the service life of Davis-Besse will continue to comply with the requirements of the ASME Code Section XI, Subsections IWB, IWC, and IWD, edition and addenda incorporated by reference in 10 CFR 50.55a(b), twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC. This is consistent with NRC Statements of Consideration associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

The Inservice Inspection Program has been augmented to include commitments made to the regulatory authorities beyond the ASME Code, Section XI. Examples include the augmented examination of auxiliary feedwater header components, high pressure injection ASME Class 1 piping welds, and decay heat removal ASME Class 1 pipe to valve welds.

The Inservice Inspection Program is an existing program that will be continued for the period of extended operation.

NUREG-1801 Consistency

The Inservice Inspection Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD.”

The Code year (e.g., 2001 Edition including the 2002 and 2003 Addenda), as endorsed by the NRC in 10 CFR 50.55a, is specifically included in the NUREG-1801 XI.M1 aging management program. Consistent with provisions in 10 CFR 50.55a to use the ASME

Code in effect twelve months prior to the start of the inspection interval, the applicable ASME Code for the current Third Ten-Year Inspection Interval for Davis-Besse is ASME Section XI, 1995 Edition, through the 1996 Addenda, as modified by 10 CFR 50.55a or relief granted in accordance with 10 CFR 50.55a.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

Based on review of plant-specific and industry operating experience, the identified aging effects require management for the period of extended operation.

Recent Davis-Besse operating experience related to inservice inspection is documented in inservice inspection outage summary reports. Specific examples of inservice inspection findings are also documented in the Corrective Action Program. Davis-Besse operating experience is consistent with industry experience; a large number of examinations are being performed, and indications are found and resolved. The extensive site-specific operating experience with the Inservice Inspection Program provides assurance that the program is effective in managing the effects of aging so that components crediting these programs can perform their intended function consistent with the current licensing basis during the period of extended operation.

The Corrective Action Program and an ongoing review of industry operating experience will be used to ensure that the program remains effective in managing the identified aging effects.

Conclusion

The Inservice Inspection Program has been demonstrated to be capable of managing cracking for components of the reactor coolant pressure boundary and steam generator secondary side components, for managing reduction of fracture toughness of cast austenitic stainless steel pump casings and valve bodies, and, in conjunction with the PWR Water Chemistry Program, for managing loss of material for steam generator secondary side components. The Inservice Inspection Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.25 LEAK CHASE MONITORING PROGRAM

Program Description

The Leak Chase Monitoring Program is an existing condition monitoring program, consisting of observation and activities to detect leakage from the spent fuel pool, the fuel transfer pit, and the cask pit liners due to age-related degradation.

The Leak Chase Monitoring Program includes periodic monitoring of the spent fuel pool, the fuel transfer pit, and the cask pit liners leak chase system. Periodic monitoring of leakage from the leak chase system permits early determination and localization of any leakage.

NUREG-1801 Consistency

The Leak Chase Monitoring Program is an existing plant-specific program for Davis-Besse. There is no corresponding aging management program described in NUREG-1801. The program is evaluated against the 10 elements described in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The Leak Chase Monitoring Program, which includes periodic monitoring of the spent fuel pool, the fuel transfer pit, and the cask pit liners leak chase system, is credited for detecting loss of material aging effects for the spent fuel pool, the fuel transfer pit, and the cask pit liners.

  The Leak Chase Monitoring Program monitors the spent fuel pool, the fuel transfer pit, and the cask pit liners for leakage using the floor and wall monitoring system. Each weld made on the stainless steel wall panels is backed by a channel, and a group of these channels is piped to a common zone drain. The floor welds are backed by a trench in the concrete and, like the wall channels, are grouped together to a common zone drain.

- Preventive Actions
  No actions are taken as part of the Leak Chase Monitoring Program to prevent aging effects or mitigate age-related degradation.

- Parameters Monitored or Inspected
  The spent fuel pool, the fuel transfer pit, and the cask pit liner leak detection drain valves are periodically opened, any leakage is collected, and the amounts are recorded. In addition, leak rates for zone valves are calculated by the volumetric method and recorded.

- Detection of Aging Effects
  The Leak Chase Monitoring Program includes activities to cycle open and close the spent fuel pool, the fuel transfer pit, and the cask pit liner drain valves on a monthly basis. Each valve on the drain line capable of being cycled is opened to allow any water that accumulated in the lines to drain into an open funnel. After a prescribed wait time, leakage is collected. The amount collected and the calculated leak rate are recorded for each of the 21 drain zones. If leakage collected from any zone drain valve is greater than 10 milliliters, then the sample is appropriately labeled and transported to a laboratory for boron analysis. Collected leakage information and boron analysis results are recorded in the work order system. Monitoring of leakage from the leak chase system permits early determination and localization of any leakage.

- Monitoring and Trending
  The Leak Chase Monitoring Program leak detection activities are performed monthly. This routine task requires recording of the leakage amount collected and the calculated leak rate. In addition, if leakage collected from any zone drain valve is greater than 10 milliliters, then the sample is analyzed for boron concentration and the results are also recorded. Leak chase channel results are reviewed by the spent fuel pool system engineer. Adverse conditions are documented in the Corrective Action Program and summarized in system health reports.

- Acceptance Criteria
  Adverse trends (continued increases of leak rates on a particular zone valve) are documented in the Corrective Action Program.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  The Leak Chase Monitoring Program operating experience has indicated minor leakage from the spent fuel pool, the fuel transfer pit, and the cask pit liners.

  The Spent Fuel Pool System health report (second quarter 2009) shows the system has a "Green" status, highest ranking available, for overall system performance. One of the leak chase drains consistently showed small amounts of leakage during the monthly test, as documented in the third quarter of 2008 health report. Two other leak chase drains showed occasional leakage during this test. The leaks were small and the fluid was captured by the leak collection system. The boron concentration appeared erratic in one sample during the third quarter of 2008. The Corrective Action Program was used to document the condition, but since the leak collection boron concentration is an information-only test, this condition was documented for trending purposes.

  Information Notice 2004-05, "Spent Fuel Pool Leakage to Onsite Groundwater," was evaluated in the Corrective Action Program as it relates to Davis-Besse. The investigation summary provided some historic operating experience on the Leak Chase Monitoring Program. Review of the results of the leak detection testing is performed by the spent fuel pool system engineer. Leakage outside the leak chase drains has been seen in several places over the years. The most extensive visible evidence of leakage was on the wall and ceiling of ECCS Pump Room No. 1 over the period from 2000 to 2001. This leakage was stopped and the area cleaned. Based on the evaluations associated with this past leak, there are no concerns regarding the strength or integrity of the concrete structure associated with these leaks. During the re-racking of the spent fuel pool during Cycle 13, underwater divers used a vacuum box on the weld seams in the spent fuel pool to determine if there were any detectable leaks; none could be located. At the time that there was visible evidence of leakage in ECCS Pump Room No. 1, little leakage was being seen in the leak chases. Additional action was taken by FENOC to open and verify open the 21 leak chase valves and piping in February 2001. It found six of the chases to be totally blocked. A significant amount of trapped fluid was found in several of the blocked leak chases. As a result of the valves found clogged, the normal position of the leak chase valves was changed from open to closed to reduce the likelihood of the boric acid solidifying and blocking the valves and piping. The leak collection isolation valves were cleaned and un-clogged.

The Corrective Action Program documented 140 milliliters of leakage collected during July 2008 for one zone valve. The leakage rate was calculated as 2.8 milliliters per minute, which was higher than the trend data average of 1.0 milliliters per minute over the previous twelve months. Based on a review of the trend data collected since 1999, occasional spikes in flow rate do occur. The Corrective Action Program item was designated for tracking and trending of a condition that occurs periodically in the plant.

Enhancements

None.

Conclusion

The Leak Chase Monitoring Program provides reasonable assurance that potentially detrimental aging effects will continue to be adequately managed such that evidence of leakage from the spent fuel pool, the fuel transfer pit, and the cask pit liners is promptly identified and the pool liner's intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

B.2.26 LUBRICATING OIL ANALYSIS PROGRAM

Program Description

The Lubricating Oil Analysis Program mitigates the effects of aging for plant components that are within the scope of license renewal and that are exposed to a lubricating oil environment. The program includes requirements to ensure the oil environment in the mechanical systems is maintained to the required quality (i.e., it maintains contaminants [water and particulates] within acceptable limits). The program requires management of the relevant conditions that could lead to the onset and propagation of loss of material due to crevice, galvanic, general, or pitting corrosion, or reduction in heat transfer due to fouling, through monitoring of the lubricating oil consistent with various manufacturers' recommendations and industry standards. The relevant parameters that are monitored, including particulate and water content, viscosity, and, under certain conditions, neutralization number and flash point, are indicative of conditions that could lead to age-related degradation of susceptible materials. The Lubricating Oil Analysis Program is a mitigation program.

The Lubricating Oil Analysis Program is supplemented by a one-time inspection of representative areas of lubricating oil systems under the One-Time Inspection to provide confirmation that loss of material and reduction in heat transfer due to fouling are effectively mitigated.

NUREG-1801 Consistency

The Lubricating Oil Analysis Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M39, "Lubricating Oil Analysis."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Lubricating Oil Analysis Program is an ongoing program that effectively incorporates the best practices of the industry. Expert recommendations and industry standards are used to establish quality requirements for lubricating oil. The program incorporates the results of operating experience from Davis-Besse and the industry to optimize testing parameters, sampling frequencies, acceptance criteria, and alarm levels, as required by the FENOC Condition Monitoring Program. The program has been, and continues to be, subject to periodic internal and external performance assessment to identify strengths and areas for improvement.

For example, a self-assessment of the Lubricating Oil Analysis Program was conducted in early 2004. The overall assessment determined that the program was effective in implementing its stated goals. The assessment identified several areas for improvement, including enhancing procedures, consolidation of lubricating oils, addition of oil reservoir breathers and vents in certain locations, addition of sampling ports, and additional training. The FENOC Corrective Action Program was used to address the areas identified for improvement in the assessment.

Review of Davis-Besse operating experience did not reveal a loss of component intended function for components exposed to lubricating oil that could be attributed to an inadequacy of the Lubricating Oil Analysis Program. Abnormal lubricating oil conditions are promptly identified, evaluated, and corrected.

Conclusion

The Lubricating Oil Analysis Program, in conjunction with the One-Time Inspection, has been demonstrated to be capable of managing loss of material and reduction in heat transfer in lubricating oil, for susceptible components, through monitoring of the relevant parameters. The Lubricating Oil Analysis Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.27 MASONRY WALL INSPECTION

Program Description

The Masonry Wall Inspection is implemented as part of the Structures Monitoring Program, conducted for the Maintenance Rule.

The Masonry Wall Inspection is an existing condition monitoring program consisting of inspection activities to detect aging and age-related degradation for masonry walls identified as performing intended functions in accordance with 10 CFR 54.4. Masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program.

NUREG-1801 Consistency

The Masonry Wall Inspection is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.S5, "Masonry Wall Program."

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Scope**

  The Masonry Wall Inspection, included in the Structures Monitoring Program, will include and list the structures within the scope of license renewal that credit the Masonry Wall Inspection for aging management.

- **Monitoring and Trending**

  The Masonry Wall Inspection, included in the Structures Monitoring Program, will follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management.

- **Acceptance Criteria**

  The Masonry Wall Inspection, included in the Structures Monitoring Program, will specify that for each masonry wall, the extent of observed masonry cracking or degradation of steel edge supports or bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking or steel degradation is sufficient to invalidate the evaluation basis. An option is to develop a new evaluation basis that accounts for the degraded condition of the wall (i.e., acceptance by further evaluation).

Operating Experience

The Masonry Wall Inspection has been effective in managing age-related degradation. Periodic visual inspections conducted by the Masonry Wall Inspection have identified age-related findings. Specifically, inspections have found minor degradation including cracks in mortar joints, construction joint voids, abandoned bolts, and unfilled drilled holes which did not require further evaluation. Acceptable minor degradation has been noted on Maintenance Rule Evaluation reports and were reviewed and re-inspected during subsequent inspections. Inspected masonry walls are acceptable and are capable of performing their design functions with no design basis violations.

Review of completed Maintenance Rule Evaluation documentation indicated age-related degradation was identified and documented. Degradation requiring repair was addressed through the work order system. Examples of conditions found were:

- Auxiliary Building Rooms 117A and 301 have minor cracking less than 1/16 inch at masonry wall to concrete interface. Auxiliary Building Rooms 122 and 509 have minor cracking less than 1/16 inch at masonry wall to concrete interface above doorway opening. Auxiliary Building Room 318 has minor cracking less than 1/16 inch and chipping on masonry walls. Auxiliary Building Room 512 had two areas of spalling in the west wall that appeared to be caused by removal of anchor bolts. Conditions were judged acceptable.
- Auxiliary Building Rooms 115, 212, 234, 240, 310, 314CC, 318, 319, 320, 320A, 321A, 419, 422A, 428A, 502, 505, 507, 508, 510, 511, and 513 have various unfilled holes or abandoned anchors observed, all of which were determined to have no structural impact.
- Auxiliary Building Rooms 112, 304, and 504 have minor cracking and spalling on the masonry wall above doorway. The area has been repaired in the past. Inspections have not found conditions where degradation penetrated through the wall. The condition was judged acceptable and rework notice was issued.
- Auxiliary Building Rooms 312, 502, 503, 505, 508, 510, 511, 512, and 603 have minor cracking less than 1/16 inch at mortar joints. The condition was judged acceptable.
- Auxiliary Building Room 404 has a small void in block joint adjacent and north of door frame. Inspections have not found conditions where degradation penetrated through the wall. The condition was judged acceptable.

- Office Building condensate storage tank area Room 345 has various unfilled holes observed, all of which were determined to have no structural impact.
- The Relay House's basement south wall has a vertical crack at the location where a future doorway is intended. The future doorway is filled in with masonry block units and the crack is located at the interface between the concrete wall and the masonry block. The work order system was used to request correction of this issue.
- Turbine Building Room 247 has minor spalling on masonry wall corner. Turbine Building Rooms 334, 335, 336, 347, 431, 432, 517, 517A, and 517B have minor cracking observed. Turbine Building Rooms 328, 335, 336, 337, 339, 347, 431, 431A, 432, 517, 517A, and 517B have had various unfilled holes observed. The conditions were judged acceptable.
- Turbine Building Room 330 has both masonry wall vertical joints that butt up to the Turbine Building degraded which required rework. The work order system was used to address this issue.
- Water Treatment Building Room 11 has diagonal crack from top corner of door to lower corner of ventilation register. Water Treatment Building Room 12A has abandoned anchors in masonry wall observed. The conditions were judged acceptable.

The Corrective Action Program and ongoing review of industry operating experience will be used to ensure that the program continues to be effective in managing the identified aging effects.

Conclusion

The Masonry Wall Inspection, with enhancement, will be capable of detecting and managing aging effects for masonry walls within the scope of license renewal. The continued implementation of the Masonry Wall Inspection, with enhancement, provides reasonable assurance that the effects of aging will be managed so that components within the scope of this inspection will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.28 NICKEL-ALLOY MANAGEMENT PROGRAM

Program Description

The Nickel-Alloy Management Program manages primary water stress corrosion cracking (PWSCC) and stress corrosion cracking / intergranular attack (SCC/IGA) for nickel-alloy pressure boundary components, other than reactor vessel closure head nozzles and steam generator tubes, exposed to reactor coolant. The Nickel-Alloy Management Program is a combination mitigative and condition monitoring program.

Mitigative actions include replacement of Alloy 600/82/182 components with materials known to be less susceptible to PWSCC and SCC/IGA or repair of those components through weld overlay, weld inlay (also known as weld underlay), mechanical stress improvement process or surface conditioning. The condition monitoring portion of the program uses a number of inspection techniques to detect cracking, including volumetric and bare metal visual examinations. The Nickel-Alloy Management Program implements the inspection of components through the Inservice Inspection Program. The program implements component evaluations, examination methods, scheduling, and site documentation as required for compliance with 10 CFR 50, the ASME Code, NRC bulletins, NRC generic letters, and NRC staff-accepted industry guidelines related to nickel-alloy issues. The Nickel-Alloy Management Program includes mitigation and repair activities and strategies to ensure long-term operability of nickel-alloy components.

NUREG-1801 Consistency

The Nickel Alloy Management Program is an existing plant-specific program for Davis-Besse. As NUREG-1801 Section XI.M11, "Nickel-Alloy Nozzles and Penetrations," does not contain program elements, the Nickel-Alloy Management Program is evaluated against the 10 elements described in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The Nickel-Alloy Management Program is credited with managing cracking due to PWSCC and SCC/IGA for nickel-alloy pressure boundary components in the reactor vessel, pressurizer, steam generator, and reactor coolant (hot and cold leg) piping.

  The Nickel-Alloy Management Program scope does not include nickel-alloy steam generator tubes (included in the Steam Generator Tube Integrity Program) or nickel-alloy reactor vessel closure head nozzles (included in the Nickel-Alloy Reactor

Vessel Closure Head Nozzles Program). The Nickel-Alloy Management Program scope also does not include non-pressure boundary, nickel-alloy reactor vessel internals components (included in the PWR Reactor Vessel Internals Program).

The Nickel-Alloy Management Program is credited for aging management in conjunction with the PWR Water Chemistry Program and the Inservice Inspection Program.

- Preventive Actions
  The Nickel-Alloy Management Program includes mitigation activities and strategies to ensure the long-term operability of nickel-alloy components. Some of the currently available mitigation techniques include a mechanical stress improvement process or surface conditioning, weld overlay, weld inlay, and replacement of Alloy 600/82/182 materials with materials known to be less susceptible to PWSCC. The program lists the mitigation strategies that are available and provides considerations for selection and implementing a mitigation strategy.

- Parameters Monitored or Inspected
  The parameters inspected by the Nickel-Alloy Management Program include cracks (flaws) in nickel-alloy components that are exposed to reactor coolant. The program maintains a comprehensive list of the components in the plant that are constructed of nickel-alloy materials susceptible to cracking and subjected to the reactor coolant environment. The effects of PWSCC on these components are either mitigated by the program's strategies, based on susceptibility and other considerations, or the components are inspected on a frequency established by the program that is consistent with industry guidelines.

  Nickel-alloy components are inspected in accordance with the Inservice Inspection plan. The Nickel-Alloy Management Program uses a number of inspection techniques to detect cracking due to PWSCC or SCC/IGA. The techniques include volumetric and bare metal visual examinations. The schedule for the examinations is described in the program plan and in the Inservice Inspection plan.

- Detection of Aging Effects
  The Nickel-Alloy Management Program uses a number of inspection techniques to detect cracking due to PWSCC and SCC/IGA, including volumetric examinations and bare metal visual examinations. Bare metal visual examinations are similar to visual (VT-2) examinations but require removal of insulation to allow direct access to the metal surface. The nickel-alloy components have been ranked based on susceptibility (in accordance with EPRI Materials Reliability Program (MRP) guidelines).

Detection of cracking due to PWSCC or SCC/IGA ensures that nickel-alloy components meet required design attributes and maintain their availability to perform their intended functions.

The Nickel-Alloy Management Program is based on ASME Code requirements and on the recommendations of NEI and the EPRI MRP. Industry experience and research has resulted in recommended techniques and frequencies for inspection to detect cracking prior to component failure. Inspection population and sample size are in accordance with ASME Code requirements and MRP guidelines. Data collection (e.g., inspection reports) is incorporated in the program.

- Monitoring and Trending
  Monitoring and trending activities for detection and sizing of cracks in nickel-alloy pressure boundary components are part of the Nickel-Alloy Management Program. The program ranks the nickel-alloy components for inspection based on susceptibility to cracking in accordance with MRP guidelines.

  Davis-Besse uses the guidelines in ASME Section XI Table IWB-2500-1, Code Case N-722, and EPRI MRP-139 Revision 1, "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline," for inspection (examination) techniques and frequencies. Flaws found during the inspections are immediately evaluated against criteria contained in ASME Section XI IWB-3000 to predict the extent of degradation and implement timely corrective or mitigative actions. Disposition by analysis is permitted by IWB-3000. Contingencies for repairs, replacement, or mitigative actions such as weld overlays are evaluated prior to each inspection outage. Monitoring of industry operating experience is performed to incorporate any required changes to the Nickel-Alloy Management Program as a result of industry experience.

- Acceptance Criteria
  The Nickel-Alloy Management Program tracks and trends cracking (flaws) in the nickel-alloy components within the scope of the program. The nickel alloy components within the scope of the program are evaluated against the acceptance criteria contained in ASME Section XI. Based on the evaluations, the flaw is accepted by either a repair or replacement activity or by analytical evaluation prior to start-up.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Recent Davis-Besse operating experience related to inspection of Alloy 600 components is documented in inservice inspection outage summary reports. Specific examples of findings are also documented in the Corrective Action Program. Periodic health reports and self-assessment reports are also issued for the Nickel-Alloy Management Program.

  The Corrective Action Program and an ongoing review of industry operating experience will be used to ensure that the program remains effective in managing the identified aging effects.

  The following examples of operating experience provide objective evidence that the Nickel-Alloy Management Program is effective in ensuring that intended functions will be maintained consistent with the current licensing basis for the period of extended operation:

  - In September of 2008, NRC inspectors conducted a review of Davis-Besse's activities regarding dissimilar metal butt weld mitigation and inspection implemented in accordance with the industry self-imposed mandatory requirements of MRP-139. The inspectors verified that the program included baseline inspections, that the baseline inspections of pressurizer locations had been completed, and that the schedule for other baseline inspections was consistent with MRP-139. The inspectors also reviewed the volumetric examinations of the high pressure injection safe end to nozzle weld and decay heat 12 inch branch connection to elbow overlay weld that were completed during the previous outage. The weld was performed in accordance with MRP-139 and the weld overlay was performed in accordance with the NRC staff-approved relief request. The welding was performed by qualified personnel and any deficiencies identified were appropriately dispositioned and resolved. As of the September 2008 date of NRC integrated inspection 50/3462008-004, seven penetrations had been mitigated by structural weld overlay and had received volumetric examinations, with further mitigation or replacement planned for the remaining susceptible welds.

- Periodic self-assessments are performed as part of the program. The most recent self-assessment, performed in September of 2008 in preparation for the NRC integrated inspection, evaluated the degree of compliance with the requirements of EPRI MRP-139 and assessed the program with respect to inspection requirements for dissimilar metal butt welds. The self-assessment noted the quality and depth of site-specific information presented to the industry as a strength of the program. The self-assessment concluded that program has adequately implemented the requirements of MRP-139 to date, and that the existing schedule for inspection or mitigation of the remaining locations would ensure compliance with the MRP-139 implementation date. Some minor discrepancies and improvements were noted during the self-assessment that have been addressed through the Corrective Action Program.

- In January of 2008, a leak in an existing weld was noted in the decay heat (and low pressure injection) nozzle during performance of the mitigative structural weld overlay repair. The structural weld overlay was completed. The extent of condition review assessed other susceptible Alloy 600 material locations associated with the pressurizer and hot leg to ensure adequate inspection or mitigation was performed.

  An ultrasonic examination of the hot leg to decay heat nozzle structural weld overlay was successfully completed, with the weld overlay establishing a surface that facilitates adequate ultrasonic examination. The Inservice Inspection program will periodically perform additional ultrasonic examinations to ensure that the flaw remains contained within the dissimilar metal butt weld. An operating experience report was developed.

Based on review of plant-specific and industry operating experience, cracking due to PWSCC or SCC/IGA of nickel-alloy components exposed to reactor coolant will be adequately managed so that intended function of the nickel-alloy (and nickel-alloy clad) components will be maintained for the period of extended operation.

Enhancements

None.

Conclusion

The Nickel-Alloy Management Program, in conjunction with the PWR Water Chemistry Program and Inservice Inspection Program, provides reasonable assurance that cracking due to PWSCC and SCC/IGA will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.29 NICKEL-ALLOY REACTOR VESSEL CLOSURE HEAD NOZZLES PROGRAM

Program Description

The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program manages cracking of the nickel-alloy control rod drive nozzles and welds in the reactor vessel closure head. The Boric Acid Corrosion Program is credited for managing wastage of associated reactor vessel closure head surfaces. The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program is a condition monitoring program.

The program ensures that inservice inspections of all nickel-alloy reactor vessel closure head penetration nozzles, and associated reactor vessel closure head surfaces, will continue to be performed in accordance with ASME Code Case N-729-1, as modified by 10 CFR 50.55a Section (g)(6)(ii)(D).

NUREG-1801 Consistency

The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M11A, "Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (PWRs only)."

NUREG-1801 program XI.M11A is based on NRC First Revised Order EA-03-009. However, since the publication of NUREG-1801, Order EA-03-009 has been withdrawn and replaced by ASME Code Case N-729-1. 10 CFR 50.55a requires that all licensees of pressurized water reactors shall augment their Inservice Inspection program with ASME Code Case N-729-1 subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(D). The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program complies with 10 CFR 50.55a(g)(6)(ii)(D).

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program detects aging effects using nondestructive examination visual and surface or volumetric techniques to detect and characterize flaws and reactor vessel closure head surface wastage. These techniques are widely used and have been demonstrated effective at detecting degradation due to PWSCC.

In March 2002, significant degradation of the original Davis-Besse reactor vessel closure head was discovered. Performance deficiencies in the implementation of the boric acid corrosion control program and Corrective Action Program allowed the reactor coolant system pressure boundary leakage to occur undetected for a prolonged period of time resulting in the head degradation. Program compliance reviews were performed to ensure proper interface with supporting plant programs, proper consideration of industry experience, proper staffing, and timely resolution of identified weaknesses. Detailed reviews were performed to ensure the programs were conducted in accordance with the governing processes. The original reactor vessel closure head was replaced in 2002.

In March 2010, ultrasonic examinations of the control rod drive mechanism nozzles identified flaws on multiple nozzles. Active leakage was identified on one nozzle. The direct cause was Primary Water Stress Corrosion Cracking. The reactor vessel closure head had been in operation approximately six years. An inside diameter temper bead half-nozzle weld repair was utilized. Post-repair inspections were completed with acceptable results. As provided in Confirmatory Action Letter, Number 3-10-001, Mark A. Satorius (NRC) to Barry S. Allen (FENOC), dated 6-23-2010, FENOC has voluntarily committed to shutdown the Davis-Besse plant no later than October 1, 2011, and replace the reactor pressure vessel head with one manufactured using materials resistant to PWSCC.

The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program has been developed based on relevant plant and industry operating experience. The Corrective Action Program and an ongoing review of industry operating experience ensure that the program is effective in managing the identified aging effects.

Conclusion

The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program manages cracking of the nickel alloy control rod drive nozzles and welds in the reactor vessel closure head and loss of material of the associated reactor vessel closure head surfaces. The Nickel-Alloy Reactor Vessel Closure Head Nozzles Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.30 ONE-TIME INSPECTION

Program Description

One-Time Inspection is a new activity that will be implemented prior to the period of extended operation.

The activity will require one-time inspections to verify the effectiveness of the Closed Cooling Water Chemistry Program, the Fuel Oil Chemistry Program, the Lubricating Oil Analysis Program, and the PWR Water Chemistry Program. One-time inspections are used to address situations where: 1) an aging effect is not expected to occur, but there is insufficient data to completely rule it out, 2) an aging effect is expected to progress very slowly in the specified environment, and the local environment may be more adverse, or 3) the characteristics of the aging effect include a long incubation period.

One-Time Inspection will provide assurance that aging which has not yet manifested itself is indeed not occurring, or that the age-related degradation is so insignificant that an aging management program is not warranted. If evidence of age-related degradation is revealed by a one-time inspection, the routine evaluation of the inspection results will trigger corrective actions to ensure the intended function of the affected components is maintained through the period of extended operation.

The elements of the one-time inspections will include:

- Determination of a representative sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience;
- Identification of the inspection locations in the system or component based on the aging effect, or based on the areas susceptible to concentration of contaminants that promote certain aging effects;
- Determination of the examination technique, including acceptance criteria that would be effective in identifying the aging effects for which the component is examined; and
- Evaluation of the need for follow-up examinations to monitor the progression of identified age-related degradation.

NUREG-1801 Consistency

One-Time Inspection is a new Davis-Besse activity that will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801 Section XI.M32, "One-Time Inspection."

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements, which are plant-specific and in addition to the NUREG-1801, Section XI.M32 elements, will be implemented in the identified program elements prior to the period of extended operation.

- **Scope**

  One-Time Inspection will include visual inspections to detect and characterize the material condition of aluminum, copper alloy (including copper alloy > 15% zinc), stainless steel, and steel (including gray cast iron) components exposed to condensation or diesel exhaust. The one-time inspections will provide direct evidence as to whether, and to what extent, cracking, loss of material, or reduction in heat transfer has occurred. Materials in these environments are either plant-specific and not addressed by another aging management program, or a plant-specific program is identified in NUREG-1801.

- **Scope, Parameters Monitored/Inspected, Detection of Aging Effects**

  One-Time Inspection will include visual and physical examination, such as manipulation and prodding, of elastomers (flexible connections). This visual and physical examination will supplement the External Surfaces Monitoring Program and provide direct evidence as to whether, and to what extent, hardening and loss of strength due to thermal exposure, ultraviolet exposure, and ionizing radiation of elastomers has occurred. This enhancement is in response to recent NRC concerns (raised during license renewal audits) that visual examination may not be adequate to identify hardening and loss of strength for elastomers prior to a loss of function.

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope

  One-Time Inspection will require one-time inspections to verify the effectiveness of mitigation aging management programs; to confirm that age-related degradation is not occurring, is insignificant, or is occurring slowly such that component intended function will be maintained through the period of extended operation.

  One-time inspections are required to verify the effectiveness of the Closed Cooling Water Chemistry Program, Fuel Oil Chemistry Program, Lubricating Oil Analysis

Program, and the PWR Water Chemistry Program for managing loss of material, cracking, or reduction in heat transfer in the closed cooling water, treated water, fuel oil, and lubricating oil environments.

The one-time inspections will also provide assurance that:

- Aging effects are not occurring for susceptible materials in environments where degradation is not expected but cannot be ruled out based on available data.
- Aging effects are not occurring, or are progressing very slowly in a specified environment, as well as where the local environment may be more adverse than the bulk environment, or the characteristics of the aging effect include a long incubation period.

The activity will include visual and physical (manipulation or prodding) examination of elastomers (flexible connections) in various environments for evidence of hardening or loss of strength due to thermal exposure, ultraviolet exposure, or ionizing radiation.

In addition, one-time inspections will characterize the material condition of susceptible materials exposed to the "Condensation" and "Diesel Exhaust" environments, which are not addressed by other aging management programs, to verify that unacceptable degradation is not occurring or to trigger additional actions that will assure the intended function of affected components will be maintained through the period of extended operation.

Furthermore, the one-time inspections will include UT exams of the internal bottom surfaces of a sample of fuel oil tanks to ensure that significant degradation is not occurring.

- Preventive Actions
  One-Time Inspection is a condition monitoring activity that will consist of inspections independent of methods to mitigate or prevent degradation. The activity does not include any preventive actions.

- Parameters Monitored or Inspected
  One-Time Inspection will require inspections to be performed by qualified personnel following procedures consistent with the requirements of the ASME Code and 10 CFR 50, Appendix B. Inspections will be performed using a variety of nondestructive examination methods, including visual, volumetric, and surface inspection techniques.

  The activity will inspect parameters directly related to degradation of the metallic components under review such as wall thickness, visual evidence of corrosion, or evidence of fouling. The parameters to be inspected for elastomers include visual evidence of surface degradation, such as cracking or discoloration, as well as hardening and loss of strength identified through manipulation or prodding.

- Detection of Aging Effects
  A representative sample of the system and component population will be inspected using a variety of nondestructive examination methods, including visual inspection, volumetric inspection, and surface inspection techniques. The sample population will be determined by engineering evaluation, and where practical, will be focused on the (bounding or lead) components considered most susceptible to aging degradation due to time in service, the severity of the operating conditions, and the lowest design margin.

  The inspections will be completed with sufficient time to ensure that the aging effects which may impact component intended functions early in the period of extended operation will be appropriately managed. At the same time, the inspections will be timed to allow the components to attain sufficient age to ensure that aging effects with long incubation periods can be identified.

  For elastomers (flexible connections), established visual examination techniques, as well as physical manipulation or prodding, will be performed by qualified personnel on a sample population of subject components to identify evidence of hardening and loss of strength (change in material properties). The sample population will be determined by engineering evaluation and, where practical, focused on the (bounding or lead) components considered most susceptible to aging degradation due to time in service, the severity of the operating conditions, and the lowest design margin.

- Monitoring and Trending
  The inspection sample size will be determined based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience. Inspection findings will be evaluated by assigned engineering personnel. Inspection findings not meeting the acceptance criteria will be evaluated and tracked through the Corrective Action Program. The Corrective Action Program will be used to identify the corrective actions including additional inspections or expansion of the inspection sample size.

- Acceptance Criteria
  Indications or relevant conditions of degradation detected during the one-time inspections will be compared to pre-determined acceptance criteria, such as design minimum wall thickness for piping. Inspection findings will be evaluated by assigned engineering personnel. If the acceptance criteria are not met, then the indications or conditions will be evaluated under the Corrective Action Program to determine

whether they could result in a loss of component intended function during the period of extended operation.

Determination of acceptance criteria will include evaluation of design standards and industry codes or standards, as applicable. Unacceptable inspection findings will include evidence of cracking, loss of material, loss of material flexibility, hardening or loss of strength, or reduction in heat transfer (fouling) that could lead to loss of intended function during the period of extended operation.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Operating experience for select components and environments within the scope of One-Time Inspection was evaluated to ensure use of a one-time inspection was appropriate.

  For example, in 2003, because of chronic rust and particulate accumulation in the diesel air start compressor and filter components, a modification was implemented for the EDG air start system. The modification replaced carbon steel piping and components with stainless steel and added air filters, air dryers, and moisture separators, etc to mitigate rust particulates and moisture effects in the EDG air start subsystem. A similar modification was implemented for the station blackout diesel generator (SBODG) air start system. Review of Davis-Besse operating experience subsequent to these modifications did not identify any aging effects that were attributed to excessive moisture in the compressed air downstream of EDG dryers or SBODG dryer-filters.

  Some corrosion caused by moisture accumulation in Station Air components with a moisture removal function (e.g., aftercooler separator drain trap) has been documented. Corrective action included removing the moisture and rust, and confirming proper trap (automatic drain) operation, but did not result in component replacement or establishment of actions to prevent recurrence.

In 2004, industry operating experience regarding corrosion of refrigeration lines due to condensation forming on cold carbon steel piping surfaces was evaluated for applicability at Davis-Besse. Units were evaluated, including some that are in the scope of One-Time Inspection, and it was determined that copper piping and tubing was not subject to the identified corrosion. Expected surface rust was also identified on many components in Davis-Besse refrigeration systems through walkdown. It was concluded that the concern raised by the OE is not an issue for Davis-Besse.

The elements that comprise the one-time inspections are consistent with industry practice.

Industry and plant-specific operating experience will be considered in the development and implementation of this activity. As additional operating experience is obtained, lessons learned will be incorporated, as appropriate.

Conclusion

Implementation of One-Time Inspection will provide reasonable assurance that the aging effects will be managed so that components within the scope of this inspection will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.31 OPEN-CYCLE COOLING WATER PROGRAM

Program Description

The Open-Cycle Cooling Water Program manages loss of material due to crevice, galvanic, general, pitting, and microbiologically-influenced corrosion (MIC), and also due to erosion for components located in the Service Water System, and for components connected to or cooled by the Service Water System, and also in the Circulating Water System. The program manages fouling due to particulates (e.g., corrosion products) and biological material (micro- and macro-organisms) resulting in reduction in heat transfer for heat exchangers within the scope of the program. In addition, the program manages cracking for copper alloy greater than 15% zinc components that are cooled by the Service Water System.

The Open-Cycle Cooling Water Program consists of inspections, surveillances, and testing to detect and evaluate fouling, loss of material, and cracking, combined with chemical treatments and cleaning activities to minimize fouling, loss of material, and cracking. The existing program is a combination condition and performance monitoring and mitigation program that implements the recommendations of Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

NUREG-1801 Consistency

The Open-Cycle Cooling Water Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801 Section XI.M20, "Open-Cycle Cooling Water System," with the following exceptions.

Exceptions to NUREG-1801

Program Elements Affected:

- **Monitoring and Trending**

  NUREG-1801 states that testing and inspections are done annually and during refueling outages. Inspection frequencies for the Open-Cycle Cooling Water Program are based on operating conditions and past history; flow rates, water quality, lay-up, and heat exchanger design, in accordance with Generic Letter 89-13. In the supplemental response to Generic Letter 89-13, Davis-Besse committed to annual heat exchanger inspections for the first three cycles following implementation of Generic Letter 89-13, with the option of then determining the best testing frequency based on past history.

Enhancements

None.

Operating Experience

The Open-Cycle Cooling Water Program for Davis-Besse is an ongoing program that has implemented the recommended actions of Generic Letter 89-13 and has justified any alternatives to those recommendations. The health of the program and corresponding systems are periodically reported, including chemistry trends and material conditions. Industry operating experience is evaluated for impact to Davis-Besse, and periodic self assessments are conducted. As a result, Davis-Besse has programs in place with operating experience to demonstrate that the effects of aging on the Service Water System, and on the safety-related heat exchangers that are served, will be effectively managed during the period of extended operation.

Annual ultimate heat sink performance, as well as related Generic Letter 89-13 systems, components, and controls, is a subject of NRC integrated inspection. In recent years, reviews were performed by NRC inspectors to verify the acceptability of test methods and conditions, acceptance criteria, use of instrument uncertainties, frequency of testing, biofouling controls, compliance with design parameters, and the extrapolation of test data to design conditions. No findings of significance with respect to the effectiveness of the existing program were identified during these integrated inspections. The Open-Cycle Cooling Water Program satisfies Generic Letter 89-13 commitments for managing aging effects due to biofouling, corrosion, protective coating failures, and silting within the various system components.

The program has identified cases (in 2008 and 2007) where ultrasonic thickness measurements of service water piping identified segments that were below procedural limits. The piping segments were evaluated and the reduced wall thicknesses were determined to exceed the minimum operable values and code stress allowable values. In addition, the program has been effective in identifying biofouling through the regular measurements of flow rate and differential pressure – in 2009, an emergency core cooling system room cooler was identified as possibly having marginal biofouling due to an increased differential pressure. The problem ultimately was found to be corrosion in nearby supply and return piping. The coolers are regularly checked for biofouling. In 2008, a thick layer of silt was identified in the service water piping between two system valves related to an auxiliary feedwater train which was undergoing maintenance activities. The affected piping was cleaned with a hydrolazer and drained. Additional cleaning was performed when silt accumulation was found remaining in the piping. The cause was determined to be low flow and stagnant water in the auxiliary feedwater supply piping (the silt decanted from the service water flowing past the auxiliary feedwater piping).

Conclusion

The Open-Cycle Cooling Water Program has been demonstrated to be capable of detecting and managing loss of material, cracking, and reduction in heat transfer for susceptible components in raw water environments. The Open-Cycle Cooling Water Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.32 PWR REACTOR VESSEL INTERNALS PROGRAM

Program Description

The PWR Reactor Vessel Internals Program is a new plant-specific program that will manage change in dimension due to void swelling; cracking due to flaw initiation and growth, SCC/IGA, and irradiation-assisted stress corrosion cracking (IASCC); loss of preload due to stress relaxation; reduction in fracture toughness due to radiation and thermal embrittlement; and loss of material due to wear, for reactor vessel internals components. The PWR Reactor Vessel Internals Program is a condition monitoring program.

The PWR Reactor Vessel Internals Program is based upon the examination requirements for Babcock & Wilcox (B&W) designed pressurized water reactors (PWRs) provided in EPRI Topical Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," along with the implementation guidance described in NEI 03-08, "Guideline for the Management of Materials Issues." MRP-227 has been submitted to the NRC for review and approval. Following NRC approval, MRP-227 will be revised to incorporate any necessary changes to the guidelines and reissued as MRP-227-A. The Davis-Besse PWR Reactor Vessel Internals Program will be revised, as necessary, to incorporate the final recommendations and requirements as published in MRP-227-A.

The EPRI inspection and evaluation guidelines establish the augmented ASME Section XI inservice inspection requirements that will be used to monitor for the aging effects that are applicable to certain susceptible or limiting reactor vessel internals components for B&W designed PWRs.

NUREG-1801 Consistency

The PWR Reactor Vessel Internals Program is a new plant-specific program for Davis-Besse. There is no corresponding aging management program described in NUREG-1801. The program is evaluated against the 10 elements described in Appendix A.1, Section A.1.2.3 of NUREG-1800, the Standard Review Plan for License Renewal (SRP-LR).

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The PWR Reactor Vessel Internals Program is credited with managing change in dimension due to void swelling; cracking due to flaw initiation and growth, SCC/IGA, and IASCC; loss of preload due to stress relaxation; reduction in fracture toughness

due to radiation and thermal embrittlement; and loss of material due to wear, for reactor vessel internals components. The program scope does not include consumable items such as fuel assemblies, control rod assemblies, and incore instrumentation. The scope also does not include welded attachments to the reactor vessel.

The Davis-Besse reactor vessel internals consist of two basic assemblies, the plenum assembly that is removed during each refueling operation to obtain access to the fuel assemblies, and the core support assembly (CSA) that remains in place in the reactor vessel during refueling, and is removed only to perform scheduled inspections of the reactor vessel interior surfaces or of the core support assembly itself.

- Preventive Actions
  The PWR Reactor Vessel Internals Program is a condition monitoring program and does not include any preventive or mitigative actions.

- Parameters Monitored or Inspected
  The PWR Reactor Vessel Internals Program is credited with managing change in dimension due to void swelling; cracking due to flaw initiation and growth, SCC/IGA, and IASCC; loss of preload due to stress relaxation; reduction in fracture toughness due to radiation and thermal embrittlement; and loss of material due to wear, for the reactor vessel internals components.

  The program contains elements that monitor and inspect for the parameters that govern the progress of each of these aging effects. Section 4 of MRP-227 describes the methodologies that provide the monitoring and inspection of these aging effects. For B&W designed plants, the aging management methodologies include visual examinations, volumetric examinations, and physical measurements. The visual (VT-3) examinations detect the general degradation conditions and the volumetric examinations (ultrasonic testing) indicate the presence of discontinuities or flaws throughout the volume of material in the area of interest. Some aging effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements.

  In addition, as part of the Inservice Inspection Program, a visual (VT-3) examination of the reactor vessel removable core support structure is conducted once per Inservice Inspection interval in accordance with ASME Section XI, Table IWB-2500-1, Examination Category B-N-3.

- Detection of Aging Effects
  MRP-227 describes the examination requirements for the PWR vessel internals Primary and Expansion components for B&W designed plants. Primary components

are highly susceptible to the effects of at least one of the subject aging mechanisms. Expansion components are highly or moderately susceptible to the effects of at least one of the subject aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at Davis-Besse. The aging management methodologies described in MRP-227 are based on well-documented and well-demonstrated examination methods with which the industry has considerable experience. The aging management methodologies for the B&W designed plants include visual examinations, volumetric examinations, and physical measurements.

The examination requirements defined in MRP-227, as approved by the NRC, will be applied through use of EPRI Topical Report 1016609, "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)."

- Monitoring and Trending
  One-time, periodic, and conditional examinations and other aging management methodologies, scheduled in accordance with MRP-227 provide timely detection of aging effects. In addition to the Primary components, 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. Flaw indications detected during the required examinations are dispositioned in accordance with the Corrective Action Program.

- Acceptance Criteria
  Section 5 of MRP-227 provides the examination acceptance criteria for the Primary and Expansion components. Any detected condition that does not satisfy these examination acceptance criteria must be dispositioned. Example methodologies that can be used to analytically disposition unacceptable conditions are discussed or referenced in Section 6 of MRP-227. However, other demonstrated and verified alternatives to the Section 6 methodologies may be used.

  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.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. For this reason, 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. A benefit of this decision was the experience gained with the ultrasonic testing examination techniques used in the inspections. In addition, the industry began substantial laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations. Another item with existing or suspected material degradation concerns that has been identified for PWR components is cracking in some high-strength bolting. This condition has been corrected primarily through bolt replacement with less susceptible material and improved control of pre-load.

  Stress corrosion cracking has occurred in Alloy A-286 internals bolting in B&W units, including Davis-Besse. The Alloy A-286 bolt failures in B&W PWR internals were subjected to a comprehensive failure analysis that is documented in BAW-1843PA, "The B&W Owners Group Evaluation of Internal Bolting Concerns in 177FA Plants." BAW-1843PA was reviewed and approved by the NRC. This failure analysis addressed probable cause of the cracking, assessment of likelihood and consequences of joint failure, and replacement bolt design. The recommended replacement bolts are Alloy X-750 HTH bolts that are less susceptible to stress corrosion cracking and have overall excellent material properties.

  Davis-Besse has replaced the majority of the Alloy A-286 bolts for the reactor vessel internals (upper core barrel, lower core barrel, lower thermal shield, and surveillance specimen holder tubes) with Alloy X-750 HTH bolts. To satisfy a needed action under NEI 03-08 protocol, Davis-Besse performed ultrasonic testing examinations of 100% of all upper core barrel bolts during the Cycle 16 refueling outage (spring 2010). This inspection did not identify any unacceptable indications.

As part of the Inservice Inspection Program, a visual (VT-3) examination of the reactor vessel removable core support structure is conducted once per Inservice Inspection interval in accordance with ASME Section XI, Table IWB-2500-1, Examination Category B-N-3. These inspections have not identified any unacceptable indications.

FENOC participates in the industry programs for investigating and managing aging effects on reactor vessel internals. Through its participation in EPRI MRP activities, FENOC will continue to benefit from the reporting of reactor vessel internals inspection information, and will share its own internals inspection results with the industry, as appropriate.

Enhancements

None.

Conclusion

The PWR Reactor Vessel Internals Program provides reasonable assurance that change in dimension due to void swelling; cracking due to flaw initiation and growth, SCC/IGA, and IASCC; loss of preload due to stress relaxation; reduction in fracture toughness due to radiation and thermal embrittlement; and loss of material due to wear, of subject reactor vessel internals components will be adequately managed so that intended functions of components within the scope of license renewal are maintained consistent with the current licensing basis for the period of extended operation.

B.2.33 PWR WATER CHEMISTRY PROGRAM

Program Description

The PWR Water Chemistry Program mitigates damage due to loss of material, cracking, and reduction in heat transfer of plant components that are within the scope of license renewal and contain or are exposed to treated water or steam in the primary, secondary, or auxiliary systems. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material, cracking, or reduction in heat transfer through proper monitoring and control consistent with EPRI TR-1014986 Revision 6, "Pressurized Water Reactor Primary Water Chemistry Guidelines" and EPRI TR-102134 Revision 5, "Pressurized Water Reactor Secondary Water Chemistry Guidelines." The relevant conditions are known detrimental contaminants such as sulfates, halogens (chlorides and fluorides), dissolved oxygen, and conductivity that could lead to, or are indicative of, conditions for corrosion, stress corrosion cracking of susceptible materials, and reduction in heat transfer, as well as erosion. The PWR Water Chemistry Program is a mitigation program.

In addition, the PWR Water Chemistry Program is credited in conjunction with the Nickel-Alloy Management Program, Inservice Inspection Program, Nickel-Alloy Reactor Vessel Closure Head Nozzles Program, PWR Reactor Vessel Internals Program, Steam Generator Tube Integrity Program, and Small Bore Class 1 Piping Inspection to manage the effects of aging for reactor pressure vessel, reactor vessel internals, reactor coolant pressure boundary, and steam generator components.

The PWR Water Chemistry Program is also supplemented by a One-Time Inspection to provide verification of the effectiveness of the program in managing the effects of aging.

NUREG-1801 Consistency

The PWR Water Chemistry Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M2, "Water Chemistry."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The PWR Water Chemistry Program is an ongoing program that effectively incorporates the best practices of industry guidance and operating experience in defining chemistry control requirements, monitoring of plant performance with implementation, and continual review of their adequacy. The program incorporates EPRI guidelines as well as "lessons learned" from site and other utility operating experience. The program has been, and continues to be, subject to periodic assessment of performance to identify strengths, potential adverse trends, and areas for improvement. In addition, quarterly program health reports are generated that address chemistry performance indicators.

Review of site-specific operating experience has revealed that PWSCC has occurred. Repair or mitigative actions included structural weld overlay of Alloy 600/82/182 welds with materials known to be less susceptible to PWSCC. Otherwise, site-specific operating experience has revealed no loss of component intended function for components exposed to treated water or steam that could be attributed to an inadequacy of the PWR Water Chemistry Program. Abnormal chemistry conditions are promptly identified, evaluated, and corrected before a loss of function could occur. For example, reactor coolant lithium unexpectedly increased above the upper control band limit in December 2008, and the delithiating demineralizer was placed in service to restore the lithium to within control band limits. Also, the spent fuel pool chemistry trends indicated that sulfates were out of specification. This condition was evaluated through the Corrective Action Program and the spent fuel pool demineralizer was sluiced and charged with fresh resin to remedy the problem.

Furthermore, the program is periodically updated to the latest guidelines. The known chemistry-related problems experienced by other utilities are a consideration in the ongoing refinement of the PWR Water Chemistry Program for Davis-Besse.

The latest self-assessments noted that the Corrective Action Program is used extensively in the Chemistry Department, and that data review and reporting requirements are in compliance with procedures. A recent (2008) self-assessment found that the pressurizer dissolved oxygen parameter prior to 250°F was in disagreement with the EPRI guideline. This noteworthy item was addressed through the Corrective Action Program. The pertinent procedure has since been revised to reflect the most recent EPRI guideline, which remedied the discrepancy. The assessment also identified an area for improvement in the frequencies of monitoring diagnostic parameters for the various makeup sources for reactor coolant. This area for improvement was also addressed through the Corrective Action Program and tasks added to the chemistry routines to ensure diagnostic sampling is performed at the specified frequencies.

Conclusion

The PWR Water Chemistry Program has been demonstrated to be capable of managing loss of material, cracking, and reduction in heat transfer for susceptible components through monitoring and control of the relevant parameters in treated water (and steam). The PWR Water Chemistry Program is supplemented by the One-Time Inspection to verify effectiveness of the program in managing aging. The PWR Water Chemistry Program is also credited in conjunction with the Nickel-Alloy Management Program, Inservice Inspection Program, Nickel-Alloy Reactor Vessel Closure Head Nozzles Program, PWR Reactor Vessel Internals Program, Steam Generator Tube Integrity Program, and Small Bore Class 1 Piping Inspection. As supplemented, the PWR Water Chemistry Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.34 REACTOR HEAD CLOSURE STUDS PROGRAM

Program Description

The Reactor Head Closure Studs Program manages cracking and loss of material for the reactor head closure stud assemblies (studs, nuts, and washers). The Reactor Head Closure Studs Program is a combination mitigative and condition monitoring program.

The Reactor Head Closure Studs Program includes the preventive measures of NRC Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs," to mitigate cracking, including the use of a stable lubricant. An approved lubricant, GN Metal Assembly Spray or equivalent, is applied to the threaded areas of studs and nuts and to the concave and convex faces of the spherical washers during each assembly. There are no metal platings applied to the closure studs, nuts, or washers. A manganese-phosphate coating was applied to the studs, nuts and washers during fabrication to act as a rust inhibitor and to assist in retaining lubricant.

The Reactor Head Closure Studs Program examines reactor vessel stud assemblies in accordance with the examination and inspection requirements specified in the ASME Code, Section XI, Subsection IWB (1995 Edition through the 1996 Addenda) and approved ASME Code Cases. Visual examinations (VT-2) for leak detection are performed during system pressure tests.

The Reactor Head Closure Studs Program inspections are implemented by the Inservice Inspection Program. The Inservice Inspection Program will continue to comply with the requirements of the ASME Code Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by the NRC.

NUREG-1801 Consistency

The Reactor Head Closure Studs Program is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M3, "Reactor Head Closure Studs."

The Code year (e.g., 2001 edition including the 2002 and 2003 Addenda), as endorsed by the NRC in 10 CFR 50.55a, is specifically included in the NUREG-1801 XI.M1 aging management program. Consistent with provisions in 10 CFR 50.55a to use the ASME Code in effect twelve months prior to the start of the inspection interval, the applicable ASME Code for the current (third) ten year inspection interval for Davis-Besse is ASME Section XI, 1995 Edition, through the 1996 Addenda, as modified by 10 CFR 50.55a or relief granted in accordance with 10 CFR 50.55a.

Exceptions to NUREG-1801

None.

Enhancements

The following enhancement will be implemented in the identified program elements prior to the period of extended operation.

- **Scope, Preventive Actions**

  The Reactor Head Closure Studs program will be enhanced to select an alternate stable lubricant that is compatible with the fastener material and the environment. A specific precaution against the use of compounds containing sulfur (sulfide), including molybdenum disulfide (MoS_2), as a lubricant for the reactor head closure stud assemblies will be included in the program.

Operating Experience

The Reactor Head Closure Studs Program detects aging effects using nondestructive examination visual, surface, and volumetric techniques to detect and characterize flaws. These techniques are widely used and have been demonstrated effective at detecting aging effects during inspections performed to meet ASME Section XI Code requirements.

Review of Davis-Besse operating experience has not revealed any reactor head closure stud cracking or loss of material.

Nondestructive examinations of reactor head closure studs have been performed during two periods for the most recent (Third) Ten-Year Inspection Interval. These include visual examinations (VT-1) of 36 nuts, 36 washers, and two bushings; ultrasonic examination of 36 studs; and ultrasonic examination of 30 sets of threads in the vessel flange. In addition, visual examination (VT-3) of all 60 studs was performed. No unacceptable indications were noted in these examinations.

The Reactor Head Closure Studs Program has been developed based on relevant plant and industry operating experience. The Corrective Action Program and an ongoing review of industry operating experience will be used to ensure that the new program is effective in managing the identified aging effects.

Conclusion

The Reactor Head Closure Studs Program manages cracking and loss of material for the reactor head closure stud assemblies. The Reactor Head Closure Studs Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.35 REACTOR VESSEL SURVEILLANCE PROGRAM

Program Description

The Reactor Vessel Surveillance Program manages reduction of fracture toughness for the low-alloy steel reactor vessel shell and welds in the beltline region. Davis-Besse participates in the Pressurized Water Reactor Owners Group (PWROG) Master Integrated Reactor Vessel Surveillance Program (MIRVSP) which includes all seven operating B&W 177-fuel assembly plants and six participating Westinghouse-designed plants having B&W fabricated reactor vessels. The MIRVSP is described in topical report BAW-1543 (NP), "Master Integrated Reactor Vessel Surveillance Program," Revision 4, including supplements, and is an NRC-approved program that implements the requirements of Appendix H to 10 CFR Part 50. The Reactor Vessel Surveillance Program is a condition monitoring program.

Data resulting from the Reactor Vessel Surveillance Program is used to:

- determine pressure-temperature limits, minimum temperature requirements, and end of life upper shelf energy in accordance with the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements," and
- determine end of life reference temperature for pressurized thermal shock values in accordance with 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock."

Six surveillance capsules containing Davis-Besse specific materials were inserted into the reactor before initial plant startup. These capsules were designated as TE1-A through TE1-F. The requirements of 10 CFR 50 Appendix H were met by the first four capsules being withdrawn and tested. The remaining two capsules, TE1-C and TE1-E, have been removed and the materials have not been tested. Capsule TE1-C contains the Davis-Besse limiting material and has been exposed to fluence slightly above the 60-year projected fluence for the Davis-Besse plant. The Reactor Vessel Surveillance Program will be enhanced to require testing of capsule TE1-C. Capsule TE1-E has been discarded.

Since Davis-Besse does not have any surveillance capsules remaining inside the reactor vessel, ex-vessel cavity dosimetry is used to monitor neutron fluence.

NUREG-1801 Consistency

The Reactor Vessel Surveillance Program is an existing Davis-Besse program that, with enhancement, will be consistent with the elements of an effective aging management program as described in NUREG-1801, Section XI.M31, "Reactor Vessel Surveillance."

Note: NUREG-1801 Section XI.M31 does not follow the typical 10-element format.

Exceptions to NUREG-1801

None.

Enhancements

The following enhancement will be implemented in the identified program elements prior to the period of extended operation.

- **Monitoring and Trending**

  The Capsule Insertion and Withdrawal Schedule for Davis-Besse will be revised to schedule testing of the TE1-C capsule.

Operating Experience

Review of plant and industry operating experience provides reasonable assurance that the Reactor Vessel Surveillance Program will be effective in managing the effects of aging so that components within the scope of the program will continue to perform their intended function consistent with current licensing basis during the period of extended operation.

Davis-Besse participates in the MIRVSP as described in reports BAW-1543 (NP), supplements to this document, and BAW-10100A, "Compliance with 10 CFR 50, Appendix H, for Oconee Class Reactors." Participation in the MIRVSP ensures that future operating experience from all participating plants will be factored into the Reactor Vessel Surveillance Program. The NRC has concurred that the MIRVSP is an acceptable program.

Conclusion

The Reactor Vessel Surveillance Program has been demonstrated to be capable of managing reduction of fracture toughness for components of the reactor vessel beltline region. The Reactor Vessel Surveillance Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.36 SELECTIVE LEACHING INSPECTION

Program Description

The Selective Leaching Inspection will detect and characterize the conditions on internal and external surfaces of subject components that are exposed to moist air (including condensation), raw water, soil (buried), and treated water (including closed cycle cooling water). This one-time inspection provides direct evidence through visual inspection, material hardness measurement, or other appropriate examinations (such as chipping, scraping, or other mechanical means), of whether, and to what extent, loss of material due to selective leaching has occurred that could result in a loss of intended function. Evidence of significant aging revealed by the Selective Leaching Inspection will be entered into the Corrective Action Program. The resolution will include evaluation for expansion of the inspection sample size, locations, and frequency.

Implementation of the Selective Leaching Inspection will provide reasonable assurance that intended functions are maintained consistent with the current licensing basis for the period of extended operation. The inspection activities will be conducted just before the beginning of the period of extended operation.

NUREG-1801 Consistency

The Selective Leaching Inspection is a new one-time inspection for Davis-Besse that will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M33, "Selective Leaching of Materials."

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The Selective Leaching Inspection is credited for evaluating the condition of selective leaching susceptible components and assessing their ability to perform their intended function during the period of extended operation. Susceptible components include filter bodies, heat exchanger components, hydrants, moisture separators, piping, pump casings, spray nozzles, strainers, trap bodies, tubing, and

valve bodies. Components within the scope of the program are formed of gray cast iron or copper alloy > 15% zinc. The components are exposed to moist air (including condensation), raw water, soil (buried), and treated water (including closed cycle cooling water and steam) environments during normal plant operations. The one-time inspection includes visual inspection, hardness measurement, or other appropriate examinations (such as chipping, scraping, or other mechanical means), of a sample set of components to determine whether, and to what extent, selective leaching is occurring in the period of extended operation.

The aging management activity is credited for the following systems:

- Auxiliary Building Chilled Water System
- Auxiliary Building HVAC System
- Auxiliary Steam and Station Heating System
- Decay Heat Removal (DH) and Low Pressure Injection System (LPI)
- Emergency Diesel Generators (EDG)
- Fire Protection Diesel (DFP)
- Fire Protection System (FP)
- High Pressure Injection System
- Instrument Air System
- Main Steam System (MS)
- Makeup Water Treatment System
- Miscellaneous Liquid Radwaste System
- Service Water System (SW)
- Station Air System
- Station Blackout Diesel Generator (SBODG)
- Station Plumbing, Drains, and Sumps System (SPDSS)

- Preventive Actions
  No actions are taken as part of the Selective Leaching Inspection to prevent aging effects or to mitigate aging degradation. Although the control of water chemistry may reduce selective leaching in treated water environments, no specific credit is taken for water chemistry control as part of this program.

- Parameters Monitored or Inspected
  The Selective Leaching Inspection will perform visual inspection, hardness measurement, or other appropriate examinations (such as chipping, scraping, or other mechanical means), of components within the scope of the program as measures of loss of material due to selective leaching. Follow-up of unacceptable findings includes additional testing, as necessary, and expansion of the inspection sample size and location.

  The Selective Leaching Inspection activities will be conducted after the issuance of the renewed operating license and prior to the end of the current operating license, with sufficient time to implement programmatic oversight prior to the period of extended operation, if necessary. The activities will be conducted just before the period of extended operation, so that conditions are more representative of the conditions expected during that time.

- Detection of Aging Effects
  The Selective Leaching Inspection will include provision for visual inspection, hardness measurement, or other appropriate examinations (such as chipping, scraping, or other mechanical means), of a sample of components with susceptible materials in environments conducive to selective leaching. The program will include the criteria for visual inspection and for hardness measurement, or other appropriate examinations. The results of the inspections will be evaluated to determine the condition of the material. Engineering evaluation in conjunction with the Corrective Action Program will determine whether components with degraded materials are capable of performing their intended functions.

  The aging management activities include (a) determination of the sample size based on an assessment of materials of fabrication, environment/conditions, time in service, and operating experience; (b) identification of the inspection locations in the susceptible system or component; (c) determination of the examination technique, including acceptance criteria; and (d) evaluation of the need for follow-up examinations to monitor the progression of aging if age-related degradation is found that could jeopardize an intended function before the end of the period of extended operation.

  The results of the inspections will be evaluated against the acceptance criteria. Additional testing will be performed, as necessary, based on review of the inspection results.

- Monitoring and Trending
  No actions are taken as part of the Selective Leaching Inspection to monitor or trend inspection results. This is a one-time inspection activity used to determine if, and to what extent, further actions, including monitoring and trending, may be required.

The inspection results will be evaluated through the Corrective Action Program, if necessary.

- Acceptance Criteria
  The Selective Leaching Inspection will utilize approved inspection techniques to identify selective leaching. Inspection results that identify selective leaching will be entered into the Corrective Action Program. The Corrective Action Program includes provision for further evaluation of degraded materials and any necessary corrective actions. The evaluation will include a root cause analysis, if necessary.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  Plant design considerations address the potential for degradation of installed components through the application of materials suitable for the expected operating environments, and inspection methods will be consistent with accepted industry practices.

  Review of Davis-Besse operating experience did not identify any instances of loss of material due to selective leaching, graphitization, or dezincification for any in-scope components. Two items were identified for heat exchanger tubing in one heat exchanger not within the license renewal evaluation boundary, and the findings were associated with stagnant and low-flow conditions when the heat exchanger was not in service.

Conclusion

Implementation of the Selective Leaching Inspection will provide reasonable assurance that the aging effect will be managed so that components within the scope of this inspection will continue to perform their intended functions consistent with the current licensing basis during the period of extended operation.

B.2.37 SMALL BORE CLASS 1 PIPING INSPECTION

Program Description

The Small Bore Class 1 Piping Inspection will detect and characterize cracking of small bore ASME Code Class 1 piping less than 4 inches nominal pipe size (NPS 4), which includes pipe, fittings, and branch connections.

The ASME Code does not require volumetric examination of Class 1 small bore piping. The Small Bore Class 1 Piping Inspection will consist of volumetric examination of a representative sample of small bore piping locations that are susceptible to cracking. The inspection sample will include both socket welds and butt welds. The sample size and inspection locations will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small bore piping locations. The guidelines of EPRI Report 1011955, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)," and the supplemental guidelines 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)," will be considered in selecting the sample size and locations. Volumetric examinations (including qualified destructive or nondestructive techniques) will be performed by qualified personnel following procedures that are consistent with Section XI of the ASME Code and 10 CFR 50, Appendix B.

If a qualified non-destructive volumetric examination technique does not become available for socket welds, an opportunistic destructive examination will be conducted. Opportunistic destructive examination is performed when a weld is removed from service for other considerations, such as plant modifications. If a socket weld does not become available on an opportunistic bases, one will be selected for destructive testing. This socket weld will be selected from a piping location that is susceptible to cracking.

The inspection provides additional assurance that either age-related degradation of small bore ASME Code Class 1 piping components is not occurring or that the aging is insignificant, such that an aging management program is not warranted during the period of extended operation.

This one-time inspection is appropriate as Davis-Besse has experienced only two instances of cracking of small bore Class 1 piping, possibly due to stress corrosion or thermal and mechanical loading. Should evidence of significant aging be revealed by the one-time inspection or through plant operating experience, periodic inspection will be considered as a plant-specific aging management program.

The Small Bore Class 1 Piping Inspection is a new one-time inspection that will be implemented prior to the period of extended operation.

NUREG-1801 Consistency

The Small Bore Class 1 Piping Inspection is a new one-time inspection for Davis-Besse that will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M35, "One-time Inspection of ASME Code Class 1 Small-Bore Piping."

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The results of an evaluation of each program element are provided below.

- Scope
  The Small Bore Class 1 Piping Inspection is a one-time inspection of a sample of ASME Code Class 1 piping less than NPS 4. The inspection will include measures to verify that unacceptable degradation is not occurring in Class 1 small bore piping, thereby confirming that an aging management program is not needed for the period of extended operation. See *Monitoring and Trending* below for a discussion of sample selection and inputs.

- Preventive Actions
  The Small Bore Class 1 Piping Inspection will consist of evaluation and inspection activities with no actions to prevent or mitigate aging effects.

- Parameters Monitored or Inspected
  The Small Bore Class 1 Piping Inspection is a one-time inspection that will include volumetric examinations (destructive or nondestructive) performed by qualified personnel, using qualified volumetric examination techniques and following procedures consistent with Section XI of the ASME Code and 10 CFR 50, Appendix B.

- Detection of Aging Effects
  This inspection will perform volumetric examinations on selected weld locations. Davis-Besse has only experienced two instances of cracking of small bore Class 1

piping, possibly due to stress corrosion or thermal and mechanical loading, and therefore this one-time inspection is appropriate. See *Operating Experience* below for discussion of site operating experience.

If a qualified volumetric examination technique does not become available for socket welds, an opportunistic destructive examination will be conducted. Opportunistic destructive examination is performed when a weld is removed from service for other considerations, such as plant modifications. If a socket weld does not become available on an opportunistic bases, one will be selected for destructive testing. This socket weld will be from a piping location that is susceptible to cracking.

- Monitoring and Trending
  The one-time inspection will consist of volumetric examination of a representative sample of small bore piping locations that are susceptible to cracking. The sample size and inspection locations will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small bore piping locations. The guidelines of EPRI Report 1011955 and the supplemental guidelines of EPRI Report 1018330 will be considered in selecting the sample size and locations. Volumetric examinations (including qualified destructive or nondestructive techniques) will be performed by qualified personnel following procedures that are consistent with Section XI of the ASME Code and 10 CFR 50, Appendix B.

  Should evidence of significant aging be revealed by the one-time inspection or through plant operating experience, periodic inspection will be considered as a plant-specific aging management program.

- Acceptance Criteria
  Unacceptable inspection findings will be evaluated by the Corrective Action Program using criteria in accordance with the ASME Code.

- Corrective Actions
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Confirmation Process
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Administrative Controls
  This element is common to Davis-Besse programs and activities that are credited with aging management during the period of extended operation and is discussed in Section B.1.3.

- Operating Experience
  The Small Bore Class 1 Piping Inspection is a new one-time inspection activity for which plant operating experience does not indicate the need for an aging management program. The evaluations and examinations to be performed by this activity will use qualified non-destructive volumetric examination techniques or destructive examination techniques with demonstrated capability and a proven industry record to detect cracking in piping weld and base metal.

  Two instances of small bore piping cracking related to stress corrosion cracking have been identified at Davis-Besse.

  The first instance of cracking due to stress corrosion cracking was found in the reactor vessel closure gasket leakage monitoring line. It was determined that the stress corrosion cracking was promoted by chlorides left after water evaporated in the line. The issue was evaluated using the Corrective Action Program and it was determined that these lines are not indicative of other small bore piping. The affected piping was replaced and the procedure was changed to require draining of the line after use.

  The second instance of cracking was an axial indication found on the Reactor Coolant System loop 1 cold leg drain line 1-1 nozzle-to-elbow weld during the Cycle 14 refueling outage. The probable cause is extensive localized weld repair during initial construction. This repair either resulted in a latent flaw or a crack initiation site. The residual stresses from the construction weld repair, combined with the environment in the Reactor Coolant System and the susceptibility of Alloy 600 material, established the presence of the three key elements for the development of primary water stress corrosion cracking in spite of the low susceptibility in cold leg drain lines. This cracking was due to an event (local weld repair) and is not indicative of general aging in small bore lines.

  The evaluation of MRP-146 applicability to Davis-Besse is documented in the Corrective Action Program. As a result of the assessment, the inspection of three Reactor Coolant System drain lines was added to the inservice inspection schedule.

  The Small Bore Class 1 Piping Inspection will be developed based on relevant plant and industry operating experience.

Conclusion

The Small Bore Class 1 Piping Inspection will verify that cracking due to stress corrosion and mechanical loading is not occurring or is insignificant, such that an aging management program is not required during the period of extended operation. The Small Bore Class 1 Piping Inspection will provide reasonable assurance that the aging effects are not occurring such that components within the scope of this inspection will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.38 STEAM GENERATOR TUBE INTEGRITY PROGRAM

Program Description

The Steam Generator Tube Integrity Program is credited for aging management of cracking, denting, loss of material, and reduction in heat transfer of the steam generator tubes; as well as cracking of tube plugs, tube sleeves, and tube support plates. The Steam Generator Tube Integrity Program is performed as part of the overall Steam Generator Management program. The Steam Generator Management program is based on Technical Specification requirements, and is implemented in accordance with NEI 97-06, "Steam Generator Program Guidelines." The Steam Generator Tube Integrity Program also includes secondary-side examinations to assist in verification of tube integrity and the condition of the tube support plates.

The Steam Generator Tube Integrity Program is a combination condition monitoring and mitigation program. The Steam Generator Tube Integrity Program manages the effects of aging through a combination of prevention, inspection, evaluation, repair, and leakage monitoring. Preventative measures are intended to inhibit degradation and consist of primary-side and secondary-side water chemistry monitoring and control, and foreign material exclusion requirements.

The Steam Generator Tube Integrity Program provides the requirements for non-destructive examinations for the detection of flaws in tubes, plugs, sleeves, and tube support plates. Degradation assessments identify both potential and existing degradation mechanisms. Inservice inspections (i.e., eddy current testing and visual inspections) are used for the detection of flaws. Condition monitoring compares the inspection results against performance criteria, and an operational assessment provides a prediction of tube conditions to ensure that the performance criteria will be met throughout the next operating cycle. Primary-to-secondary leakage is continually monitored during operation.

NUREG-1801 Consistency

The Steam Generator Tube Integrity Program is an existing Davis-Besse program that is consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.M19, "Steam Generator Tube Integrity."

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

During each refueling outage, steam generator degradation assessments are performed in accordance with the provisions of NEI 97-06 and the EPRI pressurized water reactor steam generator examination guidelines. These industry guidelines are based in part on operating experience and inspection results from other operating pressurized water reactors. Degradation assessment topics include steam generator tube degradation mechanisms, inspection and expansion requirements, tube repair criteria, structural limits, guidelines for testing, and chemical cleaning provisions.

Davis-Besse has identified several instances of tube degradation through eddy current examination. Causes were determined to be mechanical equipment degradation, which is primarily a function of time in operation, temperature of operation, and chemistry conditions. Additional causes were predicted to be primary water stress corrosion cracking, stress corrosion cracking or intergranular attack, denting, and outer diameter stress corrosion cracking. Repairs were made through the Corrective Action Program.

As a result of the Cycle 15 refueling outage inspections, 46 steam generator tubes were plugged in once-through steam generator (OTSG) 2-A, bringing the total for that steam generator to 625 tubes plugged (4%). Thirty-five steam generator tubes were plugged in OTSG 1-B, bringing the total for that steam generator to 279 tubes plugged (1.8%). As with all previous inspections, the condition of the steam generators (with the degraded tubes plugged) met industry and regulatory structural and leakage integrity guidance, and were expected to meet these criteria following the outage inspection. Steam generator inspection results are addressed in the Inservice Inspection summary reports that are submitted to the NRC following each outage.

Self assessments of the program are performed periodically and conclude that the program is being effectively implemented, meets FENOC expectations regarding engineering programs, meets current industry requirements, and has incorporated industry identified beneficial practices.

Davis-Besse has not implemented the alternate repair criteria in Generic Letter 95-05, but has amended the Technical Specifications to be consistent with Technical Specification Task Force Report TSTF-449, "Steam Generator Tube Integrity," Revision 4.

The Davis-Besse evaluation of Information Notice 2008-07 concluded that the inspection scopes defined in the degradation assessments are appropriate for monitoring cracking in the expansion transition regions as well as at the upper and lower tube ends.

Using the accepted industry approach to testing and evaluation, and incorporation of pertinent industry operating experience, insures that the Steam Generator Tube

Integrity Program manages the effects of component aging such that the steam generators will continue to perform their intended functions, consistent with the current licensing basis, during the period of extended operation.

Conclusion

The Steam Generator Tube Integrity Program has been demonstrated to be capable of managing age-related degradation of the steam generator tubes, tube plugs, tube sleeves, and tube support plates. The Steam Generator Tube Integrity Program provides reasonable assurance that the aging effects will be managed such that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.39 STRUCTURES MONITORING PROGRAM

Program Description

The Structures Monitoring Program is part of the Maintenance Rule program. It is an existing program that is designed to ensure age-related degradation of the plant structures and structural components within its scope are managed such that each structure and structural component retains the ability to perform its intended function. The Maintenance Rule program is comprised of many existing monitoring and assessment activities which collectively address potential and actual degradation conditions and their effects on the reliability of structures and components.

The Structures Monitoring Program implements provisions of the Maintenance Rule, 10 CFR 50.65, which relate to structures, masonry walls, and water control structures. It conforms to the guidance contained in Regulatory Guide 1.160 and NUMARC 93-01. Concrete, masonry walls, and other structural components that perform a fire barrier intended function are also managed by the Fire Protection Program.

The Structures Monitoring Program encompasses and implements the Water Control Structures Inspection and the Masonry Wall Inspection.

NUREG-1801 Consistency

The Structures Monitoring Program is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.S6, "Structures Monitoring Program."

Exceptions to NUREG-1801

None.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Scope**

  The program procedure will be enhanced by including and listing the structures within the scope of license renewal that credit the Structures Monitoring Program for aging management.

- **Parameters Monitored or Inspected**

  The program procedure will be enhanced by including aging effect terminology (e.g., loss of material, cracking, change in material properties, and loss of form).

- **Parameters Monitored or Inspected**

  The program procedure will be enhanced by listing American Concrete Institute (ACI) 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," and American National Standards Institute / American Society of Civil Engineers (ANSI/ASCE) 11-90, "Guideline for Structural Condition Assessments of Existing Buildings," as references and to indicate that they provide guidance for the selection of parameters monitored or inspected.

- **Parameters Monitored or Inspected**

  The program procedure will be enhanced by providing clarification that a "structural component" for inspection includes each of the component types identified within the scope of license renewal as requiring aging management.

- **Parameters Monitored or Inspected**

  The program procedure will be enhanced by requiring the responsible engineer to review site raw water pH, chlorides, and sulfates test results prior to the inspection to take into account the raw water chemistry for any unusual trends during the period of extended operation. Raw water chemistry data shall be collected at least once every five years. Data collection dates shall be staggered from year to year (summer-winter-summer) to account for seasonal variation.

- **Parameters Monitored or Inspected**

  Davis-Besse's area groundwater is aggressive and operating experience has shown that structural elements have experienced degradation. Although there is no evidence that the aggressive groundwater has contributed to structural degradation, a special provision in the program will be implemented to monitor below-grade inaccessible concrete components before and during the period of extended operation. FENOC will perform a below-grade examination of concrete below elevation 570 feet (groundwater elevation) of an in-scope structure prior to the period of extended operation. That inspection will include concrete examination using acceptance criteria from NUREG-1801 XI.S6 Program element 6. The below-grade examination of concrete below elevation 570 feet may be conducted during maintenance activities. Any degradation found that exceeds the acceptance criteria will be trended and processed through the Corrective Action Program.

- **Parameters Monitored or Inspected**

  The program procedure will be enhanced by specifying that, upon notification that a below-grade structural wall or other in-scope concrete structural component will become accessible through excavation, a follow-up action is initiated to the responsible engineer to inspect the exposed surfaces for age-related degradation. Such inspections will include concrete examination using acceptance criteria from NUREG-1801 XI.S6 Program element 6. Any degradation found that exceeds the acceptance criteria will be trended and processed through the Corrective Action Program.

- **Detection of Aging Effects**

  The Structures Monitoring Program procedure will be enhanced by listing ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," ANSI/ASCE 11-90, "Guideline for Structural Condition Assessments of Existing Buildings," and EPRI Report 1007933, "Aging Assessment Field Guide" as references and to indicate that they provide guidance for detection of aging effects.

- **Monitoring and Trending**

  The program procedure will be enhanced by including requirements to follow the documentation requirements of 10 CFR 54.37 and to submit records of structural evaluations to records management.

- **Acceptance Criteria**

  The program procedure will be enhanced by indicating that ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," provides acceptable guidelines which will be considered in developing acceptance criteria for concrete structural elements, steel liners, joints, coatings, and waterproofing membranes.

Operating Experience

The Structures Monitoring Program has been effective in managing age-related degradation. Visual inspections conducted by the Structures Monitoring Program have found some age-related issues. These age-related issues have been processed through the Corrective Action Program. Inspections also found minor degradation conditions including small shrinkage cracks, construction joint voids, rust, and surface irregularities which did not require further evaluation. Acceptable minor degradation has been noted on Maintenance Rule evaluation documents and reviewed and re-inspected during subsequent inspections. With the exception of the auxiliary feedwater pump turbine exhaust missile barrier, which has a "W" rating indicating that it is acceptable with deficiencies which are being resolved by the Corrective Action Program, all other inspected structures are acceptable and are capable of performing their design functions.

Review of completed Maintenance Rule evaluations indicated that conditions of age-related degradation were identified and documented. Degradation conditions requiring repair were processed through the work order system and the Corrective Action Program.

Examples of conditions found were:

- Auxiliary feedwater pump turbine exhaust missile barrier has spalled concrete and exposed rebar due to its periodic exposure to a harsh environment. The missile barrier continues to perform its design function and the Corrective Action Program is tracking the repair.
- Pipe tunnel has minor surface cracks and chipped area around a doorway. There were signs of water intrusion near a penetration but the condition was determined to be acceptable. ECCS Pump Room No. 1 has signs of water intrusion, no active leakage was noted and the condition was determined to be acceptable.
- Auxiliary Building has various small spalled areas and surface cracks less than 1/16 inch. Shrinkage cracks in seismic joints and block wall to concrete interface were noted in the baseline inspection and subsequent inspections. Efflorescence was noted in some areas with no active leakage. These conditions were deemed acceptable and pose no structural concerns.
- Signs of leakage from a junction box were noted during an Auxiliary Building inspection and were processed through the Corrective Action Program. Separation of expansion joint seals identified in Rooms 601 and 602 was processed through the Corrective Action Program.
- Minor rust and staining on supports from past system leakage was noted during an Auxiliary Building inspection, they weren't properly cleaned and recoated. Rust spots and minor pitting were noted on overhead floor decking. These conditions were deemed acceptable and pose no structural concerns.
- Housekeeping issues in a room with abandoned equipment and various unfilled abandoned anchor bolt holes were noted during an Auxiliary Building inspection.
- Large spalled concrete in Room 236 southwest corner was identified and evaluated through the Corrective Action Program. Large grout undercutting at a column base in Room 313 was identified and processed through the work order system for repair.

- Extensive paint flaking was noted during an Auxiliary Building inspection on structural fireproofing in Room 323 and processed through the work order system for repair. Fireproofing material appears to be unaffected.
- Auxiliary Building roof system conditions are adequate. Minor cracks in the asphalt flashing and some debris blocking roof drain screens were noted, condition was processed through the work order system for repair or rework.
- Borated Water Storage Tank (BWST) trench has active water leakage observed on majority of trench floor due to failed weather seals and a vertical expansion joint seal located in the southeast comer is degraded. The BWST Level Transmitter Building roof insulation joints are taped together with duct tape and various locations exhibit duct tape that has peeled away and active water leakage observed at southeast comer caused by ponding of water on opposite side of wall. No structural implications exist due to water leakage. The work order system was used to address these conditions.
- Containment inspections revealed various small spalled areas, chipped concrete and surface cracks less than 1/16 inch. Shrinkage cracks and worn coating on concrete floor were noted in the baseline inspection and subsequent inspections. Minor rust and staining on supports and structural steel were also noted. These conditions were deemed acceptable and pose no structural concerns.
- Electrical manhole 3005 has some minor cracks and spalling near conduit supports. There was a small amount of water present on the floor of both the north and south cubicles. The water appeared to be draining to the sump pit located in the south cubicle. The source of the water appeared to be from the bottom row of conduits and duct bank. At the upper left corner of the duct bank interface, there was a concrete void and the waterstop material was visible. The work order system was used to have the voided area filled in with new concrete.
- Minor spalling of grout was observed at the base of the Condensate Storage Tanks. The conditions were deemed acceptable and pose no structural concerns.
- The flashing on the Relay House roof has surface rust and requires re-painting. Currently there is no adverse affect to the roof. The precast concrete panels on the exterior of the building have various locations that are spalled and the basement south wall has a vertical crack at the location where a future doorway was intended. The work order system was used to request correction of this issue. The south doorway canopy has a completely sheared rod hanger. The Corrective Action Program was used to evaluate this issue.
- Service water pipe tunnel valve rooms have minor active water in-leakage. The work order system was used to address this issue.

- Small shrinkage cracks and minor spalling where concrete repairs had taken place were noted in the parapet wall at several locations and on the Shield Building dome. The cracks found do not pose any structural concerns. Digital image was taken to provide documentation and reference for future evaluations.
- Pitting corrosion was noted in the sand pocket area of the containment vessel. The vessel has been coated in this region. No new pitting was identified in this area. The existing pitting was identified and evaluated through the Corrective Action Program and found to be acceptable. Ultrasonic thickness measurements verified that the minimum recorded vessel thickness was greater than the minimum required wall thickness. Several locations within the sand pocket area contained standing water. Beveled grout has been installed in the area, the standing water was not in contact with the containment vessel.
- Switchyard structural steel has surface rust present. The surface rust does not adversely affect the structural steel's adequacy. The work order system was used to request re-painting the Switchyard's structural steel.
- Several tower and disconnect switch concrete foundations in the Switchyard are degraded to the point that concrete has spalled off and rebar is visible. The Corrective Action Program was used to evaluate this issue. The Switchyard's ground appeared to be saturated with ground water due to insufficient drainage. The Corrective Action Program was used to evaluate this issue.
- Fire walls between yard transformers have various small spalled areas and surface cracks less than 1/16 inch noted in the baseline inspections and subsequent inspections. These conditions were deemed acceptable and pose no structural concerns.
- Turbine Building elevation 565 has active water in-leakage, sections of expansion joint missing in Room 253, and degraded vertical expansion joints in Room 330 that needed re-work. The work order system was used to address these conditions. Minor spalled areas and surface cracks less than 1/16 inch noted in the baseline inspections and subsequent inspections. These conditions were deemed acceptable and pose no structural concerns.
- Water Treatment Building has minor spalled areas, surface cracks, and water stains on walls noted in the baseline inspections and subsequent inspections. These conditions were deemed acceptable and pose no structural concerns.

The Structures Monitoring Program provides reasonable assurance that aging effects are being managed for the Davis-Besse structures. This has been demonstrated through inspection reports and the Corrective Action Program.

The Corrective Action Program and ongoing review of industry operating experience will be used to ensure that the program continues to be effective in managing the identified aging effects.

Conclusion

The Structures Monitoring Program, with enhancement, will be capable of detecting and managing aging effects for structures within the scope of license renewal. The continued implementation of the Structures Monitoring Program, with enhancement, provides reasonable assurance that the effects of aging will be managed so that components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B.2.40 WATER CONTROL STRUCTURES INSPECTION

Program Description

The Water Control Structures Inspection is implemented as part of the Structures Monitoring Program, conducted for the Maintenance Rule.

The Water Control Structures Inspection is an existing condition monitoring program for detecting age-related degradation of the Intake Structure, Forebay, Service Water Discharge Structure, and those structural components within the structures.

Davis-Besse is not committed to RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants." However, enhancements pertaining to water control structure inspection elements from RG 1.127 Revision 1 will be incorporated into the Water Control Structures Inspection, implemented as part of the Structures Monitoring Program, consistent with NUREG-1801, Section XI.S7.

NUREG-1801 Consistency

The Water Control Structures Inspection is an existing Davis-Besse program that, with enhancement, will be consistent with the 10 elements of an effective aging management program as described in NUREG-1801, Section XI.S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants," with the following exceptions.

Exceptions to NUREG-1801

Program Elements Affected:

- **Scope, Parameters Monitored or Inspected, and Detection of Aging Effects**

  Dams, spillway structures, outlet works, reservoirs, and water-control structure safety and performance instrumentation are not installed at Davis-Besse. Therefore, the associated portions of the NUREG-1801, XI.S7 program are not applicable to the Water Control Structures Inspection for Davis-Besse.

- **Acceptance Criteria**

  Earthen structures falling within the regulatory jurisdiction of the Federal Energy Regulatory Commission or the U. S. Army Corps of Engineers are not installed at Davis-Besse. Therefore, the associated portions of the NUREG-1801, XI.S7 program are not applicable to the Water Control Structures Inspection for Davis-Besse.

Enhancements

The following enhancements will be implemented in the identified program elements prior to the period of extended operation.

- **Scope**

  The Water Control Structures Inspection, included in the existing Structures Monitoring Program, will include the Service Water Discharge Structure which is within the scope of license renewal.

- **Parameters Monitored or Inspected**

  The Water Control Structures Inspection, included in the existing Structures Monitoring Program, will include parameters monitored and inspected for water control structures, including the Service Water Discharge Structure, in accordance with applicable inspection elements listed in Section C.2 of RG 1.127 Revision 1. Descriptions of concrete conditions will conform with the appendix to the American Concrete Institute (ACI) publication, ACI 201, "Guide for Making a Condition Survey of Concrete in Service." The use of photographs for comparison of previous and present conditions will be included as a part of the inspection program.

- **Detection of Aging Effects**

  The Water Control Structures Inspection, included in the existing Structures Monitoring Program, will specify that water control structure periodic inspections are to be performed at least once every five years.

- **Monitoring and Trending**

  The Water Control Structures Inspection, included in the existing Structures Monitoring Program, will include requirements to follow the documentation requirement of 10 CFR 54.37, including submittal of records of structural evaluations to records management.

- **Acceptance Criteria**

  The Water Control Structures Inspection, included in the existing Structures Monitoring Program, will list ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," as a reference and will indicate that it will be considered in developing acceptance criteria for inspection of water control structures.

Operating Experience

The Water Control Structures Inspection has been effective in managing age-related degradation. Visual inspections conducted by the Water Control Structures Inspection have found some age-related problems. Age-related issues have been processed through the Corrective Action Program and repaired.

Monitoring for degradation of the ultimate heat sink embankments has historically been performed by system engineer walkdown looking for obvious signs of erosion and possible displacement. The only degradation that had been found prior to 2007 was erosion of the earthen embankment during 1999 in the nonsafety-related portion of the canal, which was promptly repaired. During a routine inspection in 2007 of the intake canal under the Preventive Maintenance program, the north side of the embankment in the safety-related portion of the intake canal (Forebay) was found to have settled for a length of approximately 200 feet, which greatly reduced the slope of the embankment. Evaluation of this area found that it is stable. The slope stability study performed as a corrective action found the degradation in the north sidewall of the Forebay between stations 500-1000 feet occurred as a result of the presence of low compressive strength Lacustrine till (brown clay). Diver inspection of this area revealed the toe of the embankment does not appear to have moved, suggesting the degradation is limited to the embankment above the water surface. The degradation found during 2007 is believed to have occurred slowly over a period of time so that it was not distinguishable until gross slope degradation was observed. Based on this finding and to identify any future degradation of the embankments, preventive maintenance was established that will measure the slope, width, elevation, and length of the intake canal to preserve the volume of water available. The frequency of the preventive maintenance task is every three years. The results of the inspections are documented in the work order system used to perform the preventive maintenance, in the Corrective Action Program (as needed), and in the system chronological log. An engineering modification has been planned to repair the degraded area of the north wall of the Intake Canal.

In September 2008, the NRC conducted a triennial inspection of Davis-Besse's ultimate heat sink performance. No findings of significance were identified. The NRC inspectors verified that FENOC's inspection of the ultimate heat sink was thorough and of significant depth to identify degradation of the shoreline protection or loss of structural integrity. The inspectors verified vegetation present along the slopes was trimmed, maintained, and was not adversely impacting the embankment. The inspectors verified that FENOC ensured sufficient reservoir capacity by trending and removing debris or sediment buildup in the ultimate heat sink. The inspectors performed a system walkdown of the service water Intake Structure and verified FENOC's assessment of structural integrity and component functionality. This inspection included the verification that FENOC ensured proper functioning of traveling screens and strainers, and structural integrity of component mounts. In addition, the inspectors verified that service water pump bay silt accumulation is monitored, trended, and maintained at an acceptable level. The Corrective Action Program documentation related to the heat sink performance issues was reviewed to verify that FENOC had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions.

Review of completed Maintenance Rule inspection results indicated that age-related degradation was identified and documented through the Corrective Action Program. Water control structures were found to be in good condition below and above the water level. Normal silt sedimentation and biological growth were dredged and cleaned. Underwater inspections were documented via written report and video. Examples of conditions found were:

- Intake Structure concrete is in good condition above and below water level.
- Steel sheet piling at Forebay area by the Intake Structure had surface rust but material thickness was acceptable.
- Degraded earthen dikes were identified and repaired.
- Vegetation on earthen dikes was identified and cleared.
- Isolated small holes due to burrowing animals were identified, but no structural stability concerns were noted.

Review of program health reports has concluded that water control structures within license renewal scope are in good working condition with the exception of the erosion of the earthen embankment discussed above.

The Corrective Action Program and ongoing review of industry operating experience will be used to ensure that the program continues to be effective in managing the identified aging effects.

Conclusion

The Water Control Structures Inspection, with enhancement, will be capable of detecting and managing aging effects for structures within the scope of license renewal. The continued implementation of the Water Control Structures Inspection, with enhancements, provides reasonable assurance that the effects of aging will be managed so that components within the scope of this inspection will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

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APPENDIX C

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APPENDIX D

TECHNICAL SPECIFICATION CHANGES

10 CFR 54.22 requires that an application for license renewal include any Technical Specification changes or additions necessary to manage the effects of aging during the period of extended operation.

No changes to the Davis-Besse Technical Specifications are required to support the License Renewal Application.

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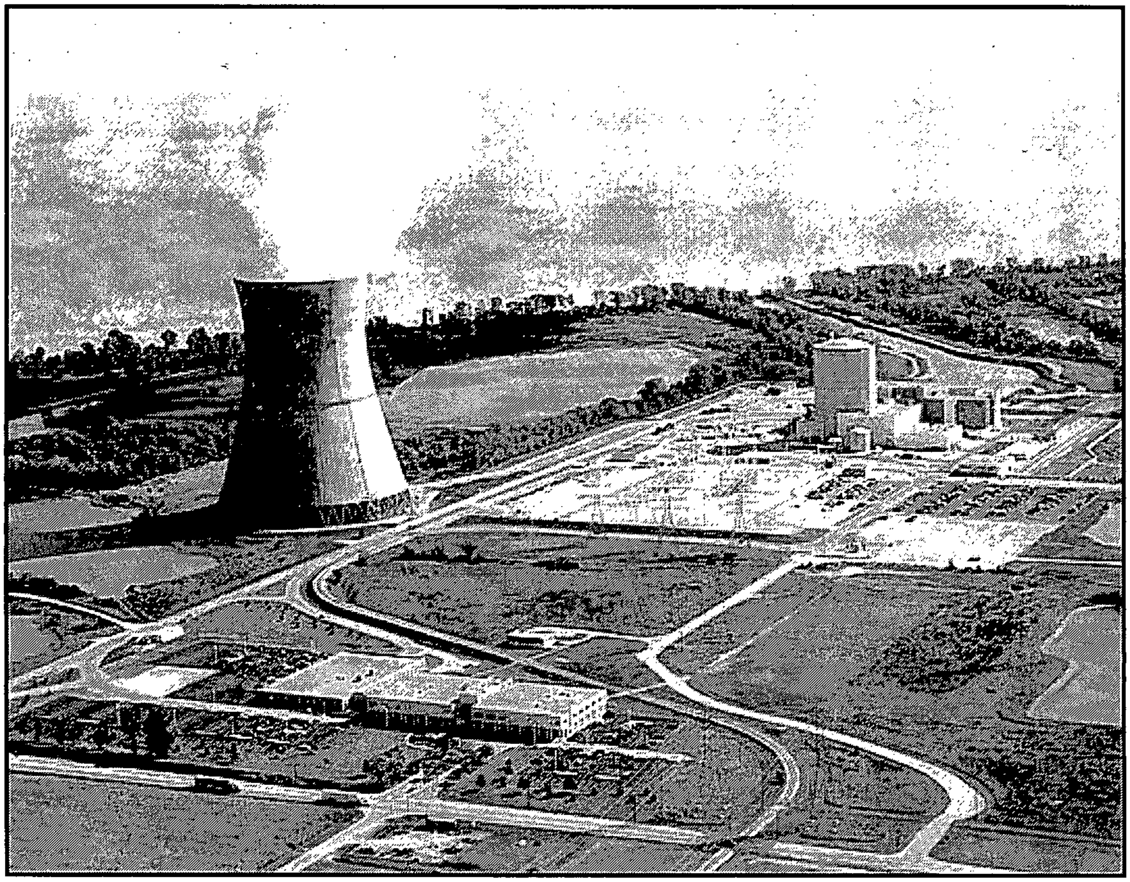
Appendix E

Applicant's Environmental Report
Operating License Renewal Stage

Davis-Besse Nuclear Power Station



August 2010

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Acronyms and Abbreviations

Acronym	Definition
AADT	annual average daily traffic
AEC	Atomic Energy Commission
BSBO	Black Swamp Bird Observatory
BVPS	Beaver Valley Power Station
Btu	British thermal unit
°C	degrees Celsius
CDF	core damage frequency
CEQ	Council on Environmental Quality
CET	containment event tree
CFR	Code of Federal Regulations
cfs	cubic feet per second
CO	carbon monoxide
CO_2	carbon dioxide
CWA	Clean Water Act
CWS	Circulating Water System
Davis-Besse	Davis-Besse Nuclear Power Station
DSM	demand-side management
EFH	Essential Fish Habitat
EIA	Energy Information Administration
EPRI	Electric Power Research Institute
ER	environmental report
ESA	Endangered Species Act
°F	degrees Fahrenheit
FBC	Fluidized-bed-combustion
FE	FirstEnergy Corporation
FENGenCo	FirstEnergy Nuclear Generation Corp.
FENOC	FirstEnergy Nuclear Operating Company
FERC	Federal Energy Regulatory Commission

Acronyms and Abbreviations
(continued)

Acronym	Definition
FES	Final Environmental Statement
fps	feet per second
ft^3	cubic feet
gal	gallon
GEIS	Generic Environmental Impact Statement
gpd	gallons per day
gpm	gallons per minute
IGCC	integrated gasification combined cycle
IPA	Integrated Plant Assessment
kWh	kilowatt-hour
kV	kilovolt
lb	pound
lb/MMBtu	pounds per million British thermal units
LOS	level of service
m^3	cubic meters
mA	milliampere
MAAP	Modular Accident Analysis Program
MACCS2	MELCOR Accident Consequence Code System
MDC	Minimum Detection Concentration
mg/l	milligrams per liter
mgd	million gallons per day
MM	million
MSW	municipal solid waste
MW	megawatt
MWd/MTU	megawatt-days per metric ton uranium
MMBtu	million British thermal unit
MWe	megawatts-electric
MWh	megawatt-hour

Acronyms and Abbreviations
(continued)

Acronym	Definition
MWt	megawatts-thermal
NAAQS	National Ambient Air Quality Standards
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NESC	National Electrical Safety Code
NMFS	National Marine Fisheries Service
NO_x	nitrogen oxides
NOAA	National Oceanic and Atmospheric Administration
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NRHP	National Register of Historic Places
NRR	Office of Nuclear Reactor Regulation
OAC	Ohio Administrative Code
OCMP	Ohio Coastal Management Program
ODCM	Off-site Dose Calculation Manual
ODNR	Ohio Department of Natural Resources
OEPA	Ohio Environmental Protection Agency
OHPO	Ohio Historic Preservation Office
ONWR	Ottawa National Wildlife Refuge
OPSB	Ohio Power Siting Board
pCi/L	picoCuries per liter
PDS	plant damage state
PEIS	programmatic environment impact statement
PCBs	polychlorinated byphenyls
PM	particulate matter
PM_{10}	particulates with diameters less than 10 microns
$PM_{2.5}$	particulates with diameters less than 2.5 microns
ppb	parts per billion

Acronyms and Abbreviations
(continued)

Acronym	Definition
ppm	parts per million
ppt	parts per thousand
PRA	probabilistic risk assessment
psig	pounds per square inch gauge
rms	root mean square
RC	release category
RCS	Reactor Coolant System
ROW	Right of Way
SAMA	Severe Accident Mitigation Alternatives
scf	standard cubic feet
SHPO	State Historic Preservation Officer
SO_2	sulfur dioxide
SO_x	sulfur oxides
SU	standard units
SWS	Service Water System
USACE	U.S. Army Corps of Engineers
USAR	Updated Safety Analysis Report
USCB	U.S. Census Bureau
USDOD	U.S. Department of Defense
USDOE	U.S. Department of Energy
USEPA	U.S. Environmental Protection Agency
USFWS	U.S. Fish and Wildlife Service
USGS	U.S. Geological Survey
USOSHA	U.S. Occupational Safety and Health Administration
wt%	percent by weight
yr	year

1.0 INTRODUCTION

1.1 PURPOSE OF AND NEED FOR ACTION

FirstEnergy Nuclear Operating Company (FENOC) prepared this Environmental Report (ER) to support renewal of the Class 103 facility operating license for Davis-Besse Nuclear Power Station, Unit 1 (Davis-Besse) (facility operating license NPF-3) for a period of 20 years beyond the expiration of the current license term. License renewal would extend the facility operating license from midnight on April 22, 2017, to midnight on April 22, 2037. Davis-Besse Operating License NPF-3 was issued on April 22, 1977, and the plant began commercial operation on July 31, 1978 (**FENOC 2010**, Section 1.1). Per 10 CFR 50.51, the license allows the plant to operate up to 40 years, and may be renewed for a period of up to an additional 20 years (10 CFR 54.31).

For license renewal, the U.S. Nuclear Regulatory Commission (NRC) has defined (**NRC 1996a**, Page 28,472) the purpose and need for the proposed action as follows:

> *The purpose and need for the proposed action (renewal of an operating license) is to provide an option that allows for power generation capability beyond the term of a current nuclear power plant operating license to meet future system generating needs, as such needs may be determined by State, utility, and, where authorized, Federal (other than NRC) decision makers.*

The proposed action would provide FENOC the option to operate Davis-Besse for an additional 20 years beyond the current licensed operating period.

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1.2 ENVIRONMENTAL REPORT SCOPE AND METHODOLOGY

NRC regulation 10 CFR 51.53(c) requires that an applicant for license renewal submit with its application a separate document entitled *Applicant's Environmental Report - Operating License Renewal Stage*. This report fulfills that requirement and is an appendix to the Davis-Besse license renewal application.

The requirements regarding information to be included in the environmental report (ER) are codified in 10 CFR 51.45 and 51.53(c). Table 1.2-1 lists the regulatory requirements and identifies the ER sections that respond to the requirements. In addition, affected ER sections are prefaced by a boxed quote of the relevant regulatory language.

The ER has been developed to meet the format and content of Supplement 1 to Regulatory Guide 4.2 (**NRC 2000**). Additional insight regarding content was garnered from the NRC's Generic Environmental Impact Statement (GEIS) for license renewal (**NRC 1996b**) and standard review plans for environmental reviews (**NRC 1999**), and supplements to the GEIS.

Table 1.2-1: Environmental Report Responses to License Renewal Environmental Regulatory Requirements

Regulatory Requirement	Description	ER Section(s)
10 CFR 51.53(c)(1)	Operating license renewal stage ER.	Entire Document
10 CFR 51.53(c)(2)	Proposed action description.	3.0
10 CFR 51.53(c)(2) and 10 CFR 51.45(b)(3)	Environmental impacts and comparison of alternatives.	7.3, 8.0
10 CFR 51.53(c)(2) and 10 CFR 51.45(b)(1)	Proposed action impact on the environment.	4.0
10 CFR 51.53(c)(2) and 10 CFR 51.45(b)(2)	Unavoidable adverse environmental impacts.	6.3
10 CFR 51.53(c)(2) and 10 CFR 51.45(b)(4)	Local short-term uses vs. long-term productivity of the environment.	6.5
10 CFR 51.53(c)(2) and 10 CFR 51.45(b)(5)	Irreversible and irretrievable commitments of resources.	6.4
10 CFR 51.53(c)(2) and 10 CFR 51.45(c)	Environmental analysis of the proposed action and mitigating actions,	4.0, 6.2
	environmental impacts of alternatives, and	7.3
	alternatives available for reducing or avoiding adverse environmental effects.	8.0
10 CFR 51.53(c)(2) and 10 CFR 51.45(d)	Status of compliance.	9.0
10 CFR 51.53(c)(2) and 10 CFR 51.45(b)(2) and (e)	Proposed action impact on the environment and unavoidable adverse impacts.	4.0, 6.3
10 CFR 51.53(c)(3)(ii)(A)	Water use conflicts (plants using cooling towers or ponds and withdrawing from a small river).	4.1, 4.6
10 CFR 51.53(c)(3)(ii)(B)	Entrainment, impingement, and heat shock assessment (plants using once-through cooling or cooling ponds).	4.2, 4.3, 4.4
10 CFR 51.53(c)(3)(ii)(C)	Groundwater use conflicts (plants using Ranney wells or >100 gpm groundwater).	4.5, 4.7

Table 1.2-1: Environmental Report Responses to License Renewal Environmental Regulatory Requirements
(continued)

Regulatory Requirement	Description	ER Section(s)
10 CFR 51.53(c)(3)(ii)(D)	Groundwater quality degradation.	4.8
10 CFR 51.53(c)(3)(ii)(E)	Impact of refurbishment on terrestrial resources, and	4.9
	threatened or endangered species.	4.10
10 CFR 51.53(c)(3)(ii)(F)	Assessment of air quality during refurbishment (nonattainment areas).	4.11
10 CFR 51.53(c)(3)(ii)(G)	Impact on public health from thermophilic organisms.	4.12
10 CFR 51.53(c)(3)(ii)(H)	Potential shock hazard from transmission lines.	4.13
10 CFR 51.53(c)(3)(ii)(I)	Assessment of refurbishment on housing,	4.14
	public water supply,	4.15
	public schools, and	4.16
	land use.	4.17
10 CFR 51.53(c)(3)(ii)(J)	Assessment of local highway traffic during refurbishment.	4.18
10 CFR 51.53(c)(3)(ii)(K)	Assessment of historic or archaeological properties.	4.19
10 CFR 51.53(c)(3)(ii)(L)	Alternatives to mitigate severe accidents.	4.20
10 CFR 51.53(c)(3)(iii)	Reducing adverse impacts.	6.2
10 CFR 51.53(c)(3)(iv)	New and significant information.	5.0
10 CFR Part 51, Appendix B, Table B-1, Footnote 6	Environmental Justice.	2.6.2, 4.21

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1.3 DAVIS-BESSE NUCLEAR POWER STATION LICENSEE AND OWNERSHIP

Davis-Besse is owned by FirstEnergy Nuclear Generation Corp. (FENGenCo). Both FENGenCo and FENOC are the licensees. FENOC is the applicant and, acting on behalf of FENGenCo, is also the operator with exclusive responsibility and control over the operation and maintenance of Davis-Besse. (**FENOC 2010**, Section 1.4.1)

FENOC is a wholly owned direct subsidiary of FirstEnergy Corp., a public utility holding company.

FirstEnergy Nuclear Generation Corp. is a wholly owned direct subsidiary of FirstEnergy Solutions Corp., and a wholly owned second-tier subsidiary of FirstEnergy Corp (FE).

FirstEnergy Solutions Corp. is a wholly owned direct subsidiary of FirstEnergy Corp.

References to a previous owner, the Toledo Edison Company, have been retained, where appropriate, for historical purposes. (**FENOC 2010**, Section 1.4.1)

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1.4 REFERENCES

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NRC 1996b. Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants (GEIS), NUREG-1437, Volumes 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, May 1996.

NRC 1999. Standard Review Plans for Environmental Reviews for Nuclear Power Plants, NUREG-1555, Supplement 1, Operating License Renewal, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, October 1999.

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2.0 SITE AND ENVIRONMENTAL INTERFACES

This chapter describes the overall character of the Davis-Besse site and local environment. Its purpose is to portray the plant's setting and the environment affected, with particular attention to information required to address the environmental issues designated by the GEIS (**NRC 1996**) as Category 2.

2.1 LOCATION AND FEATURES

Davis-Besse is located on the southwestern shore of Lake Erie in Ottawa County, Ohio, in Section 12 of Township 8 North, Range 15 East. Nearby communities include Oak Harbor approximately 8 miles southeast, Fremont 16 miles south, and Toledo 25 miles west northwest. Prominent features of the surrounding area out to 50 miles are shown in Figure 2.1-1. The area within six miles is shown on Figure 2.1-2.

The station structures are located approximately in the center of the site 3,000 feet from the shoreline, which provides a minimum exclusion distance of 2,400 feet from any point on the site boundary. The reactor is located at 41° 35' 49" north Latitude and 83° 05' 16" west Longitude. The approximate Universal Transverse Mercator coordinates are 4,607,000 meters north and 326,100 meters east (**FENOC 2010**, Section 2.1.1).

The low population zone is an area outside the site boundary within a radius of two miles from the center of the containment structures (**FENOC 2010**, Section 2.1.3.3). Figure 2.1-3 shows the site boundaries and exclusion area. Section 3.1 describes key features of Davis-Besse, including reactor and containment systems, cooling water system, and transmission system.

The site consists of 954 acres, of which approximately 733 acres are marshland that is leased to the U.S. Government as a national wildlife refuge (**FENOC 2010**, Section 2.1.2). To the west is the main unit of the Ottawa National Wildlife Refuge and the state of Ohio Magee Marsh Wildlife Area. On the southern boundary is the Toussaint River, which empties into Lake Erie 700 feet from the lake shoreline site boundary (Figure 2.1-3). The land area surrounding the site is generally agricultural with no major industry in the vicinity.

The topography of the site and vicinity is flat with marsh areas bordering the lake and the upland area rising to only 10 to 15 feet above the lake low water datum level in the general surrounding area. The site itself varies in elevation from marsh bottom, below lake level, to approximately six feet above lake level (**FENOC 2010**, Section 1.2.1.1).

Motor vehicle access to the site is by a two-lane road off State Highway 2, which is a two-lane artery located west of the station (**FENOC 2010**, Section 2.2.2.1). U.S. Highway 80 is about 14 miles south of the site (Figure 2.1-1). The nearest scheduled passenger air service is located 38 miles west, in Toledo (**FENOC 2010**, Section 2.2.2.3). Section 2.9.5 describes local and regional transportation in more detail.

2.1.1 References

FENOC 2010. Updated Safety Analysis Report (USAR) Davis-Besse Nuclear Power Station No. 1 Docket No: 50-346 License No: NPF-3, FirstEnergy Nuclear Operating Company (FENOC), Revision 27, June 2010.

NRC 1996. Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants (GEIS), NUREG-1437, Volumes 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, May 1996.

Figure 2.1-1: Project Area Map, 50-Mile Radius

Figure 2.1-2: Project Area Map, 6-Mile Radius

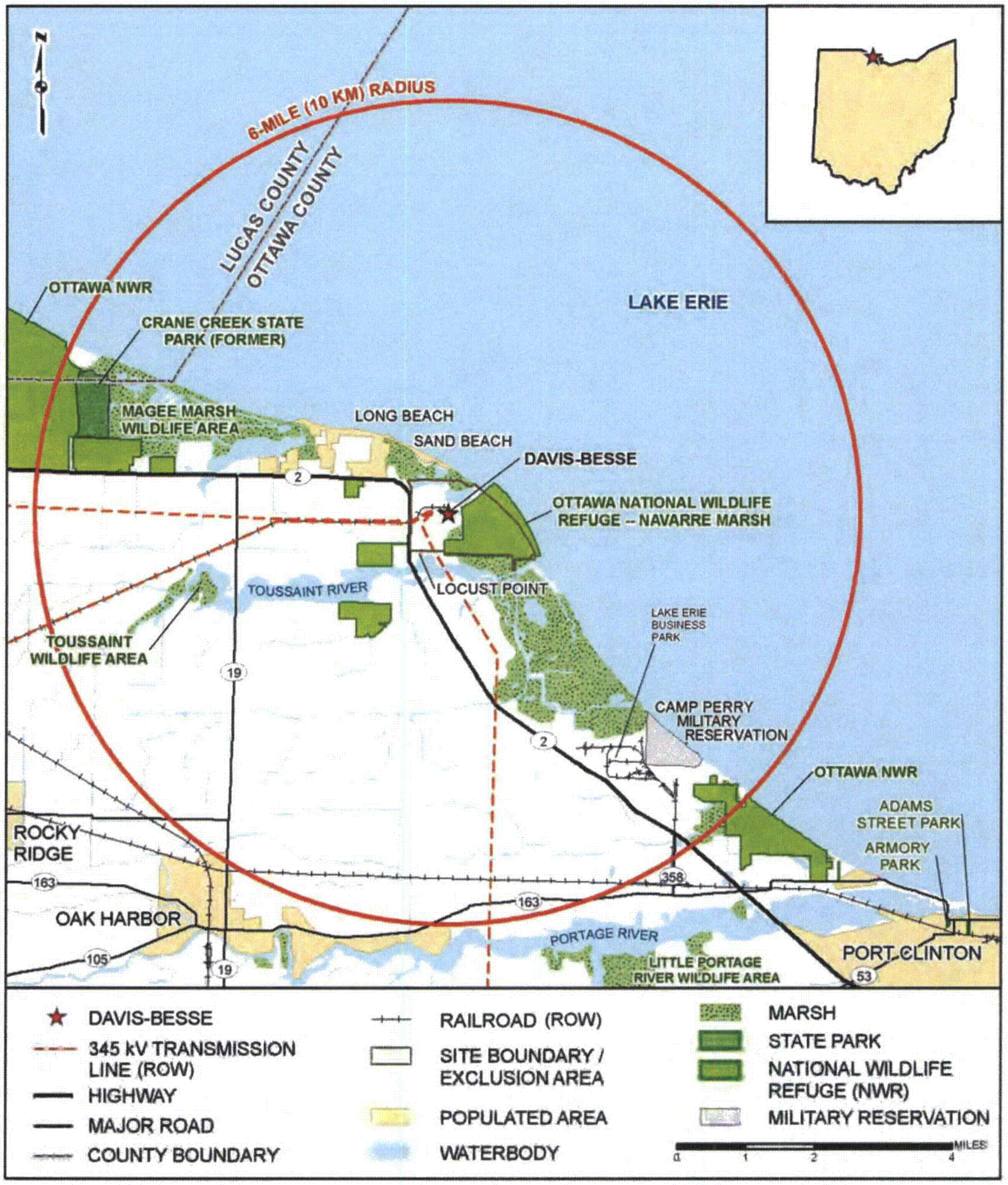
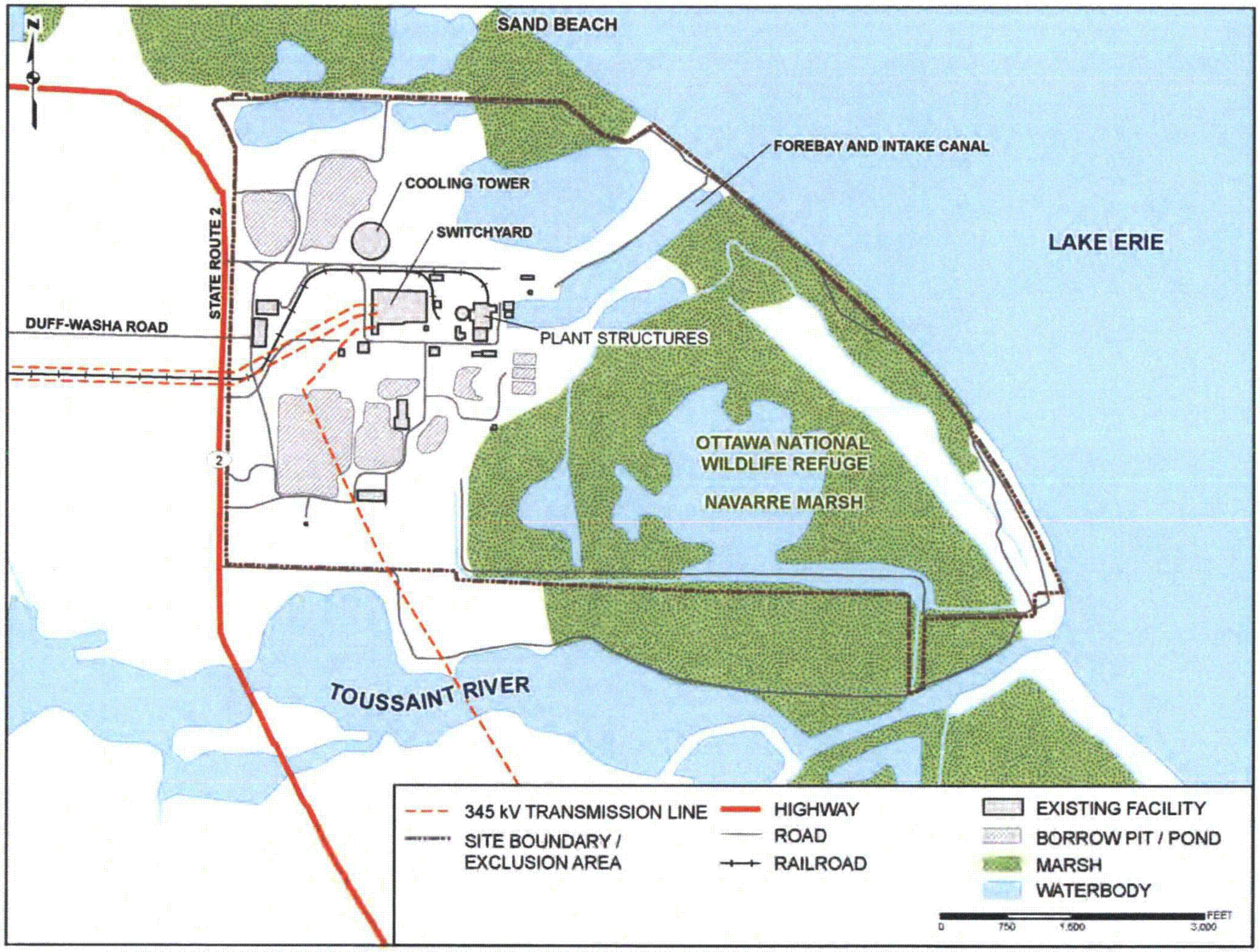


Figure 2.1-3: Site Area Map



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2.2 AQUATIC AND RIPARIAN ECOLOGICAL COMMUNITIES

Davis-Besse is located on the south shore of western Lake Erie. Hydrologic features of the site include Lake Erie surface waters, 733 acres of associated site wetlands, and the nearby Toussaint River and the Cedar-Portage Watershed (**NRC 1975** Section 2.5.1; **ODNR 2007**, Chapter 3; **LELamp 2008**, Section 6.1). Lake Erie and its associated watersheds, like that of the Great Lakes System, represent a vitally important economic and natural resource that supports recreation, fisheries, agriculture, transportation, and industrial processes. The lake also represents an important source of potable water. However, rapid industrial and population growth surrounding the lake have contributed to water quality degradation and eutrophication. As a result, numerous state, federal, and international partnerships have been established to develop resource management goals and to perform long-term ecological monitoring (**GLFC 2007**; **LELamp 2008; USEPA 2004**; **OHLEC 2008**; **NRCS 2005**; **IJC 2005**). Together, these partnerships provide critical information to assess ecosystem health. This environmental report section summarizes aquatic resources associated with western Lake Erie and the Davis-Besse site.

2.2.1 HYDROLOGY AND WATER QUALITY

2.2.1.1 Hydrology

Lake Erie is the smallest of the Great Lakes by volume and second smallest by surface area. The southernmost and most-shallow of the Great Lakes, Lake Erie extends west to east approximately 240 miles and averages about 57 miles wide. Water surface area is approximately 9,906 square miles and its volume is about 116 cubic miles. Average depth is about 60 feet. The lake's surface area represents about 44% of the total land-water area of the Lake Erie Basin which includes 22,720 square miles and parts of New York, Pennsylvania, Ohio, Michigan and Ontario (**Downhower 1988**, Chapter 4; **USEPA 1984**, Pages 3-10; **LELamp 2008**, Section 2.1; **Bolsenga and Herdendorf 1993**, Pages 11-26).

Lake Erie is divided into three distinct topographic basins: the western basin bounded on the east by a series of islands near Port Clinton; the central basin extending approximately 124 miles to the east; and the eastern basin bounded on the west by a shallow sand and gravel bar near Erie, PA (**Herdendorf and Monaco 1988**, Page 30). Average depths in the basins are 24 feet, 60 feet, and 80 feet for the western, central and eastern basins, respectively. Maximum water depth in the western basin is 62 feet (**Bolsenga and Herdendorf 1993**, Pages 11-19).

The hydrologic budget in Lake Erie is determined largely by inflow from the Detroit River, and to a lesser extent by direct precipitation and basin drainage (runoff). The estimated contributions are about 80% from the Detroit River inflow, about 11% from precipitation, and about 9% from basin runoff (**Bolsenga and Herdendorf 1993**, Pages 173-180; **LELamp 2008**, Section 2.1; **Herdendorf and Monaco 1988**, Pages 30-38). Annual precipitation in the Lake Erie region is approximately 34 inches. Within the western basin, the Maumee River is the largest contributor of runoff; approximately 25% of the runoff to the lake is attributed to this river. The Toussaint River enters Lake Erie about 1.75 miles southeast of the station. Outflow from the lake is primarily through the Niagara River.

Current patterns in Lake Erie are influenced by the inflow of water from the Detroit River and gyres created by prevailing winds. In the western basin, flows from the Detroit River predominate and pass along the north shore through the Pelee Passage with some recirculation along the south shore attributed to the islands extending northward from Port Clinton toward Pelee Point in Ontario, Canada. Currents in the central and eastern basins generally occur as gyres circulating either clockwise or counterclockwise depending on the prevailing wind direction. Water levels fluctuate due to seasonal changes in inflow but also due to the prevailing winds that run along the axis of the lake. The average annual water level fluctuation is about 1.2 feet. Changes in lake level have been recorded to increase up to 9.8 feet at Toledo due to northeasterly storms and decrease up to 6.6 feet due to southwesterly winds. Differences in water level between Toledo and Buffalo have been recorded as high as 14 feet due to extreme wind driven seiches (**Bolsenga and Herdendorf 1993**, Page 188-194 and **LELamp 2008**, Section 2.1). At the same time, Lake Erie can experience large wind driven waves. Herdendorf and Monaco suggest that, based on U.S. Army Corps of Engineers (USACE) estimates, waves as high as 12 feet can be obtained in the area of Marblehead Peninsula, reaching up to 3.6 feet along the shoreline (**Herdendorf and Monaco 1988**, Page 38).

2.2.1.2 Water Quality

Water quality in Lake Erie and its western basin reflects the demographics of the surrounding watersheds. About one-third of the total population of the Great Lakes Basin, about 11.6 million people, resides within the Lake Erie watershed. As a result, Lake Erie has been disproportionately affected by urbanization, industrialization, and agriculture (**LELamp 2008**, Section 2.1; **OHLEC 2008**, Introduction). Eutrophication of the lake was first identified as an emerging ecological and human health concern in the 1950s, with the occurrence of noxious algal blooms and oxygen depletion. The increasing discharge of phosphorous and its recycling in sediments were considered to be the main contributors. The accumulation of persistent toxic chemicals, attributed to increased industrialization, was also being observed in water, sediment, fish and wildlife. Subsequent ecological concerns have focused on invasive species such as the zebra mussel that have altered food web dynamics.

Addressing these ecological and human health issues in the Great Lakes led to the formation of an international partnership and the development of a long term strategy expressed in the Great Lakes Water Quality Agreement of 1978 (**IJC 2005**; **LELamp 2008**). Lakewide Management Plans, envisioned for each of the Great Lakes, include a set of performance measures tied to beneficial use criteria. Complementary plans were established by the Ohio Lake Erie Commission and the U.S. Department of Agriculture. Various research institutions and scientific forums were also developed to carry out the required water quality and ecological monitoring and to provide data analyses that allow for continued management oversight and refinement of lake management goals. The current status of the Lake Erie Management Plan (Plan) is found in the Lake Erie Lake Management Report for 2008 (**LELamp 2008**).

While results of these international efforts have demonstrated some progress, water quality impairments still exist for most of the performance criteria established in the Plan. Areas of continued concern include the occurrence of chemical contaminants in biota and sediments, elevated levels of bacteria, invasive species (and impact on biodiversity), habitat loss, degradation of fish populations (and supporting food web) and continued eutrophication. Similar trends have been observed in the terrestrial environment associated with Lake Erie and its basins (**LELamp 2008**).

Despite the legacy issues and ongoing concerns with respect to water quality and ecosystem health, drinking water taken from the lake following treatment meets the primary maximum contaminant levels for finished drinking water (**OHLEC 2004**, Page 14).

Lake Erie monthly average water temperatures recorded at Cleveland between 1961 and 1990 varied between 33° F in February to 74°F in July and August (**NOAA 2009**). At Hatchery Bay, near South Bass Island, water temperatures were reported for various years between 1984 and 1995. Minimum winter water temperature reached 32°F in January and February and 80.6°F in late July and early August (**Beeton et al. 1996**). Comparable water temperature ranges were also reported for long time series recorded at Put-in-Bay and Sandusky Bay (**McCormick 1996a, b**).

Lake Erie water quality data spanning the period 1974 to 2001 were collected in the vicinity of Davis-Besse by various investigators. Following is a summary of the results, demonstrating the seasonal and temporal range of values for the different parameters.

Water quality data were collected in the vicinity of Davis-Besse between 1974 and 1979 as part of a preoperational and operational study (**Reutter et al. 1980**, Pages 49 and 72). Parameters were measured monthly at three stations during ice-out periods.

Parameters were reported as averages for the intake and discharge station for the preoperational and operational periods. Annual average water temperature over the study period varied between 60.8 and 61.5°F. Average dissolved oxygen ranged between 9.1 and 10.0 parts per million (ppm), although variations between 3.0 ppm and 14.1 ppm were reported. Water pH averaged 8.3 standard units (SU) across all stations and varied seasonally between 7.2 and 8.9 SU. Alkalinity ranged between 94 ppm and 96.2 ppm. Transparency (clarity) was low and varied between 1.6 and 1.8 feet. Phosphorus concentrations were relatively high but decreased over time from an average of about 70 parts per billion (ppb) to 40 ppb between the preoperational and operational periods, respectively.

Additional water quality data are found in U.S. Environmental Protection Agency (USEPA) (**USEPA 1984**), Herdendorf and Monaco (**Herdendorf and Monaco 1988**), Bolsenga and Herdendorf (**Bolsenga and Herdendorf 1993**) and OHLEC (**OHLEC 2004**). Bolsenga and Herdendorf (**Bolsenga and Herdendorf 1993**, Pages 251-270) report selected parameters for the period 1967-1982 drawing on data provided by USEPA (Table 2.2-1). Similar to the results reported by Reutter (**Reutter et al. 1980**), Bolsenga and Herdendorf indicate that Lake Erie tends to be alkaline with an average alkalinity of 95 milligrams per liter (mg/l) as $CaCO_3$, ranging from 82.3 ppm in the western basin to 103.9 ppm in the eastern basin (**Reutter et al. 1980**; **Bolsenga and Herdendorf 1993**). Average pH ranged from 8.23 to 8.42 SU across the three basins. Average annual water temperatures in the three basins varied from 63 °F to 58.5 °F. Secchi depth readings showed that water clarity was greater in the central and eastern basins and averaged only 2.6 feet in the western basin which was the most turbid of the three basins, this attributed to the Maumee River inflow. Annual average dissolved oxygen varied between 9.4 and 9.9 ppm in the basins.

Average annual total phosphorous concentrations were 29.1 and 20.7 ppb for the central and eastern basins. No value was reported for the western basin although the concentration of dissolved phosphorus was between two to three times higher than that of the central and eastern basins. Total phosphorus loadings during this study period showed a dramatic decrease of up to 50% due to improvements in sanitary sewage treatment. Despite these improvements, periods of anoxia in the central basin continued to exist. Average chlorophyll a concentrations were greatest (13.5 ppb) in the western basin, reflecting the inputs from the Maumee River, and lowest in the eastern basin (3.1 ppb) (**Bolsenga and Herdendorf 1993**).

OHLEC provides corresponding data for the period 1983-2001 (**OHLEC 2004**, Pages 4-8). During this study period, the concentration of total phosphorus in the western basin ranged between just over 25 ppb to just over 10 ppb and showed a general decrease over time. The 5-year average concentration as of 2001 was 16.2 ppb, just above the 15 ppb target for the western basin. During this same period, average phosphorus concentration in the central basin was 6 ppb. Despite these past improvements, phosphorus loadings have been increasing since the early 1990s attributed largely to agricultural practices (**LELamp 2008; OHLEC 2004**). Corresponding increases in loadings have been observed for nitrate-nitrite.

Water clarity improved during the period 1970 through 1996 in the western basin increasing from approximately two to three feet to between 6 and 7 feet. The improvement resulted in part from the infestation of zebra mussels in 1988. A comparison of water clarity and phytoplankton diatom densities pre- and post-zebra mussel occurrence showed a 100% increase in water clarity, and a corresponding decrease of 86% in diatoms (**Holland 1993**). However, water clarity decreased through 2001 due to increased sediment loads and algal concentrations, and continues to be impaired in certain parts of the lake (**LELamp 2008**, Section 4.2). Increasing sediment loads contributing to this trend appear to be linked to increased drainage basin flows in the major tributaries and the corresponding increases in nutrient loadings.

Bathymetry of western Lake Erie and sediment composition near Davis-Besse were reported by Herdendorf in anticipation of station construction near Locust Point (**Herdendorf 1972 a, b, c**). Depth profiles taken from the shoreline out to about 4,000 feet show the depth increasing gradually to approximately 11 feet at 3,000 feet from shore, the location of the intake crib (**AEC 1973**, Section 3.3.2). Sediment composition was variable but generally had a higher percentage of sand near shore, and tending toward gravel further offshore.

2.2.2 Aquatic Communities

Information describing the ecological characteristics of western Lake Erie in the vicinity of Davis-Besse is available from preoperational and operational studies (**Reutter et al. 1980**) and from research conducted subsequently by various state, federal and international organizations, some for the purpose of monitoring and assessing ecological conditions relative to lake management plans (**Herdendorf and Monaco 1985**; **Bolsenga and Herdendorf 1993**; **LELamp 2008**; **OHLEC 2008; NRCS 2005**; and **GLFC 2007**).

2.2.2.1 General

Lake Erie aquatic community data dating from as early as 1930 were collected in the general vicinity of Davis-Besse by various investigators. Following is a summary of the results, demonstrating the abundance and diversity of the aquatic communities.

The abundance and diversity of aquatic organisms in Lake Erie has been influenced historically by altered habitat conditions. As discussed above, key factors that have impacted the lake's ecological balance include eutrophication, hypoxia, toxics, habitat loss and invasive species. Phytoplankton, as an example, respond to excess concentrations of phosphorus and other nutrients in Lake Erie and form algal blooms. The algae die, settle to the bottom and decompose consuming oxygen in the process. The effect is exacerbated when a hypolimnion occurs separating oxygen rich surface waters from anoxic bottom waters. This problem was most acute in the 1960s and provided the impetus for coordinated efforts to improve water quality and protect ecosystem health (**LELamp 2008**, Section 2.1).

Phytoplankton species composition and abundance were studied from 1974 through 1979 as part of the Davis-Besse preoperational and operational monitoring programs (**Reutter et al. 1980,** Page 31, 57). Among the three groups of phytoplankton, diatoms were most numerous and typically peaked in spring. Mean densities during the preoperational and operational periods were 127,669 cells/gallon (gal) and 521,415 cells/gal, respectively. Monthly densities ranged from 346 cells/gal in June to 1,572,684 cells/gal in May. The dominant species typically included *Melosira, Fragillaria, Asterionella, Stephanodiscus and Synedra* (**Herdendorf and Monaco 1985**, Page 17; **Reutter et al. 1980**).

Green algae were least abundant. Mean densities ranged between 16,758 cells/gal and 58,665 cells/gal during the preoperational and operational study periods. Mean monthly densities varied between 392 cells/gal in April and 452,177 cells/gal in November. The dominant species were *Mugeotia, Pediastrum and Scenedesmus*. Blue-green algae mean densities ranged from 62,919 cells/gal to 223,180 cells/gal in the two study periods and were most abundant in summer (**Reutter et al.1980**). Blue-green algal blooms observed during the mid 1960s, consisting of *Microcystis, Aphanizomenon and Anabena*, were less common in the 1970s (**Herdendorf and Monaco 1985**).

Algae that adhere to substrates, periphyton, are also common in Lake Erie and are most abundant in the littoral zone. A discussion of these algal species is provided by Herdendorf and Monaco (**Herdendorf and Monaco 1985**) who studied the limnology of the island region near Port Clinton. The benthic alga, *Cladophora glomerata*, is known for its formation of massive algal mats in late spring and summer that create noxious odors and foul submerged structures. Excess growth of this species has been linked to increased phosphorus concentrations and hypoxia (**Lorenz and Monaco 1988**, Page 65). Benthic algal species also include diatoms, and green and blue-green algae.

Zooplankton in the western basin of Lake Erie include both herbivores and carnivores from three basic groups: protozoans, rotifers and microcrustaceans (cladocerans and copepods) (**Herdendorf and Monaco 1985**, Page 18; **Reutter et al. 1980**, Pages 36 and 57). Mean densities of rotifers reported by Reutter (**Reutter et al. 1980**) ranged between 858/gal and 442/gal during the Davis-Besse preoperational and operational study periods. Monthly mean densities ranged between 58/gal in November (operational) and 2,619/gal in October (preoperational). The dominant species included *Brachionus, Keratella, Polyarthra and Synchaeta*. Copepods were most abundant in spring and fall during this same study. Mean densities during the preoperational and operational periods ranged between 515/gal and 550/gal, respectively. Mean monthly densities ranged between 92/gal in April and 3,273/gal in May. Calanoid and cyclopoid forms were most common, including their nauplii. Cladoceran mean densities ranged between 254/gal and 296/gal during the two study periods. Mean monthly densities were comparable to those reported by Herdendorf and Monaco (**Herdendorf and Monaco 1985**).

A composite description of the Lake Erie benthic community in the vicinity of Davis-Besse is also provided by Herdendorf and Monaco, (**Herdendorf and Monaco 1985,** Page 25), and Reutter (**Reutter et al. 1980**, Page 64). Typical of benthic macroinvertebrate communities, species composition was determined by the substrate type and included attached and borrowing forms: coelenterates, annelids, arthropods, mollusks and crustaceans were all represented. The burrowing forms were dominated by oligochaetes and chironomid midge larvae. Gastropod snails were found mostly on submerged vegetation. Among the mollusks, the freshwater mussels and fingernail clams were the dominant forms. Crustaceans included the amphipod, *Gammarus fasciatus,* and various forms of water fleas, isopods, ostracods (seed shrimp) and decapods (crayfish). Insects typically included dipterans (true flies) and mayflies. Densities of the four major groups were provided by Reutter (**Reutter et al. 1980**) as part of the Davis-Besse monitoring programs.

A historical perspective on the benthic fauna of western Lake Erie including the invasion by zebra mussels was provided by Manny and Schloesser and Austen (**Manny and Schloesser 1999**; **Austen et al. 2002**). From 1930 to 1961, the average densities of most benthic macroinvertebrates increased dramatically while the mayflies decreased. However, from 1961 through 1982, there were large decreases in gastropods, fingernail clams, and chironomids (midge larvae) and the disappearance of mayflies. As of 1982, the benthic infauna was dominated by oligochaete and polychaete worms, suggesting continued water quality impairment. In 1993, burrowing mayflies began to recover, yet the native unionid freshwater mussel died throughout most of western Lake Erie as a result of competition from the zebra mussel. Recent evidence suggests, however, that the abundance of mayflies in the western basin is increasing (**GLFC 2003**). Information on historical changes in benthic communities of the nearby island region is provided by Fink and Wood (**Fink and Wood 1988**). Similar to the findings of Manny and Schloesser, Fink and Wood report the demise of the mayfly, decreasing numbers of caddisfly species and the dominance of *Gammarus* in the littoral zone (**Manny and Schloesser 1999**; **Fink and Wood 1988**). Monitoring of zebra and quagga mussel densities continues as part of the Lake Erie Management Plan activities (**LELamp 2008**, Section 10.2). Results during 2004 suggest that while the density and mass of zebra mussels has changed little from 1992 to the present, they are now distributed mostly within the western basin. In general, quagga mussels were more abundant than zebra mussels. Mean density of quagga mussels was 235 individuals/square foot compared to 22.4 individuals/square foot for zebra mussels. A detailed discussion of the affects of invasive mussels on energy flow and biodiversity within the Lake Erie benthic community is provided by Austen (**Austen et al. 2002**).

Because most benthic infauna are relatively immobile, they have been used as bioindicators of toxic contaminants in sediments and related impairments. Of particular concern are metals and organic chemicals. Based on the USEPA Lamp study programs, portions of western Lake Erie remain impaired based on sediment contaminant concentrations and indicator species abundance. While concentrations of key contaminants such as polychlorinated byphenyls (PCBs), dioxin, chlordane, mercury and poly aromatic hydrocarbons (PAHs) have been steadily declining over the past two decades, most remain above their probable effect concentrations near industrial-urban areas (**LELamp 2008**, Section 5.0).

2.2.2.2 Fisheries

Changes in the species composition and abundance of Lake Erie fishes over the last century have been attributed to a number of stresses, including exploitation, habitat deterioration, contaminants and invasive species (**LELamp 2008**; **GLFC 2003**; **Reutter and Hartman 1988**, Page 163). The perturbation of trophic structure led to corresponding impacts on standing fish stocks. Several native species such as the lake trout, lake sturgeon, lake herring and whitefish have been nearly extirpated. The abundance of key recreational and commercial species such as walleye and yellow perch had declined significantly. Despite these historical impacts, Lake Erie maintains a substantial fishery, and long-term management goals have been established by the Great Lakes Fishery Commission to restore and maintain stability of the standing fish stocks (**GLFC 2003**).

The Lake Erie basin supports an estimated 143 fish species; 95 species are present in the lake. Thirty-four species (24%) of fish in Lake Erie proper are nonindigenous (**Austen et al. 2002**). Thirty-five species have been harvested and 19 are considered commercially significant (**Reutter and Hartman 1988**; **Van Meter and Trautman 1970**). Key commercial and recreational species include yellow perch, walleye, smallmouth bass, steelhead trout, lake whitefish and white bass. The abundance of these and other species is monitored by the Ohio Department of Natural Resources (**ODNR 2008**). Trawl surveys have been conducted in the western basin during summer and fall since 1990. Up to 38 locations are sampled at four depths. Gill nets were also deployed at seven historic sites. Corresponding samples were collected in the central basin.

Information on growth and diet are also collected. Hydroacoustic surveys are conducted to assess forage fish abundance.

Walleye abundance (mean catch/acre of age-1 and older fish in the western basin) in summer trawls ranged between 0.04 in 1996 and 2007 and 7.5 in 2004. Except for 2004, catches in the last 5 years were below the long-term average of 1.6 fish/acre. Walleye catches in fall trawls ranged between 0.0 in 2007 and 4.1 in 2004. The long term mean was 0.85 fish/acre. Yellow perch abundance ranged between 1.6 fish/acre in 2003 and 85.3 fish/acre in 2004. The long-term average catch was 22.2 fish/acre. Catches during 2005-2007 were well below the long term average. Similar trends were found in Fall catches of yellow perch. The long term catch of white bass in summer trawls was 33.8 fish/acre and was lowest in 2007 at 3.3 fish/acre. Fall abundance of white bass averaged 2.2 fish/acre and was highly variable in recent years. The abundance of freshwater drum was comparatively high and averaged 53.3 fish/acre over the study period, and was consistently higher between 2000 and 2004. The fall average abundance of drum was 32.5 fish/acre (**ODNR 2008**, Section 6).

Monitoring performed as part of the Davis-Besse monitoring programs through 1979 yielded a total of fifty-one fish species in the Locust Point area (**Reutter et al. 1980**, Page 44, 66). Gillnet, trawl and seine samples were typically dominated by seven species: alewife, emerald shiner, freshwater drum, gizzard shad, spottail shiner, white bass and yellow perch. Together these species contributed over 90% of the catch. Walleye were not commonly found. Yellow perch, gizzard shad, white perch, carp and spottail shiner were the most common fish species caught in gill nets set at Locust Point. Yellow perch were consistently the most abundant. A total of 20 species of fish were captured in trawls and included several benthic species such as bullheads and channel catfish. Species composition in seines was similar to that found in gill net catches.

Records of sport catch in Ohio waters by private and charter boats are available for the period 1975-2007 (Table 2.2-2). Some of the earlier catches represent averages over two or more years but are generally recorded as annual catches between 1995 and 2007. Total annual catch (x1000) of walleye during this period ranged between 374 in 2005 and 1,790 fish in 1998. Catches in 2006 and 2007 were 1,195 and 1,414, respectively. Catch rates in 2006 and 2007 averaged 0.68 fish per angler hour and were the highest harvest rates for the period of record (**ODNR 2008**, Section 4).

Lake-wide commercial and sport harvest of walleye during 1975 through 2007 is shown in Figure 2.2-1. Total lake-wide harvest of walleye peaked during the late 1980s at about 10,000,000 individuals and declined thereafter, although increases were observed in 2005-2007. The total estimated lake-wide harvest was 4.67 million fish in 2007. Harvest per-unit effort also increased during these later years to levels last seen during the 1980s. The lake-wide population estimates of walleye in Lake Erie show similar trends over the period 1978-2008 (Figure 2.2-2).

Yellow perch sport catch in Ohio waters varied between 248 (x1000) fish average per year in 1990 - 1994 and 4,174 (x1000) fish in 2003 (Table 2.2-2). Catches in recent years appeared to be consistent with the long-term average. Long-term trends in western Lake Erie yellow perch population size are shown in Figure 2.2-3. Trends across the various basins of the lake show similar results with decreasing population size during the early 1990s and increases in recent years, but not surpassing levels seen historically. Sport catch of smallmouth bass appears to have declined in recent years. Total private and charter boat catch varied between 2.7 (x1000) fish in 2007 to a high of 77.4 (x1000) fish in 1995. Catches between 2004 and 2007 were less than 7.6 (x1000) fish. Corresponding harvest rates were also low. (**LEC 2008b**)

Commercial harvests of fish taken from Lake Erie are available through the National Marine Fisheries Service (**NMFS 2009**). Over the period 1971 through 2005, annual walleye catches varied between a low of 33 fish in 2002 and a high of 153,595 fish in 1973 (Table 2.2-3). Harvests of yellow perch taken in Lake Erie varied from a peak 3,157,417 fish in 1980 to 235,078 fish in 1984. Catches were greatest during the 1970s, declined during the 1980s and more recently have increased reaching 1,586,154 fish in 2005. White bass commercial harvests for Lake Erie varied from 3,249,763 fish in 1980 and 95,466 fish in 1995. Catches were highest during the 1970s and have been consistently lower since, although harvests in Ohio consistently exceeded 300,000 fish from 2004 through 2007. Freshwater drum harvests were also higher during the 1970s and have consistently decreased since. Only 253,086 fish were harvested in 2002 compared to a peak of 1,332,971 in 1979. Similar trends were observed in other commercial catches of fish landed in Ohio, as listed in Table 2.2-4.

Affecting the quality of the sport and commercial fisheries are consumption health advisories attributed to toxic contaminants. Studies of toxic chemical concentrations in sport fish from the Canadian waters of Lake Erie from 1976-2000 continue to show elevated levels of mercury, PCBs and other contaminants although concentrations continue to decline. Mean mercury concentrations in 12 in. white bass decreased from 0.22 ppm in 1976-80 to 0.13 ppm in 1996-2000. Similarly, mean mercury concentrations in 18 in. walleye have decreased from 0.30 ppm to 0.12 ppm over the same time period. Only fish larger than 16 in. exceeded the 0.45 ppm consumption advisory. PCB concentrations in channel catfish have also decreased over the same study periods. PCB concentrations (3,225 ppb) in 1981-1985 had decreased to 1143 ppb in 1996-2000. However, PCB concentrations in benthic feeding species such as carp and catfish continue to exceed the consumption guideline of 500 ppb (**LELamp 2008**, Section 10.4).

Another factor affecting fish species composition and abundance in Lake Erie has been the invasion of nonindigenous fish species. As stated above, it is estimated that the resident fish community now includes approximately thirty-four nonindigenous species. Approximately 40% of the commercial catches in Ontario in the late 1990s were nonindigenous fish species (**Austen et al. 2002**). Changes have occurred within the various trophic levels. Historically, lake herring, sculpins and shiners dominated the forage fish community. Many of the sculpin species are no longer found. Alewife, rainbow smelt, gizzard shad and round gobies now dominate.

A review of the U.S. Fish and Wildlife Service Critical Habitat portal indicated that the Locust Point area of Lake Erie near Davis-Besse does not contain critical habitat for any threatened or endangered fish species (**USFWS 2009a**). Notwithstanding the historical impacts to Lake Erie and its fisheries described above, none are related to Davis-Besse operation and the lake continues to maintain a substantial fishery, both in the species composition and abundance.

2.2.2.3 Entrainment and Impingement.

Year class strength of most fish species is determined within the egg and larval stage. As a result, the abundance and distribution of ichthyoplankton relative to the location and amount of water withdrawal by cooling water intakes can influence the potential impact of entrainment on fish populations. Studies of entrainment and the abundance of ichthyoplankton relative to the Davis-Besse Station and other steam-electric stations located on Lake Erie were performed by the Ohio State University Center for Lake Erie Area Research from 1974 through the first few years of Davis-Besse operation , as requested by the Ohio Department of Natural Resources (**Reutter et al. 1980**) by the U.S. Environmental Protection Agency (**Cooper et al. 1981**), and by the Toledo Edison Company (**Reutter 1981a**).

In general, emerald shiner, common shiner, freshwater drum, gizzard shad, white bass and yellow perch were the majority of the larval fish species collected in the Davis-Besse intake area, although gizzard shad were clearly the most abundant. Larval densities were highest in late May and June. The relative abundance of yellow perch and walleye was highly variable from year to year. During the period 1976-1980, the percent composition of larval yellow perch ranged between 2% in 1978 to 70% in 1975. Walleye percent composition varied from 0.2% in 1976 and 1979 to 22% in 1980. In 1980, mean densities (number/3531 cubic feet (ft^3)) of the abundant species, freshwater drum, gizzard shad, white bass and yellow perch, were 130.67, 189.18, 23.8 and 91.0, respectively. Peak density estimates in 1979 based on a composite of stations off Locust Point were as follows: gizzard shad, 200.4/3531 ft^3; yellow perch, 66.1/3531 ft^3; emerald shiner, 7.6/3531 ft^3. Estimates of equivalent female adult losses due to entrainment in 1980, based on mean adult fecundity, were very low, i.e., 71 gizzard shad, one (1) walleye, and 153 yellow perch (**Reutter 1981a**). Reutter concluded that there was no indication that the Davis-Besse intake location was a significant spawning area that could be detrimentally impacted by the operation of the facility, and there was no indication that the activities of the plant, including the thermal discharge, have significantly altered the populations of the local larval fish species (**Reutter et al. 1980**, Page 72). Reutter also states that the research and other research performed by the authors has indicated that the design features at Davis-Besse, i.e., cooling tower, off-shore intake, closed intake canal, bottom intake, and a high velocity discharge nozzle, may be the optimal design features to minimize aquatic environmental impacts due to cooling water intakes and thermal discharges (**Reutter et al. 1980**, Page 79).

An assessment of entrainment impacts from power plants distributed throughout Lake Erie was performed by Cooper (**Cooper et al. 1981**). Data included samples collected in 1975-1977 in the western basin and 1978 in the central basin. A total of 22 larval fish taxa was collected in the western basin. Dominant species included gizzard shad (87% of the catch) followed by rainbow smelt, whitefish, carp, white bass, yellow perch, sauger, walleye and freshwater drum. The data shows that the percentage of fish entrained by the Davis-Besse cooling water intake as compared to three other Lake Erie western basis generating stations is a small fraction (i.e., 6% or less by fish type) of total fish entrainment (**Cooper et al. 1981**, Page 108).

More recently, McKenna (**McKenna et al. 2008**) studied the relationship between larval fish assemblages in West-Central Lake Erie and habitat type. Ichthyoplankton species composition, abundance and distribution were examined in the vicinity of the major river mouths. Samples were collected in 2000-2002 from April through September. A total of 26 fish species was recorded. Fourteen were found in each year of the study. Species composition and seasonal occurrence were similar to that found in earlier studies. White bass larvae were most common in April, percids in May and June, and cyprinids (shiners) in summer.

Each of the studies discussed here demonstrates that the occurrence of fish larvae and their vulnerability to entrainment is limited to a very short period. While walleye are known to spawn over the offshore reefs near Locust Point (**ODNR 2007**, Page 131), the relative abundance of larval walleye in entrainment samples was low (**Cooper et al. 1981**).

Samples of fish impinged on the Davis-Besse traveling screens were collected at the request of Toledo Edison during 1980 (**Reutter 1981b**). Estimates of total impingement were extrapolated from periodic sampling by normalizing impingement counts to fish impinged/hour. Total 1980 estimated impingement was 9,056 fish. Goldfish and gizzard shad dominated the impingement samples and were most commonly impinged during winter. Over half (51%) of the annual impingement occurred during January, and this January total (4,626) was composed primarily of goldfish (53.5%) and gizzard shad (37.0%). Other species that occurred but at much lower numbers included yellow perch, shiners and freshwater drum. The number of yellow perch estimated to have been impinged was 750 fish, compared to the yellow perch sport and commercial harvest from Ohio waters of 22,248,000 fish. During this same period, 45 white bass were impinged compared to the white bass sport and commercial harvest of 3,909,000 fish. When compared to the sport and commercial harvest of the key species from Ohio waters during the study period, impingement of fish in the Davis-Besse cooling water intake was judged to be insignificant.

Low entrainment and impingement at Davis-Besse are attributable to the use of closed cycle cooling (average intake flow of 21,000 gallons per minute (gpm)) and low intake velocities (< 0.25 fps) (**AEC 1973**, Pages 3-6).

2.2.2.4 Riparian zone

The Lake Erie riparian zone at the Davis-Besse site is one of transition from shoreline beach, to a beach ridge community, a hardwood swamp zone, extensive wetlands and then to upland. The shoreline beach consists of a sand-shell mixture and is considered to be stable "non-critical erosion area, not protected" (**AEC 1973**, Section 2.5.1). The beach ridge plant community consists of several grass species, willow, and sumac. Dominant plants of the hardwood swamp include cottonwood, black willow, hackberry, sycamore, sumac and river-bank grape. The largest freshwater marsh on site (about 733 acres) is the Navarre Marsh which is part of the larger Ottawa National Wildlife Refuge. A series of dikes and pumps are employed to maintain adequate water levels and to manage vegetation and species composition. The marsh is typical of palustrine systems that are flooded seasonally. Vegetation consists mostly of rooted herbaceous hydrophytes (**USFWS 2009b**). The Navarre marsh vegetation includes cattail, soft-stem bulrush, white water lily, milfoil, sago pondweed and curly-leafed pondweed (**AEC 1973**, Pages 2-6, 2-28, 2-40, 4-6).

2.2.3 References

<u>Note to reader</u>: This list of references identifies web pages and associated URLs where reference data were obtained. Some of these web pages may likely no longer be available or their URL addresses may have changed. FENOC has maintained hard copies of the information and data obtained from the referenced web pages.

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Table 2.2-1: Mean Chemical Composition of Lake Erie and Connecting Waterways (1967-1982)

Parameter Units		St. Clair River	Lake St. Clair	Detroit River	Western Lake Erie	Central Lake Erie	Eastern Lake Erie	Niagara River
Water Temperature	°F	53.2	65.9	58.2	63.1	58.7	58.5	59.7
Secchi depth	ft	1.3	4.9	3.3	2.6	9.8	14.1	---
Dissolved oxygen (D.O.) ppm		10.4	9.5	9.3	9.8	9.4	9.9	9.7
D.O. percent saturation	%	97.4	102.0	91.9	98.1	90.6	96.6	98.4
Conductivity @ 25 °C	μmhos/cm	329	224	256	282	298	304	330
Dissolved solids	ppm	142.7	134.6	140.3	193.7	211.2	197.6	169.4
Suspended solids	ppm	21.62	12.14	15.42	19.86	6.63	5.32	17.92
Alkalinity, total	ppm	91.6	81.6	83.4	82.3	89.8	103.9	95.9
Alkalinity, phenolphthalein pH	ppm	---	---	---	4.2	3.7	---	7.3
pH SU		8.09	8.27	8.03	8.42	8.23	8.26	7.83
Calcium, total	ppm	51.2	29.1	29.8	34.4	39.7	31.3	43.6
Magnesium, total	ppm	18.2	7.6	7.5	7.6	9.5	8.8	9.9
Potassium, total	ppm	3.2	1.0	1.0	1.2	1.4	1.3	1.7
Sodium, total	ppm	47.4	4.9	6.1	8.9	10.1	9.2	13.3
Chlorides, total	ppm	20.1	8.1	17.2	---	24.4	21.6	27.7
Sulfates, total	ppm	16.6	16.7	16.1	32.7	25.7	25.5	30.1
Fluoride, total	ppm	0.12	0.12	0.11	0.24	0.16	0.20	0.25
Silica, dissolved	ppb	1.11	0.72	0.83	---	---	0.32	0.19
Ammonia, dissolved	ppb	0.018	---	0.047	0.061	0.023	0.017	---
Nitrate + nitrite, dissolved	ppb	0.290	---	0.300	0.325	0.165	0.263	---
Phosphorus, total	ppb	---	44.5	---	---	29.1	20.7	---
Phosphorus, dissolved	ppb	11.9	8.1	33.8	29.3	11.8	8.1	---
Phosphorus, ortho	ppb	12.2	---	12.1	9.2	5.8	3.4	---
Chlorophyll a ppb		11.9	4.7	3.4	13.5	5.6	3.1	---

Source: **Bolsenga and Herdendorf 1993**, Pages 251-270

Table 2.2-2: Sport Harvest of Selected Fish Species in Western Lake Erie, 1975-2007 (thousands of fish)

Year	Species			
	Walleye	White Bass	Yellow Perch	Smallmouth Bass
1975-77	937	173	6,567	21.2
1978-79	2,424	--	--	--
1980-84	2,520	312	7,982	33.8
1985-89	3,496	166	4,906	20.5
1990-94	1,378	28	1,242	25.6
1995	1,161	19	2,838	77.4
1996	1,442	31	4,020	30.7
1997	929	36	3,464	32.8
1998	1,790	49	3,708	55.7
1999	812	45	3,262	67.8
2000	674	71	3,062	28.0
2001	941	83	2,642	25.1
2002	516	72	3,290	22.4
2003	715	23	4,174	35.0
2004	515	26	2,603	5.9
2005	374	79	2,593	5.2
2006	1,195	93	3,173	7.6
2007	1,414	89	2,817	2.7

Source: **ODNR 2008**

Table 2.2-3: Commercial Harvest, in Numbers of Fish, for Selected Fish Species Taken from Lake Erie during 1971 through 2005

Year	Species					
	Walleye	Yellow Perch	Whitefish	White Bass	White Perch	Freshwater Drum
1971	55,525	2,641,392	114	996,333	--	838,863
1972	91,215	1,917,615	554	770,503	--	917,371
1973	153,595	1,887,321	2,390	2,424,667	--	999,754
1974	113,136	2,376,685	758	2,912,884	--	694,038
1975	127,053	1,914,326	681	1,691,852	--	853,832
1976	69,032	1,885,272	28	1,523,579	25	1,034,677
1977	72,487	2,868,959	28	1,121,201	--	833,458
1978	69,493	2,580,025	1,077	1,732,218	2	1,214,939
1979	101,873	3,147,031	99	1,968,538	53	1,332,971
1980	80,505	3,157,417	2,396	3,249,763	186	1,063,793
1981	66,158	2,422,699	2,274	1,134,536	3,882	1,281,724
1982	68,072	57,314	347	726,804	28,404	1,064,553
1983	79,380	387,748	2,617	864,901	120,682	1,006,962
1984	84,851	235,078	481	980,896	206,367	735,968
1985	131,322	349,963	953	1,350,486	300,358	669,290
1986	14,617	270,390	2,252	729,930	346,724	798,790
1987	14,618	588,442	16,274	474,523	422,039	976,647
1988	12,223	996,187	15,424	144,706	593,992	710,775
1989	9,542	1,926,620	42,013	558,100	607,863	508,929
1990	10,190	1,765,886	123,707	398,226	851,228	658,225
1991	10,532	858,049	336,049	446,122	1,021,149	514,470
1992	9,779	396,635	228,405	383,002	865,402	621,922
1993	29,567	381,441	373,185	227,080	354,901	809,934
1994	28,163	670,282	404,844	366,698	419,395	761,460
1995	41,145	473,245	225,233	95,466	412,702	750,996
1996	81	632,641	51,416	103,603	188,029	600,211
1997	193	774,729	29,028	358,196	259,511	714,839
1998	417	586,754	45,459	236,230	119,647	578,764
1999	229	700,936	48,292	221,562	131,519	359,659
2000	186	959,368	41,475	319,455	182,583	429,227
2001	73	1,042,006	47,639	227,199	155,982	288,199
2002	33	1,413,030	6,564	165,496	270,422	253,086
2003	129	1,501,939	13,337	318,413	312,638	262,004
2004	300	1,588,901	10,620	360,635	387,617	297,708
2005	830	1,586,154	5,176	349,152	432,647	441,975

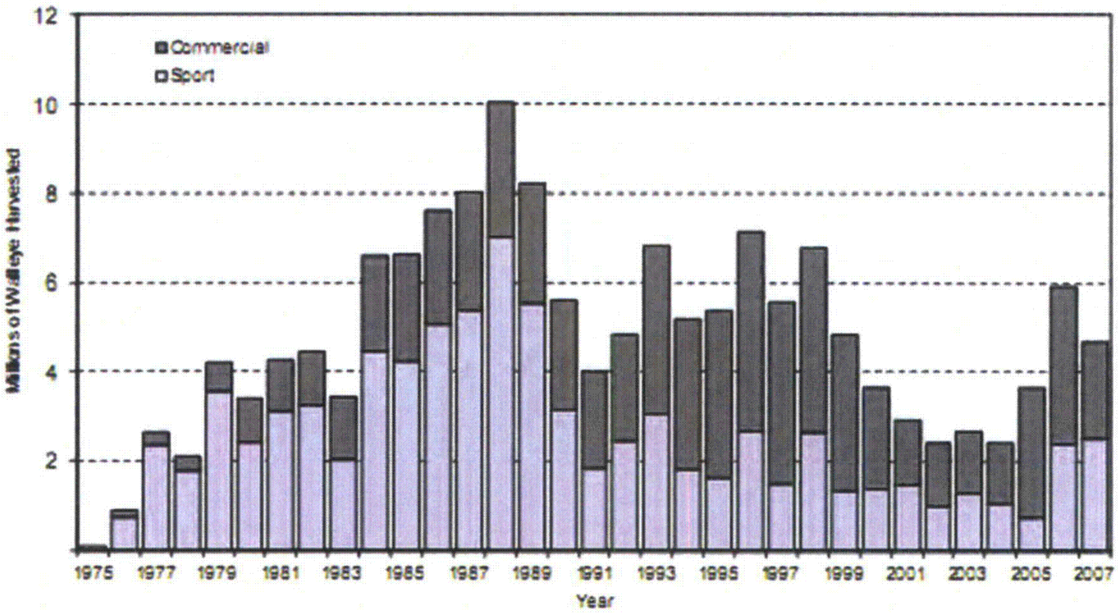
Source: **NMFS 2009**

Table 2.2-4: Annual Commercial Harvest (pounds) from Ohio Waters of Lake Erie, by species, 1998 - 2007

Year	Buffalo	Bullhead	Burbot	Carp	Channel Catfish	Freshwater Drum	Gizzard Shad	Goldfish	Quillback	Suckers	White Bass	White Perch	Whitefish	Yellow Perch
1998	295,904	17,897	1,458	1,336,450	302,056	553,659	172,425	7,992	226,603	50,785	234,791	118,946	41,990	580,151
1999	258,160	24,502	1,145	1,111,504	317,642	358,714	105,068	20,726	170,988	32,415	221,443	131,308	47,622	697,332
2000	162,477	41,695	78	956,218	260,512	428,660	2,809	19,473	140,183	30,195	317,336	182,254	41,472	962,841
2001	257,621	24,106	47	857,694	322,488	284,883	1,970	18,837	149,549	41,040	226,664	155,555	47,639	1,089,247
2002	281,955	23,409	59	523,539	311,824	248,567	545,151	10,625	170,096	32,641	161,664	269,512	6,539	1,438,215
2003	278,544	21,815	192	582,035	319,378	261,068	45	31,406	227,195	15,469	318,327	312,240	13,244	1,505,840
2004	234,673	11,005	857	469,059	271,627	298,336	85,540	23,834	195,931	30,836	358,810	386,800	10,529	1,577,113
2005	230,426	17,012	363	340,399	310,115	438,589	219,800	35,396	363,818	41,763	347,657	428,822	4,613	1,563,200
2006	263,396	25,118	305	271,190	385,134	411,840	195	58,812	250,052	33,233	483,314	655,551	29,795	1,050,614
2007	268,884	25,790	47	322,323	341,843	320,747	55,259	29,148	211,208	17,165	334,721	573,996	41,554	1,950,661
Mean	253,204	23,235	455	677,041	314,262	360,506	118,826	25,625	200,562	32,554	300,473	321,498	28,500	1,241,521

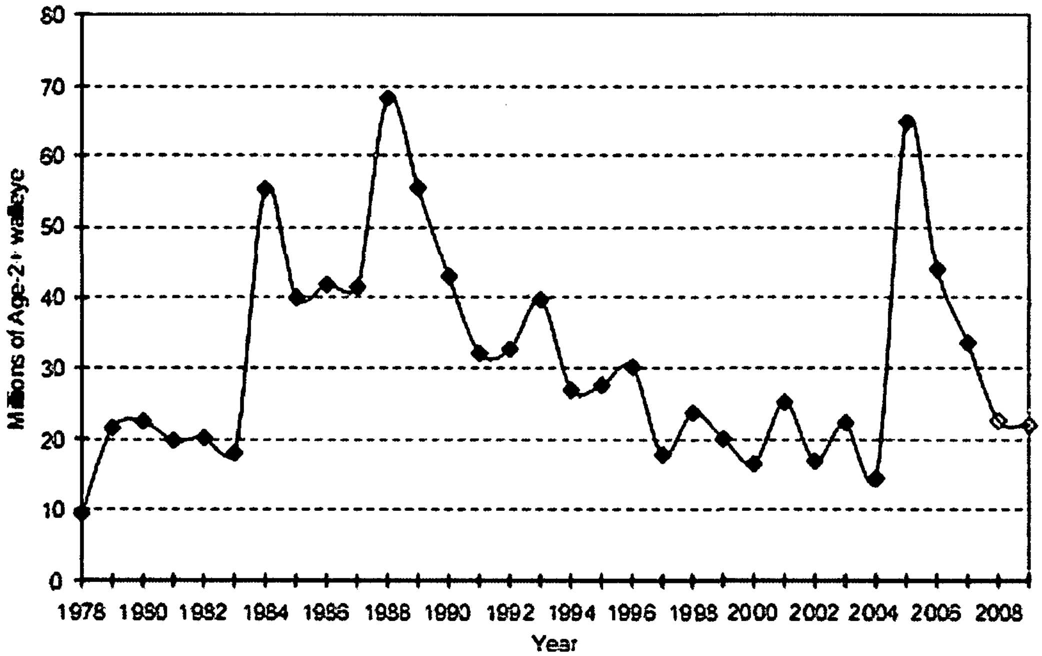
Source: **ODNR 2008**

Figure 2.2-1: Lake-wide Harvest of Lake Erie Walleye by Sport and Commercial Fisheries, 1975-2007



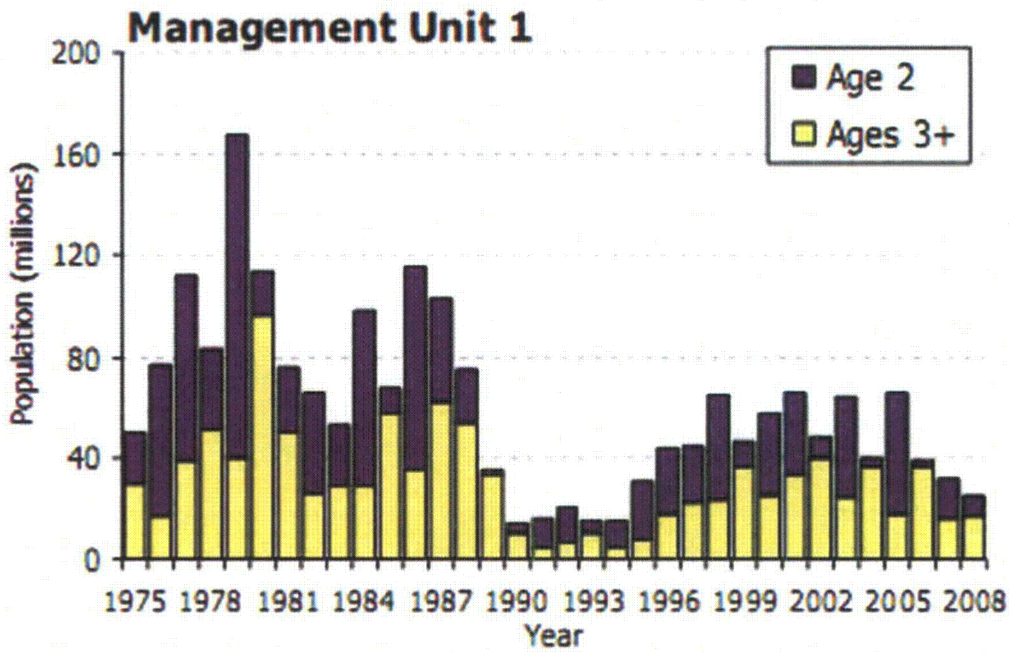
Source: LEC 2008a

Figure 2.2-2: Abundance of Lake Erie Walleye from 1978-2007 (Two Additional Years are Forecasted)



Source: **LEC 2008a**

Figure 2.2-3: Western Lake Erie (Great Lakes Fishery Commission Management Unit 1) Yellow Perch Population Estimates, 1975-2007 The Estimate for 2008 is Projected



Source: LEC 2008b

2.3 GROUNDWATER RESOURCES

The Davis-Besse site is underlain by glaciolacustrine and glacial till deposits, which overlie sedimentary bedrock. The surficial deposits, which are dominantly silty clay, have very low permeability. Site bedrock consists of the Tymochtee formation underlain by the Greenfield formation. These formations consist of nearly horizontal beds of argillaceous dolomite with interbeds of shale, gypsum and anhydrite to a depth of at least 200 feet below ground surface. (**FENOC 2010**, Section 2.4.1.2.3)

The presence of the low permeability surficial deposits has produced an artesian groundwater condition in the site vicinity bedrock. This effect is the result of the surficial deposits acting as an aquiclude to the underlying water-bearing carbonate bedrock, and the influence of Lake Erie's water table producing a potentiometric surface above the water-bearing zone. The potentiometric surface of the confined water-bearing zone is generally a few feet above the level of Lake Erie, indicating that groundwater flow at the site is generally east to northeast, towards the lake and adjacent marshes, with a gradient of approximately 2 feet per mile, which is similar to the surface water gradient in the area. Groundwater elevation fluctuations historically correlate to lake level fluctuations. (**FENOC 2010**, Section 2.4.13.2.3) Assuming heterogeneous hydraulic conductivity of 1×10^{-2} centimeters per second for the bedrock, the maximum value obtained in field tests, the groundwater flow in the Davis-Besse site vicinity is calculated to be approximately 83 feet/year (**FENOC 2010**, Section 2.4.13.3). The groundwater at the site discharges primarily into Lake Erie and the adjacent marshes to the east/northeast (**ERM 2008**, Section 5.0).

Davis-Besse does not use groundwater at the site for plant operations (**FENOC 2010** Section 2.4.13.1.5). Groundwater use in the site vicinity is limited due to the naturally poor water quality exhibited by the carbonate water-bearing zones. There are no identified drinking water wells within 5 miles of the site (**ERM 2007**, Section 3.4). Local residents obtain drinking water from the Carroll Township Water Treatment Plant, which uses surface waters from Lake Erie (**ERM 2007**, Section 3.4). The intake for this water treatment facility is located approximately three miles northwest of the Davis-Besse site. Privately owned wells within 2 to 3 miles of the site are used for farm irrigation and sanitary purposes, and not used as drinking water sources (**ERM 2007** Section 3.4).

The groundwater at the plant site is characterized by strong hydrogen sulfide odors resulting from naturally occurring interaction with local deposits of gypsum and anhydrite. Naturally high levels of carbonate and total dissolved solids cause this aquifer to be unsuitable for use as drinking water (**ERM 2007**, Section 3.4). The Ohio Environmental Protection Agency (OEPA) indicated the absence of any sole-source aquifer in the plant region (**OEPA 2005**).

The historic groundwater monitoring network at the Davis-Besse site consisted of 78 monitoring wells, of which 54 (27 couplets) remain functional (**ERM 2008**, Section 5.0).

The couplets are nested wells screened in bedrock units designated for the site as the upper dolomite and the lower dolomite. These wells were installed during plant construction to monitor groundwater conditions.

In June 2007, Davis-Besse implemented a plan to conform with the voluntary policy of the Nuclear Energy Institute (NEI) Groundwater Protection Initiative (**NEI 2007**). Selected existing monitoring wells were sampled to determine the necessity and location of additional monitoring wells as needed to characterize and monitor the groundwater conditions at the site. In August 2007, 16 new monitoring wells were installed in six distinct locations (**ERM 2008**, Section 3.3). Five of the locations provide nested monitoring wells screened in three distinct zones: the base of the glacial till, the upper dolomite, and the lower dolomite. One of these nested locations is located as a background well up-gradient of the plant power block on the southwest side of the site. The other four nested locations are in the northeast portion of the site, down-gradient from plant structures. Three of these down-gradient wells are located near the extreme northeast corner of the site, allowing for determination of down-gradient offsite contaminant migration. The sixth location is a monitoring well screened only in the glaciolacustrine/glacial till and is located to the northeast and down-gradient from the power block. Historical and 2007-installed well locations are shown in Figure 2.3-1.

Concentrations of gamma-producing radionuclides were below the minimum detection concentration (MDC) in all groundwater samples analyzed between 2007 and 2009. In early 2010, five of seven historic wells showed tritium levels slightly greater than the plant action level of 2,000 pCi/l. Another well, MW-105A, which has been on a slow increasing trend since the spring of 2009, had a tritium level of 4,158 pCi/l. As a result, FENOC is pursuing a root cause approach to identify the source of the tritium in the wells. No tritium concentrations have been detected at or above the USEPA drinking water limit of 20,000 pCi/l (40 CFR 141.66).

Analysis results from three periods of groundwater sampling performed in 2007 revealed the following (**ERM 2008**, Section 5.0). July tritium concentrations above the plant action level of 2,000 picoCuries per liter (pCi/l) occurred in three historical down-gradient wells screened in the upper dolomite. The highest concentration from this sampling period was 7,535 pCi/l in the upper dolomite at a down-gradient well. September groundwater samples from the wells screened in the soil had a range from below the MDC, under about 200 pCi/l to 1,832 pCi/l, with three wells displaying tritium concentrations above background levels. Background tritium levels have been statistically determined by up-gradient groundwater sampling and sampling of Lake Erie waters to be between 178 and 348 pCi/l. Samples in the upper dolomite had a range from the MDC to 3,149 pCi/l, with none of the new monitoring wells having tritium concentrations outside the range of background levels. Samples from the lower dolomite included three wells with tritium concentrations above background levels, but none of these were the new down-gradient monitoring wells. September sampling showed a decrease in tritium concentrations from the June and July samplings.

Sampling in 2008 showed no wells with tritium concentrations above the plant action level. Sampling in 2009 resulted in one well above the plant action level with a tritium concentration of 2,352 pCi/l. During the same 2009 sampling period, six well locations had tritium values below the MDC, with the remainder showing tritium levels below the plant action level.

Concentrations of gamma-producing radionuclides were below the MDC in all groundwater samples analyzed between 2007 and 2009. No tritium concentrations have been detected at or above the USEPA drinking water limit of 20,000 pCi/l (40 CFR 141.66).

2.3.1 References

Note to reader: This list of references identifies a web page and associated URL where reference data were obtained. This web page may likely no longer be available or its URL address may have changed. FENOC has maintained hard copies of the information and data obtained from the referenced web page.

FENOC 2010. Updated Safety Analysis Report (USAR) Davis-Besse Nuclear Power Station No. 1, Docket No: 50-346, License No: NPF-3, FirstEnergy Nuclear Operating Company (FENOC), Revision 27, June 2010.

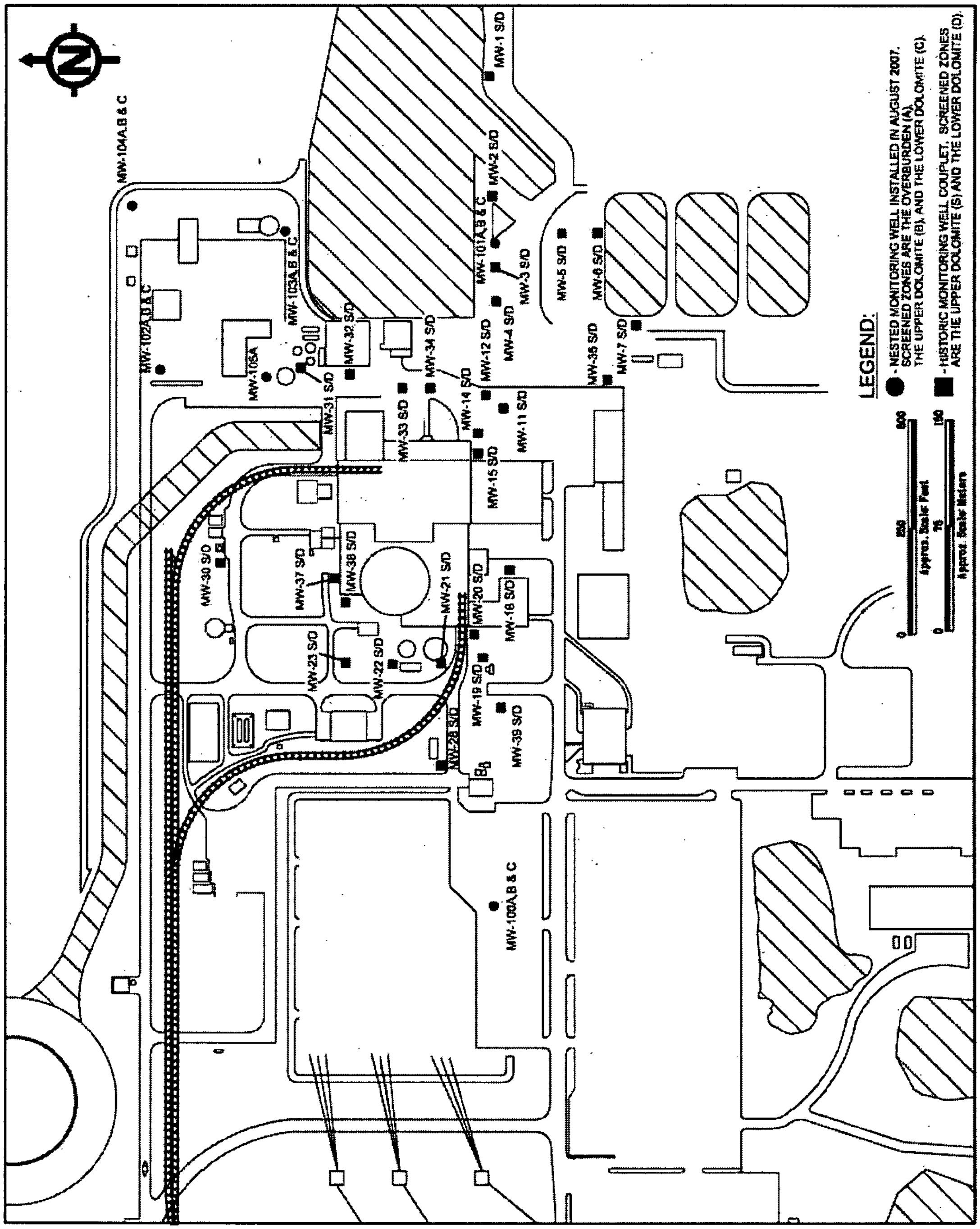
ERM 2007. Groundwater Flow Characteristics Report Davis-Besse Nuclear Power Station, Oak Harbor, Ohio, Environmental Resources Management (ERM) Reference 55194, FirstEnergy Service Company, January 2007.

ERM 2008. Groundwater Monitoring Well Installation & Monitoring Report Davis-Besse Nuclear Power Station, Environmental Resources Management (ERM) Reference 0065992.2, FirstEnergy Nuclear Operating Company, March 2008.

OEPA 2005. Sole Source Aquifers in Ohio, Ohio Environmental Protection Agency, Division of Drinking and Ground Waters, http://www.epa.state.oh.us/ddagw/pdu/swap_ssa.html, accessed March 3, 2009.

NEI 2007. Industry Ground Water Protection Initiative – Final Guidance Document, NEI-07-07 (Final), Nuclear Energy Institute, August 2007.

Figure 2.3-1: Groundwater Well Monitoring Locations



Source: **ERM 2007**