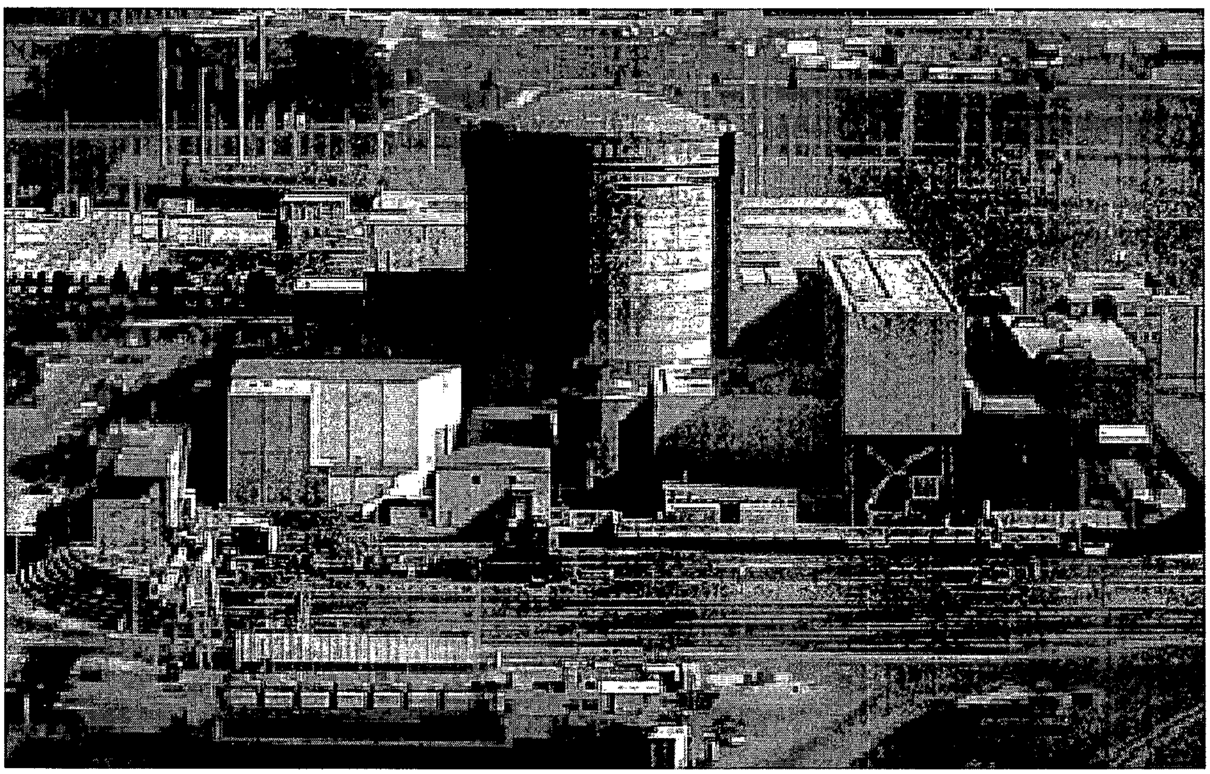
Progress Energy

Crystal River Unit 3



License Renewal Application

3.5 AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

3.5.1 INTRODUCTION

Section 3.5 provides the results of the aging management reviews (AMRs) for those components identified in Subsection 2.4, Scoping and Screening Results - Structures, subject to aging management review. The systems or portions of systems are described in the indicated subsections.

1. Reactor Building (Subsection 2.4.1)

2. Other Class I and In-Scope Structures:

   a. Auxiliary Building (Subsection 2.4.2.1)

   b. Wave Embankment Protection Structure (Subsection 2.4.2.2)

   c. Borated Water Storage Tank Foundation and Shield Wall (Subsection 2.4.2.3)

   d. Cable Bridge (Subsection 2.4.2.4)

   e. Control Complex (Subsection 2.4.2.5)

   f. Intake and Discharge Canals (Subsection 2.4.2.6)

   g. Circulating Water Discharge Structure (Subsection 2.4.2.7)

   h. Circulating Water Intake Structure (Subsection 2.4.2.8)

   i. Diesel Generator Building (Subsection 2.4.2.9)

   j. EFW Pump Building (Subsection 2.4.2.10)

   k. Dedicated EFW Tank Enclosure Building (Subsection 2.4.2.11)

   l. Fire Service Pumphouse (Subsection 2.4.2.12)

   m. Intermediate Building (Subsection 2.4.2.13)

   n. Machine Shop (Subsection 2.4.2.14)

   o. Miscellaneous Structures (Subsection 2.4.2.15)

p. Switchyard for Crystal River Site (Subsection 2.4.2.16)

q. Switchyard Relay Building (Subsection 2.4.2.17)

r. Turbine Building (Subsection 2.4.2.18)

Table 3.5.1, Summary of Aging Management Evaluations in Chapter II and III of NUREG-1801 for Containments, Structures, and Component Supports, provides the summary of the programs evaluated in NUREG-1801 that are applicable to component/ commodity groups in this Section. Table 3.5.1 uses the format of Table 1 described in Section 3.0 above.

3.5.1.1 Operating Experience

The AMR methodology applied at CR-3 included use of operating experience (OE) to confirm the set of aging effects that had been identified through material/environment evaluations. Plant-specific and industry OE was identified and reviewed in conjunction with the aging management review. The OE review consisted of the following:

Site: CR-3 site-specific OE has been captured by a review of the Action Tracking, Maintenance Rule, and OE databases and the results of inspections and assessments applicable to CR-3 structures. Relevant information provided by the Structural Systems Engineer was considered. The site-specific OE review identified stress corrosion cracking (SCC) of the Fuel Pool liner. The metallurgical investigation determined that the liner defects were due to SCC in the heat affected zone of the liner seam welds.

Industry: Industry OE has been captured in NUREG-1801, "Generic Aging Lessons Learned (GALL)," and is the primary method for verifying that a complete set of potential aging effects is identified. An evaluation of industry OE published since the effective date of NUREG-1801, Revision 1, was performed to identify any additional aging effects requiring management. This was performed using the Progress Energy internal OE review process which directs the review of OE and requires that it be screened and evaluated for site applicability. OE sources subject to review include INPO and WANO items, NRC documents (Information Notices, Generic Letters, Notices of Violation, and staff reports), 10 CFR 21 reports, and vendor bulletins, as well as corporate internal OE information from Progress Energy nuclear sites. The industry OE review identified no additional unpredicted or unique aging effects requiring management.

On-Going On-going review of plant-specific and industry operating experience is continuing to be performed in accordance with the Corrective Action Program and the Progress Energy OE review program.

3.5.2 RESULTS

The following tables summarize the results of the aging management review for Containments, Structures and Component Supports.

Table 3.5.2-1 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Reactor Building

Table 3.5.2-2 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Auxiliary Building

Table 3.5.2-3 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Wave Embankment Protection Structure

Table 3.5.2-4 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Borated Water Storage Tank Foundation and Shield Wall

Table 3.5.2-5 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Cable Bridge

Table 3.5.2-6 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Control Complex

Table 3.5.2-7 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Intake and Discharge Canals

Table 3.5.2-8 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Circulating Water Discharge Structure

Table 3.5.2-9 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Circulating Water Intake Structure

Table 3.5.2-10 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Diesel Generator Building

Table 3.5.2-11 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – EFW Pump Building

Table 3.5.2-12 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Dedicated EFW Tank Enclosure Building

Table 3.5.2-13 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Fire Service Pumphouse

Table 3.5.2-14 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Intermediate Building

Table 3.5.2-15 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Machine Shop

Table 3.5.2-16 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Miscellaneous Structures

Table 3.5.2-17 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Switchyard for Crystal River Site

Table 3.5.2-18 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Switchyard Relay Building

Table 3.5.2-19 Containments, Structures and Component Supports – Summary of Aging Management Evaluation – Turbine Building

These tables use the format of Table 2 described in Section 3.0 above.

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

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

3.5.2.1.1 Reactor Building

Materials

The materials of construction for the Reactor Building components are:

- Aluminum
- Carbon Steel
- Elastomers
- Fire Proofing Materials
- Fluorogold
- Galvanized Carbon Steel
- Insulation
- Reinforced Concrete
- Stainless Steel

Environment

The Reactor Building components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Borated Water Leakage
- Reinforced Concrete
- Soil
- Treated Water

Aging Effects Requiring Management

The following Reactor Building aging effects require management:

- Change in Material Properties
- Cracking
- Delamination
- Lock-Up
- Loss of Leak Tightness in Closed Condition
- Loss of Mechanical Function
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation
- Separation

Aging Management Programs

The following AMPs manage the aging effects for the Reactor Building components:

- 10 CFR Part 50, Appendix J Program
- ASME Section XI, Subsection IWE Program
- ASME Section XI, Subsection IWL Program
- ASME Section XI, Subsection IWF Program
- Bolting Integrity Program
- Boric Acid Corrosion Program
- Fire Protection Program
- Inspection of Overhead Heavy Load and Light Load Handling Systems Program
- Structures Monitoring Program
- Water Chemistry Program

3.5.2.1.2 Auxiliary Building

Materials

The materials of construction for the Auxiliary Building components are:

- Aluminum
- Boral
- Carbon Steel
- Carborundum (B_4C)
- Concrete Block
- Copper
- Elastomers
- Fire Proofing Materials
- Fluorogold
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Auxiliary Building components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Borated Water Leakage
- Raw Water - Seawater
- Reinforced Concrete
- Soil
- Treated Water

Aging Effects Requiring Management

The following Auxiliary Building aging effects require management:

- Change in Material Properties
- Cracking
- Delamination
- Loss of Material
- Loss of Mechanical Function
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation
- Separation

Aging Management Programs

The following AMPs manage the aging effects for the Auxiliary Building components:

- ASME Section XI, Subsection IWF Program
- Boric Acid Corrosion Program
- Carborundum (B_4C) Program
- Fire Protection Program
- Inspection of Overhead Heavy Load and Light Load Handling Systems Program
- Masonry Wall Program
- Structures Monitoring Program
- Water Chemistry Program

3.5.2.1.3 Wave Embankment Protection Structure

Materials

The materials of construction for the Wave Embankment Protection Structure components are:

- Earth
- Reinforced Concrete (includes Unreinforced Concrete and Fabriform)

Environment

The Wave Embankment Protection Structure components are exposed to the following:

- Air-Outdoor
- Soil

Aging Effects Requiring Management

The following Wave Embankment Protection Structure aging effects require management:

- Change in Material Properties
- Cracking
- Loss of Form
- Loss of Material

Aging Management Programs

The following AMP manages the aging effects for the Wave Embankment Protection Structure components:

- Structures Monitoring Program

3.5.2.1.4 Borated Water Storage Tank Foundation and Shield Wall

Materials

The materials of construction for the Borated Water Storage Tank Foundation and Shield Wall components are:

- Aluminum
- Carbon Steel
- Elastomers
- Galvanized Carbon Steel
- Reinforced Concrete

Environment

The Borated Water Storage Tank Foundation and Shield Wall components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Borated Water Leakage
- Reinforced Concrete

Aging Effects Requiring Management

The following Borated Water Storage Tank Foundation and Shield Wall aging effects require management:

- Change in Material Properties
- Cracking
- Loss of Material
- Loss of Mechanical Function
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMPs manage the aging effects for the Borated Water Storage Tank Foundation and Shield Wall components:

- Boric Acid Corrosion Program
- Structures Monitoring Program

3.5.2.1.5 Cable Bridge

Materials

The materials of construction for the Cable Bridge components are:
- Aluminum
- Carbon Steel
- Fluorogold
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Cable Bridge components are exposed to the following:
- Air-Indoor
- Air-Outdoor
- Raw Water - Seawater
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Cable Bridge aging effects require management:
- Change in Material Properties
- Cracking
- Lock-up
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMP manages the aging effects for the Cable Bridge components:
- Structures Monitoring Program

3.5.2.1.6 Control Complex

Materials

The materials of construction for the Control Complex components are:

- Aluminum
- Carbon Steel
- Concrete Block
- Copper
- Elastomers
- Fire Proofing Materials
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel
- Willtec Foam (Control Room Ceiling Panels)

Environment

The Control Complex components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Control Complex aging effects require management:

- Change in Material Properties
- Cracking
- Delamination
- Loss of Material
- Loss of Mechanical Function
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation
- Reduction or Loss of Isolation Function
- Separation

Aging Management Programs

The following AMPs manage the aging effects for the Control Complex components:

- ASME Section XI, Subsection IWF Program
- Fire Protection Program
- Masonry Wall Program
- Structures Monitoring Program

3.5.2.1.7 Intake and Discharge Canals

Materials

The material of construction for the Intake and Discharge Canals components is:

- Earth

Environment

The Intake and Discharge Canals components are exposed to the following:

- Air-Outdoor
- Raw Water - Seawater

Aging Effects Requiring Management

The following Intake and Discharge Canals aging effects require management:

- Loss of Form
- Loss of Material

Aging Management Programs

The following AMP manages the aging effects for the Intake and Discharge Canals components:

- Structures Monitoring Program

3.5.2.1.8 Circulating Water Discharge Structure

Materials

The material of construction for the Circulating Water Discharge Structure components is:

- Reinforced Concrete

Environment

The Circulating Water Discharge Structure components are exposed to the following:

- Air-Outdoor
- Raw Water - Seawater
- Soil

Aging Effects Requiring Management

The following Circulating Water Discharge Structure aging effects require management:

- Change in Material Properties
- Cracking
- Loss of Material

Aging Management Programs

The following AMP manages the aging effects for the Circulating Water Discharge Structure components:

- Structures Monitoring Program

3.5.2.1.9 Circulating Water Intake Structure

Materials

The materials of construction for the Circulating Water Intake Structure components are:

- Carbon Steel
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Circulating Water Intake Structure components are exposed to the following:

- Air-Outdoor
- Raw Water - Seawater
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Circulating Water Intake Structure aging effects require management:

- Change in Material Properties
- Cracking
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMPs manage the aging effects for the Circulating Water Intake Structure components:

- Inspection of Overhead Heavy Load and Light Load Handling Systems Program
- Structures Monitoring Program

3.5.2.1.10 Diesel Generator Building

Materials

The materials of construction for the Diesel Generator Building components are:

- Aluminum
- Carbon Steel
- Elastomers
- Fire Proofing Materials
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Diesel Generator Building components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Diesel Generator Building aging effects require management:

- Change in Material Properties
- Cracking
- Delamination
- Loss of Material
- Loss of Mechanical Function
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation
- Separation

Aging Management Programs

The following AMPs manage the aging effects for the Diesel Generator Building components:

- ASME Section XI, Subsection IWF Program
- Fire Protection Program
- Structures Monitoring Program

3.5.2.1.11 EFW Pump Building

Materials

The materials of construction for the EFW Pump Building components are:
- Aluminum
- Carbon Steel
- Elastomers
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The EFW Pump Building components are exposed to the following:
- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following EFW Pump Building aging effects require management:
- Change in Material Properties
- Cracking
- Loss of Material
- Loss of Mechanical Function
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMPs manage the aging effects for the EFW Pump Building components:
- ASME Section XI, Subsection IWF Program
- Fire Protection Program
- Inspection of Overhead Heavy Load and Light Load Handling Systems Program
- Structures Monitoring Program

3.5.2.1.12 Dedicated EFW Tank Enclosure Building

Materials

The materials of construction for the Dedicated EFW Tank Enclosure Building components are:

- Aluminum
- Carbon Steel
- Elastomers
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Dedicated EFW Tank Enclosure Building components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil
- Treated Water

Aging Effects Requiring Management

The following Dedicated EFW Tank Enclosure Building aging effects require management:

- Change in Material Properties
- Cracking
- Loss of Mechanical Function
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMPs manage the aging effects for the Dedicated EFW Tank Enclosure Building components:

- ASME Section XI, Subsection IWF Program
- Structures Monitoring Program

3.5.2.1.13 Fire Service Pumphouse

Materials

The materials of construction for the Fire Service Pumphouse components are:
- Aluminum
- Carbon Steel
- Concrete Block
- Elastomers
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Fire Service Pumphouse components are exposed to the following:
- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Fire Service Pumphouse aging effects require management:
- Change in Material Properties
- Cracking
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMPs manage the aging effects for the Fire Service Pumphouse components:
- Masonry Wall Program
- Structures Monitoring Program

3.5.2.1.14 Intermediate Building

Materials

The materials of construction for the Intermediate Building components are:

- Aluminum
- Carbon Steel
- Elastomers
- Fire Proofing Materials
- Fluorogold
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Intermediate Building components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Borated Water Leakage
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Intermediate Building aging effects require management:

- Change in Material Properties
- Cracking
- Delamination
- Loss of Material
- Loss of Mechanical Function
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation
- Reduction or Loss of Isolation Function
- Separation

Aging Management Programs

The following AMPs manage the aging effects for the Intermediate Building components:

- ASME Section XI, Subsection IWF Program
- Boric Acid Corrosion Program
- Fire Protection Program
- Structures Monitoring Program

3.5.2.1.15 Machine Shop

Materials

The materials of construction for the Machine Shop components are:
- Aluminum
- Carbon Steel
- Elastomers

Environment

The Machine Shop components are exposed to the following:
- Air-Indoor
- Air-Outdoor

Aging Effects Requiring Management

The following Machine Shop aging effects require management:
- Loss of Material
- Reduction or Loss of Isolation Function

Aging Management Programs

The following AMP manages the aging effects for the Machine Shop components:
- Structures Monitoring Program

3.5.2.1.16 Miscellaneous Structures

Materials

The materials of construction for the Miscellaneous Structures components are:
- Aluminum
- Carbon Steel
- Elastomers
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Miscellaneous Structures components are exposed to the following:
- Air-Indoor

- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Miscellaneous Structures aging effects require management:
- Change in Material Properties
- Cracking
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMPs manage the aging effects for the Miscellaneous Structures components:
- Fire Protection Program
- Structures Monitoring Program
- One-Time Inspection Program

3.5.2.1.17 Switchyard for Crystal River Site

Materials

The materials of construction for the Switchyard for Crystal River Site components are:
- Aluminum
- Carbon Steel
- Galvanized Carbon Steel
- Reinforced Concrete

Environment

The Switchyard for Crystal River Site components are exposed to the following:
- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Switchyard for Crystal River Site aging effects require management:
- Change in Material Properties
- Cracking

- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMP manages the aging effects for the Switchyard for Crystal River Site components:

- Structures Monitoring Program

3.5.2.1.18 <u>Switchyard Relay Building</u>

Materials

The materials of construction for the Switchyard Relay Building components are:

- Aluminum
- Carbon Steel
- Concrete Block
- Elastomers
- Galvanized Carbon Steel
- Reinforced Concrete

Environment

The Switchyard Relay Building components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Switchyard Relay Building aging effects require management:

- Change in Material Properties
- Cracking
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation

Aging Management Programs

The following AMPs manage the aging effects for the Switchyard Relay Building components:

- Masonry Wall Program
- Structures Monitoring Program

3.5.2.1.19 Turbine Building

Materials

The materials of construction for the Turbine Building components are:

- Aluminum
- Carbon Steel
- Concrete Block
- Elastomers
- Fire Proofing Materials
- Galvanized Carbon Steel
- Reinforced Concrete
- Stainless Steel

Environment

The Turbine Building components are exposed to the following:

- Air-Indoor
- Air-Outdoor
- Reinforced Concrete
- Soil

Aging Effects Requiring Management

The following Turbine Building aging effects require management:

- Change in Material Properties
- Cracking
- Delamination
- Loss of Material
- Reduction In Concrete Anchor Capacity due to Local Concrete Degradation
- Reduction or Loss of Isolation Function
- Separation

Aging Management Programs

The following AMPs manage the aging effects for the Turbine Building components:

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

3.5.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 identifies those aging management activities that warrant further evaluation. For the Containments, Structures, and Component Supports, these activities are addressed in the following subsections.

3.5.2.2.1 PWR and BWR Containments

3.5.2.2.1.1 Aging of Inaccessible Concrete Areas

For the Reactor Building (RB) structure, the ASME Section XI, Subsection IWL Program is used to manage aging of accessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel.

The CR-3 site groundwater is non-aggressive based on samples taken in February 2007 from two wells:

- Well CR3-1S - pH 7.19, chlorides 450 ppm, sulfates 140 ppm, phosphates < 0.5 ppm; and
- Well CR3-2 - pH 7.64, chlorides 37 ppm, sulfates 11 ppm, phosphates < 0.5 ppm.

For inaccessible areas of plants with non-aggressive groundwater/soil, the following is required: (1) Examination of the exposed portions of the below grade concrete, when excavated for any reason, and (2) Periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations.

With respect to monitoring inaccessible areas, the below grade portions of RB concrete are surrounded by other concrete structures. Below-grade RB concrete cannot be examined unless the concrete of surrounding structures is removed. However, examination of exposed representative portions of below grade concrete in the same groundwater environment for the surrounding structures is performed when uncovered during removal of backfill. This is considered equivalent to examining the RB concrete.

In addition, the Structures Monitoring Program is used to ensure that groundwater is monitored on a periodic basis including consideration of potential seasonal variations.

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

Aging effects caused by settlement are managed by the Structures Monitoring Program. A de-watering system is not relied upon for control of settlement.

The structures were founded on 1,500 psi fill concrete placed over competent existing limerock. For the RB, a settlement analysis determined the upper limit of total settlement to be on the order of 0.875 in., and that all but a very small fraction of settlement would occur during construction (Refer to FSAR Section 2.5.7.2.). No cracking due to settlement is expected or has been observed; however, the Structures Monitoring Program examines concrete for cracking and is credited for managing the aging effect of cracking.

The NUREG-1801 item regarding erosion of porous concrete subfoundations is not applicable. CR-3 does not have a porous concrete subfoundation.

3.5.2.2.1.3 Reduction of Strength and Modulus of Concrete Structures Due to Elevated Temperature

The NUREG-1801 item regarding concrete degradation from elevated temperatures is not applicable, because no RB pressure boundary concrete structural components exceed the specified temperature limits.

The RB Cooling System maintains the RB general area below an average temperature of 130°F. The local area concrete in the cylinder wall where hot pipes pass through is maintained at below 200°F either by insulation on the pipe or a combination of insulation and a Penetration Cooling System on several penetrations.

RB non-pressure boundary concrete is discussed in Subsection 3.5.2.2.2.3.

3.5.2.2.1.4 Loss of Material Due to General, Pitting, and Crevice Corrosion

The aging effect for the RB liner, liner anchors, and integral attachments is managed by the ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J Programs.

Loss of material due to corrosion is not significant for inaccessible areas, i.e., embedded containment steel liner, based on meeting the conditions specified as follows:

1. Concrete meeting ACI 318 was used in contact with the embedded steel liner. ACI 201.2R was not used as guidance for concrete mix proportions, but ACI 301-66 was used, and it provides similar guidance to produce a low permeability, dense, air entrained, low water-cement ratio concrete, properly placed and cured.

2. The RB Liner is monitored for corrosion or degraded protective coatings by the ASME Section XI, Subsection IWE Program.

3. The moisture barrier is monitored for aging effects by the ASME Section XI, Subsection IWE Program.

4. Borated water spills and water ponding on the RB floor are not common, and are cleaned up promptly when identified. The design of the RB floor provides for collection of water in a sump area that is maintained pumped-down.

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

The RB is a prestressed concrete containment. Loss of prestress forces due to relaxation, shrinkage, creep, and elevated temperature for the CR-3 RB is a TLAA as defined in 10 CFR 54.3. The evaluation of this TLAA is addressed separately in Section 4.5 of the LRA.

3.5.2.2.1.6 Cumulative Fatigue Damage

Fatigue is a TLAA for the expansion bellows associated with the Fuel Transfer Tube inside the RB and inside the Auxiliary Building (AB). The evaluation of this TLAA is provided in Section 4.6 of the LRA. A fatigue analysis does not exist in the current licensing basis (CLB) for the liner plate, penetration sleeves, and dissimilar metal welds between the fuel transfer tubes and the penetration sleeves.

Penetration bellows are installed outside the RB and are not part of the Containment pressure boundary; therefore, they are not in the scope of License Renewal.

Also, the NUREG-1801 BWR components, i.e. suppression pool shell and unbraced downcomers are not applicable to the CR-3 PWR containment.

3.5.2.2.1.7 Cracking Due to Stress Corrosion Cracking (SCC)

Cracking due to SCC is not an applicable effect for penetration sleeves and dissimilar metal welds; because, to be susceptible to SCC, stainless steel must be subjected to both high temperature (>140°F) and an aggressive chemical environment, unless there is plant specific operating experience showing SCC. The penetration sleeves and the dissimilar metal weld components are in the Air-Indoor environment and not subject to an aggressive chemical environment.

The exterior surface of the stainless steel fuel transfer tubes and associated components located in the RB Refuel Canal are included in this commodity group because the fuel transfer tubes are examined by the ASME Section XI, Subsection IWE and the 10 CFR Part 50, Appendix J Programs. During refueling activities, the exterior surface of the stainless steel fuel transfer tubes and associated components are exposed to a treated water environment. The aging effect of cracking due to SCC and use of the Water Chemistry Program is addressed in Table 3.3.1, Item 3.3.1-90, for the stainless steel fuel transfer tubes and associated components.

Penetration bellows are installed outside the RB and are not part of the Containment pressure boundary; therefore, they are not in the scope of License Renewal.

3.5.2.2.1.8 Cracking Due to Cyclic Loading

Cracking due to cyclic loading is managed by the ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J Programs for the penetration sleeves, dissimilar metal welds, and fuel transfer tubes and cover plates in the RB.

No operating experience has been found for aging effect of fine cracking of these components and CR-3 does not expect fine cracking of the penetration sleeves, dissimilar metal welds, fuel transfer tubes, and cover plates to occur. The supplemental aging effect of fine cracking is a result of cyclic loading or fatigue. Use of the ASME Section XI, Subsection IWE Program together with the 10 CFR Part 50, Appendix J Program is adequate for monitoring the aging effects for penetrations sleeves, dissimilar metal welds, and fuel transfer tubes due to cyclic loading.

Penetration bellows are installed outside the RB and are not part of the Containment pressure boundary; therefore, they are not in the scope of License Renewal.

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

Loss of material due to freeze-thaw is not an applicable effect, because CR-3 is located in a negligible weathering region per ASTM C33 and is not subject to freeze-thaw. Examinations of the accessible concrete performed by the ASME Section XI, Subsection IWL Program have not identified any aging effects due to freeze-thaw.

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 is not an applicable aging effect. Fine and coarse aggregates were tested with each brand of cement for possible alkali reaction in accordance with ASTM C227; and aggregates did not react within the reinforced concrete. In addition, concrete was constructed to ACI 301-66, which provides guidance similar to ACI 201.2R for producing high density, low permeability concrete.

For increase in porosity and permeability due to leaching of calcium hydroxide, concrete was constructed to ACI 301-66, which provides guidance similar to ACI 201.2R for producing high density, low permeability concrete. However, an increase in porosity and permeability due to leaching of calcium hydroxide is conservatively considered to be an aging effect requiring management, because minor indications of leaching in below grade concrete exists in the RB tendon access gallery. The aging effect of change in material properties has been assigned, as equivalent to an increase in porosity and permeability, and is managed by the ASME Section XI, Subsection IWL Program.

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

Further evaluation is provided because the following structure/aging effect combinations are not covered by the Structures Monitoring Program:

1. Loss of material (i.e., spalling and scaling) and cracking due to freeze-thaw for NUREG-1801 Group 1-3, 5, and 7-9 structures is not applicable; because CR-3 is located in a negligible weathering region per ASTM C33 and is not subject to freeze-thaw.

2. Cracking due to expansion and reaction with aggregates for Group 1-5 and 7-9 structures is not applicable; because CR-3 fine and coarse aggregates were tested with each brand of cement for possible alkali reaction in accordance with ASTM C227, and aggregates did not react within the reinforced concrete. In addition, concrete was constructed to ACI 301-66, which provides guidance similar to ACI 201.2R for producing high density, low permeability concrete mix designs.

3. Cracking and distortion due to increased stress levels from settlement for Group 1-3 and 5-9 structures is applicable with the exception of the Group 8 Borated Water Storage Tank Foundation and Shield Wall, which is supported on the AB and is not in a soil environment.

4. Reduction in foundation strength, cracking, differential settlement due to erosion of porous concrete subfoundation for Groups 1-3 and 5-9 structures is not applicable because CR-3 does not have a porous concrete subfoundation; and a dewatering system is not relied upon to control erosion of cement from porous concrete.

Further evaluation is provided because the following structure/aging effect combination is not covered by the Structures Monitoring or ASME Section XI, Subsection IWF Program:

Lock up due to wear for Lubrite® radial beam seats in BWR drywell, RPV support shoes for PWR with nozzle supports, steam generator supports, and other sliding support bearings and sliding support surfaces is not applicable because CR-3 does not utilize Lubrite in these applications.

The Structures Monitoring Program is utilized to manage the aging effects due to corrosion of embedded steel, aggressive chemical attack, settlement for concrete, and loss of material for steel elements.

3.5.2.2.2.2 Aging Management of Inaccessible Areas

1. Freeze-Thaw

Loss of material and cracking due to freeze-thaw is not an applicable effect because CR-3 is located in a negligible weathering region per ASTM C33 which is not subject to freeze-thaw. Examinations of the accessible concrete performed by the Structures Monitoring Program have not identified any aging effects due to freeze-thaw.

2. Reaction with Aggregates

Cracking due to expansion and reaction with aggregate is not an applicable effect; because CR-3 fine and coarse aggregates were tested with each brand of cement for possible alkali reaction in accordance with ASTM C227, and aggregates did not react within the reinforced concrete. In addition, concrete was constructed to ACI 301-66, which provides guidance similar to ACI 201.2R for producing high density, low permeability concrete mix designs.

3. Increased Stress Levels from Settlement and Erosion of Porous Concrete

The aging effect of cracking and distortion due to increased stress levels due to settlement is managed by the Structures Monitoring Program with the exception of the Group 3 Borated Water Storage Tank Foundation and Shield Wall which is supported on the AB roof slab and is not in a soil environment. A de-watering system is not relied upon for control of settlement.

The structures were founded on 1,500 psi fill concrete placed over competent existing limerock (over crushed grouted limerock for the RB), cement-grouted limerock (Control Complex, Intermediate Building), or compacted backfill. A settlement analysis for the RB determined the upper limit of total settlement was found to be on the order of 0.875 in., but that all but a very small fraction of settlement would occur during construction. No cracking due to settlement is expected or has been observed; however, the Structures Monitoring Program examines concrete for cracking due to settlement and is credited for managing the aging effect of cracking.

The NUREG-1801 item regarding erosion of porous concrete subfoundations is not applicable. CR-3 does not have a porous concrete subfoundation.

4. Aggressive Chemical Attack and Corrosion of Embedded Steel

Groundwater chemistry is non-aggressive at CR-3 based on groundwater samples taken from two plant wells in 2007 as follows:

- Well CR3-1S – pH 7.19, chlorides 450 ppm, sulfates 140 ppm, phosphates < 0.5 ppm; and
- Well CR3-2- pH 7.64, chlorides 37 ppm, sulfates 11 ppm, phosphates < 0.5 ppm.

However, concrete cracking, loss of material, and change in material properties are conservatively assumed to be applicable to CR-3 in the soil environment.

The Structures Monitoring Program will continue to monitor groundwater on a periodic basis including consideration of potential seasonal variations. The Structures Monitoring Program will also continue to examine the exposed portions of the below-grade concrete when excavated for any reason.

5. Leaching of Calcium Hydroxide

Concrete was constructed to ACI 301-66, which provides guidance similar to ACI 201.2R for producing high density, low permeability concrete mix designs. However, an increase in porosity and permeability due to leaching of calcium hydroxide is conservatively considered to be an aging effect requiring aging management because of the existence of minor indications of leaching in below-grade concrete in the RB tendon access gallery. Therefore, any below grade concrete in the scope of License Renewal will be examined whenever excavated for any reason in accordance with the Structures Monitoring Program.

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

The NUREG-1801 item regarding concrete degradation from elevated temperatures is not applicable, because neither the RB non-pressure boundary concrete nor the concrete structures outside the RB exceed the specified temperature limits.

The RB is maintained below an average ambient temperature of 130°F with the RB Cooling System; and the area between the primary shield wall and the reactor vessel is maintained at a temperature below 200°F. The local area inside "D"-Ring above 119 ft. elevation near the top of the pressurizer is subject to a temperature of 164.3°F but the area is open to the RB general area environment and is not enclosed.

The normal temperature for structures outside the RB vary from 140°F in the Intermediate Building to 85°F in the Control Complex with no general areas temperatures exceeding 150°F or local area temperatures exceeding 200°F.

RB pressure boundary concrete is discussed in Subsection 3.5.2.2.1.3.

3.5.2.2.2.4 Aging Management of Inaccessible Areas for Group 6 Structures

1. Aggressive Chemical Attack and Corrosion of Embedded Steel

Groundwater chemistry is non-aggressive at CR-3 based on groundwater samples taken from two plant wells in 2007 as follows:

- Well CR3-1S – pH 7.19, chlorides 450 ppm, sulfates 140 ppm, phosphates < 0.5 ppm; and
- Well CR3-2- pH 7.64, chlorides 37 ppm, sulfates 11 ppm, phosphates < 0.5 ppm.

However, concrete cracking, loss of material, and change in material properties are conservatively assumed to be applicable to CR-3 in the soil environment.

The Structures Monitoring Program will continue to monitor groundwater on a periodic basis including consideration of potential seasonal variations during the period of extended operation. The Structures Monitoring Program will also continue to examine the exposed portions of the below-grade concrete when excavated for any reason.

2. Freeze-Thaw

Loss of material due to freeze-thaw is not an applicable effect because CR-3 is located in a negligible weathering region per ASTM C33 which is not subject to freeze-thaw. Examinations of the accessible concrete performed by the Structures Monitoring Program have not identified any aging effects due to freeze-thaw.

3. Reaction with Aggregates and Leaching of Calcium Hydroxide

Cracking due to expansion and reaction with aggregate is not an applicable aging effect. Fine and coarse aggregates were tested with each brand of cement for possible alkali reaction in accordance with ASTM C227 and aggregates did not react within the reinforced concrete. In addition, concrete was constructed to ACI 301-66, which provides guidance similar to ACI 201.2R for producing high density, low permeability concrete mix designs.

For increase in porosity and permeability due to leaching of calcium hydroxide, concrete was constructed to ACI 301-66, which provides similar guidance to produce high density, low permeability concrete as ACI 201.2R. However, an increase in porosity and permeability due to leaching of calcium hydroxide is conservatively considered to be an aging effect requiring aging management because of the existence of minor indications of leaching in below-grade concrete in the RB tendon access gallery. The aging effect of change in material properties has been assigned as equivalent to an increase in porosity and permeability and is managed by the Structures Monitoring Program..

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 of stainless steel tank liners is not applicable to CR-3. CR-3 does not have tanks with stainless steel liners. Aging management of tanks is addressed with the mechanical system in which the tanks are located.

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 including (1) loss of material due to general and pitting corrosion for Groups B2-B5 supports; (2) reduction in concrete anchor capacity due to degradation of the surrounding concrete for Groups B1-B5 supports; and (3) reduction/loss of isolation function due to degradation of vibration isolation elements for Group B4 supports. The following apply to CR-3 supports:

1. The Structures Monitoring Program is used to manage loss of material due to general and pitting corrosion for Groups B2-B5 supports for CR-3 structures within the scope of License Renewal.

2. The Structures Monitoring Program is used to manage reduction in concrete anchor capacity due to degradation of the surrounding concrete, for Groups B1-B5 supports for CR-3 structures within the scope of License Renewal.

3. Reduction/loss of isolation function due to degradation of vibration isolation elements for Group B4 supports (i.e., NUREG-1801, Volume 2, related item T-31) is applicable only in the Control Complex, Intermediate Building, Machine Shop, and Turbine Building for ventilation equipment.

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 CLB fatigue analysis exists.

There are no fatigue analyses in the CLB applicable to component supports; therefore, cumulative fatigue damage of component supports is not a TLAA as defined in 10 CFR 54.3.

3.5.2.2.3 <u>Quality Assurance for Aging Management of Non-Safety Related Components</u>

QA provisions applicable to License Renewal are discussed in Section B.1.3.

3.5.2.3 Time-Limited Aging Analysis

The Time-Limited Aging Analyses (TLAA) identified below are associated with the Containments, Structures, and Component Support components. The section of the application that contains the TLAA review results is indicated in parenthesis.

1. Tendon Stress Relaxation (Section 4.5)
2. Expansion Bellows Cyclic Fatigue (Section 4.6)
3. Bedrock Dissolution from Groundwater (Section 4.7)

3.5.3 CONCLUSIONS

The Containments, Structures, and Component Support components/commodities having aging effects requiring management have been evaluated, and aging management programs have been selected to manage the aging effects. A description of the aging management programs is provided in Appendix B, along with a demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstration provided in Appendix B, the effects of aging will be adequately managed so that there is reasonable assurance that the intended functions of Containments, Structures, and Component Support components/ commodities will be maintained consistent with the CLB during the period of extended operation.

TABLE 3.5.1 SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
PWR Concrete (Reinforced and Prestressed) and Steel Containment BWR Concrete (Mark II and III) and Steel (Mark I, II, and III) Containment					
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 environment is non-aggressive. A plant specific program is to be evaluated if environment is aggressive.	Yes, plant-specific, if the environment is aggressive	Consistent with NUREG-1801. The CR-3 groundwater is non-aggressive and no further evaluation is required. Refer to Subsection 3.5.2.2.1.1 for additional information regarding groundwater parameters and examination of representative samples of below grade concrete for the RB.
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	Consistent with NUREG-1801. Aging effects due to settlement are managed by the Structures Monitoring Program and a de-watering system is not relied upon for control of settlement. Therefore, further evaluation is not required. Additional information regarding settlement is provided in Subsection 3.5.2.2.1.2.
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 to control erosion of cement from porous concrete subfoundations, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Yes, if not within the scope of the applicant's structures monitoring program or a de-watering system is relied upon	The aging mechanism erosion of porous concrete subfoundation is not applicable to the RB. Refer to evaluation in Subsection 3.5.2.2.1.2.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-04	Concrete elements: dome, wall, basemat, ring girder, buttresses, containment, concrete fill-in annulus (as applicable)	Reduction of strength and modulus due to elevated temperature	Plant-specific	Yes, plant-specific if temperature limits are exceeded	Reduction of strength and modulus due to elevated temperature is not applicable to the RB. Refer to the evaluation in Subsection 3.5.2.2.1.3.
3.5.1-05	BWR Only				
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 due to corrosion is not significant for inaccessible areas (embedded containment steel liner). Refer to the evaluation in Subsection 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	Consistent with NUREG-1801. A TLAA is evaluated in accordance with 10 CFR 54.21(c) for applicable components. Refer to the evaluation in Subsection 3.5.2.2.1.5.
3.5.1-08	BWR Only				
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. A TLAA is evaluated in accordance with 10 CFR 54.21(c) for applicable components. Refer to the evaluation in Subsection 3.5.2.2.1.6.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	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	Cracking due to SCC is not an applicable effect for the penetration sleeves and dissimilar metal welds in an Air-Indoor environment. Refer to the evaluation in Subsection 3.5.2.2.1.7 for a discussion of these components and the external surface of the fuel transfer tubes and associated components in a treated water environment.
3.5.1-11	BWR Only				
3.5.1-12	Steel, stainless steel elements, dissimilar metal welds: penetration sleeves, penetration bellows; suppression pool shell, unbraced downcomers	Cracking due to cyclic loading	ISI (IWE) and 10 CFR Part 50, Appendix J supplemented to detect fine cracks	Yes, detection of aging effects is to be evaluated	The aging effect of cracking for the penetration sleeves, dissimilar metal welds, and fuel transfer tubes and cover plates in the RB are managed by the ASME Section XI, Subsection IWE and 10 CFR Part 50 Appendix J Programs. However, CR-3 does not supplement the programs to detect fine cracks. Refer to evaluation in Subsection 3.5.2.2.1.8.
3.5.1-13	BWR Only				
3.5.1-14	Concrete elements: dome, wall, basemat ring girder, buttresses, containment (as applicable)	Loss of material (Scaling, cracking, and spalling) due to freeze-thaw	ISI (IWL). Evaluation is needed for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557).	Yes, for inaccessible areas of plants located in moderate to severe weathering conditions	Freeze-thaw is not applicable to CR-3. Refer to evaluation in Subsection 3.5.2.2.1.9.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-15	Concrete elements: walls, dome, basemat, ring girder, buttresses, containment, concrete fill-in annulus (as applicable).	Increase in porosity, permeability due to leaching of calcium hydroxide; cracking due to expansion and reaction with aggregate	ISI (IWL) for accessible areas. None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R.	Yes, if concrete was not constructed as stated for inaccessible areas	This NUREG-1801 item is not applicable with respect to cracking due to expansion and reaction with aggregate. This item is consistent with NUREG-1801 with respect to increase in porosity and permeability due to leaching of calcium hydroxide. Refer to the evaluation in Subsection 3.5.2.2.1.10.
3.5.1-16	Seals, gaskets, and moisture barriers	Loss of sealing and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers (caulking, flashing, and other sealants)	ISI (IWE) and 10 CFR Part 50, Appendix J	No	Consistent with NUREG-1801. The ASME Section XI, Subsection IWE and 10 CFR Part 50 Appendix J Programs are used to manage the aging effects of cracking and change in material properties which result in loss of sealing and leakage through the RB due to deterioration of joint seals, gaskets; and moisture barrier at the RB liner to concrete slab interface. The ISI (IWE) Program is applicable to the Moisture Barrier at the RB liner-to-concrete floor slab interface. The 10 CFR Part 50, Appendix J program is applicable to the penetration Seals and Gaskets.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-17	Personnel airlock, equipment hatch and CRD hatch locks, hinges, and closure mechanisms	Loss of leak tightness in closed position due to mechanical wear of locks, hinges and closure mechanisms	10 CFR Part 50, Appendix J and Plant Technical Specifications	No	Consistent with NUREG-1801. The 10 CFR Part 50 Appendix J Program is used to confirm loss of leak tightness of Personnel Airlocks and the Equipment Hatch in closed position in accordance with the CR-3 Technical Specifications
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. The ASME Section XI, Subsection IWE and 10 CFR Part 50 Appendix J Programs are used to manage loss of material due to corrosion.
3.5.1-19	BWR Only				
3.5.1-20	BWR Only				
3.5.1-21	BWR Only				
3.5.1-22	Prestressed containment: tendons and anchorage components	Loss of material due to corrosion	ISI (IWL)	No	Consistent with NUREG-1801. The ASME Section XI, Subsection IWE Program is used to manage loss of material due to corrosion.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
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. The Structures Monitoring Program is used to manage accessible concrete of the non-pressure boundary RB concrete (including the Equipment Access Structure and Tendon Gallery) and the structures outside the RB. See Section 3.5.2.2.2.1 for further discussion.
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. The Structures Monitoring Program is used to manage accessible concrete of the non-pressure boundary RB concrete (including the Equipment Access Structure and the Tendon Gallery) and the structures outside the RB. See Section 3.5.2.2.2.1 for further discussion.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-25	All Groups except Group 6: steel components: all structural steel	Loss of material due to corrosion	Structures Monitoring Program. If protective coatings are relied upon to manage the effects of aging, the structures monitoring program is to include provisions to address protective coating monitoring and maintenance.	Yes, if not within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. The Structures Monitoring Program is used to manage loss of material due to corrosion for the group Steel Components: All structural steel which includes the steel inside the RB, the Equipment Access Structure, and the structures outside the RB. Protective coatings are not relied upon to manage the effects of aging. See Section 3.5.2.2.2.1 for further 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	This NUREG-1801 item is not applicable. See Subsections 3.5.2.2.2.1 and 3.5.2.2.2.2.1 for further discussion.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-27	All Groups except Group 6: accessible and inaccessible interior/exterior concrete	Cracking due to expansion due to reaction with aggregates	Structures Monitoring Program None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R-77.	Yes, if not within the scope of the applicant's structures monitoring program or concrete was not constructed as stated for inaccessible areas	This NUREG-1801 item is not applicable. See Subsections 3.5.2.2.2.1 and 3.5.2.2.2.2.2 for further discussion.
3.5.1-28	Groups 1-3, 5-9: All	Cracks and distortion due to increased stress levels from settlement	Structures Monitoring Program. If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Yes, if not within the scope of the applicant's structures monitoring program or a de-watering system is relied upon	Consistent with NUREG-1801, with the exception of the Borated Water Storage Tank Foundation and Shield Wall. Aging effects due to settlement are managed by the Structures Monitoring Program except for the Borated Water Storage Tank Foundation and Shield Wall which is supported on the Auxiliary Building roof slab, not located in a soil environment, and not subject to settlement. A de-watering system is not relied upon for control of settlement. See Subsections 3.5.2.2.2.1 and 3.5.2.2.2.2.3 for further discussion.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	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	This NUREG-1801 item is not applicable. CR-3 does not have a porous concrete subfoundation, and a dewatering system is not relied upon to control erosion of cement from porous concrete. See Subsections 3.5.2.2.2.1 and 3.5.2.2.2.2.3 for further discussion.
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	This NUREG-1801 item is not applicable. Lubrite plates are not utilized in these applications. See Subsections 3.5.2.2.2.1 for further discussion.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-31	Groups 1-3, 5, 7-9: below-grade concrete components, such as exterior walls below grade and foundation	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack; Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Structures monitoring Program; Examination of representative samples of below-grade concrete, and periodic monitoring of groundwater, if the environment is non-aggressive. A plant specific program is to be evaluated if environment is aggressive.	Yes, plant-specific, if environment is aggressive	Consistent with NUREG-1801. The Structures Monitoring Program is used to manage accessible concrete of the non-pressure boundary concrete of the RB (i.e., the Equipment Access Structure and the Tendon Gallery), and the structures outside the RB. This item is not applicable to the BWST because the structure is located on the Auxiliary Building concrete roof slab and all the concrete structure is above grade. Further evaluation and a plant specific program are required if the environment is aggressive. The environment is non-aggressive as documented in Subsection 3.5.2.2.2.2.4.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-32	Groups 1-3, 5, 7-9: exterior above and below grade reinforced concrete foundations	Increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide.	Structures Monitoring Program for accessible areas. None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R-77.	Yes, if concrete was not constructed as stated for inaccessible areas	Consistent with NUREG-1801. The Structures Monitoring Program is used to manage accessible concrete of the non-pressure boundary concrete of the RB (Equipment Access Structure and the Tendon Gallery), and the structures outside the RB. See Subsections 3.5.2.2.2.2.5 for further discussion of inaccessible concrete and ACI Codes used for concrete mix designs.
3.5.1-33	Groups 1-5: concrete	Reduction of strength and modulus due to elevated temperature	Plant-specific	Yes, plant-specific if temperature limits are exceeded	This NUREG-1801 item is not applicable. The concrete for the non-pressure boundary of the RB (including the Equipment Access Structure and the Tendon Gallery), and the structures outside the RB do not exceed the specified temperature limits specified for general area or local area concrete. Additional information regarding reduction of strength and modulus due to elevated temperature is provided in Subsection 3.5.2.2.2.3.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-34	Group 6: Concrete; all	Cracking, loss of bond, loss of material due to corrosion of embedded steel; increase in porosity and permeability, cracking, loss of material due to aggressive chemical attack	Inspection of Water-Control Structures Assoc with Nuclear Power Plants and for inaccessible concrete, exam of rep. samples of below-grade concrete, and periodic monitoring of groundwater, if environment is non-aggressive. Plant specific if environment is aggressive.	Yes, plant-specific if environment is aggressive	Consistent with NUREG-1801 for material, environment, and aging effect, but the Structures Monitoring Program is credited as the applicable aging management program as allowed by NUREG 1801, Section XI.S7. The Structures Monitoring Program meets the requirements of RG 1.127, Inspection of Water-Control Structures. Exposed portions of below-grade concrete will be examined under the provisions of the Structures Monitoring Program when excavated for any reason. Refer to the additional information in Subsection 3.5.2.2.2.4.1.
3.5.1-35	Group 6: exterior above and below grade concrete foundation	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Inspection of Water-Control Structures Associated with Nuclear Power Plants. 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	This NUREG-1801 item is not applicable. Additional information is provided in Subsection 3.5.2.2.2.4.2.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	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 Associated with Nuclear Power Plants. 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	This NUREG-1801 item is not applicable. See Section 3.5.2.2.2.4.3 for further discussion.
3.5.1-37	Group 6: exterior above and below grade reinforced concrete foundation interior slab	Increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide	For accessible areas, Inspection of Water-Control Structures Associated with Nuclear Power Plants. None for inaccessible areas if concrete was constructed in accordance with the recommendations in ACI 201.2R-77.	Yes, if concrete was not constructed as stated for inaccessible areas	Consistent with NUREG-1801 for material, environment, and aging effect, but the Structures Monitoring Program is credited as the applicable aging management program as allowed by NUREG 1801, Section XI.S7. The Structures Monitoring Program meets the requirements of RG 1.127, Inspection of Water-Control Structures. The Structures Monitoring Program is used to manage accessible concrete. See Subsections 3.5.2.2.2.4.3 for further discussion of inaccessible concrete and ACI Codes used for concrete mix designs.
3.5.1-38	Groups 7, 8: Tank liners	Cracking due to stress corrosion cracking; loss of material due to pitting and crevice corrosion	Plant-specific	Yes, plant specific	This NUREG-1801 item is not applicable. See Section 3.5.2.2.2.5 for further discussion.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-39	Support members; welds; bolted connections; support anchorage to building structure	Loss of material due to general and pitting corrosion	Structures Monitoring Program	Yes, if not within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. The Structures Monitoring Program is used to manage loss of material due to general and pitting corrosion of Group B2-B5 supports for CR-3 structures with Refer to the information in Subsection 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	Consistent with NUREG-1801. The Structures Monitoring Program is used to manage reduction in concrete anchor capacity due to local concrete degradation / service-induced cracking or other concrete aging for CR-3 structures within the scope of License Renewal. Refer to the information in Subsection 3.5.2.2.2.6. Note that Group B1.3 is applicable to a BWR and is not applicable to CR-3.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-41	Vibration isolation elements	Reduction or loss of isolation function/ radiation hardening, temperature, humidity, sustained vibratory loading	Structures Monitoring Program	Yes, if not within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. The Structures Monitoring Program is used to manage the reduction or loss of isolation function for non-metallic vibration isolation elements within the scope of License Renewal in the Control Complex, Intermediate Building, Machine Shop, and the Turbine Building for ventilation equipment. Refer to the information in Subsection 3.5.2.2.2.6.
3.5.1-42	Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	This NUREG-1801 item is not applicable. A fatigue analysis does not exist in the CLB for the supports for Groups B1.1 and B1.2 at CR-3. Therefore, no TLAA evaluation is necessary as specified in NUREG-1801. Group B1.3 is applicable to a BWR and not applicable for a PWR. Refer to the information in Subsection 3.5.2.2.2.7.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
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. The Masonry Wall Program is used to manage cracking due to restraint shrinkage, creep, and aggressive environment for the structures with masonry block walls (i.e., the Auxiliary Building, Control Complex, Fire Service Pump House, CR-3 Switchyard Relay Building, and Turbine Building).
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. The Structures Monitoring Program is used to manage aging effects of cracking and change in material properties for applicable structures which result in loss of sealing. These structures are not in Group 6, but the alignment is based on the same material, environment, aging effect and aging management program. This is applicable to the Auxiliary Building, Borated Water Storage Tank Foundation and Shield Wall, Control Complex, Diesel Generator Building, EFW Pump Building, Dedicated EFW Tank Enclosure Building, Fire Service Pump House, Intermediate Building, (continued)

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-44 (contin-ued)	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	Miscellaneous Structures, Switchyard Relay Building, Reactor Building, and Turbine Building. Cracking and change in material properties for elastomers results in loss of sealing and is considered an equivalent aging effect.
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 Associated with Nuclear Power Plants	No	Consistent with NUREG-1801 for material, environment, and aging effect, but the Structures Monitoring Program is credited as the applicable aging management program as allowed by NUREG 1801, Section XI.S7. The Structures Monitoring Program meets the requirements of RG 1.127, Inspection of Water-Control Structures. The Structures Monitoring Program is used to manage concrete for loss of material due to concrete abrasion and cavitation for the Cable Bridge Structure, Circulating Water Intake Structure, Circulating Water Discharge Structure, and the Nuclear Service Seawater Pump Sump of the Auxiliary Building.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	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 and level of fluid in the leak chase channel.	No	Consistent with NUREG-1801. Cracking due to SCC and loss of material due to pitting and crevice corrosion are managed by the Water Chemistry Program, and by monitoring the water level of the Spent Fuel Pool (SFP) and the fluid level in the leak chase test hopper. Plant Technical Specifications require monitoring of the spent fuel pool water level, and the leak chase test hoppers are monitored daily. The expansion bellows in the CR-3 SFP is also included based on the same material, environment, aging effect and aging management program. The SFP environmental conditions do not normally exceed the 140°F pool temperature threshold that can result in cracking due to the SCC for stainless steel components. However, CR-3 site-specific operating experience has determined that there is a potential for SCC of the stainless steel SFP Liner Plate. Susceptibility to SCC is generally limited to the stainless steels that have relatively high carbon content. Specifically, the SFP manufactured from (continued)

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-46 (contin-ued)	Group 5: Fuel pool liners	Cracking due to stress corrosion cracking; loss of material due to pitting and crevice corrosion	Water Chemistry and Monitoring of spent fuel pool water level and level of fluid in the leak chase channel.	No	Types 302 and 304 stainless steels can have carbon content that would increase susceptibility even for pool temperatures below the 140°F threshold. Therefore, the SFP Liner Plate will be considered to be susceptible to SCC. This NUREG-1801 table line item is specific to the SFP where spent fuel is stored. However, inside the RB, cracking due to SCC and loss of material due to corrosion for the Reactor Cavity Liner/Refueling Canal Liner/ Expansion Bellows is managed by the Water Chemistry Program during refueling outages when the Reactor Cavity/Refueling Canal is flooded. These commodities are included based on the same material, environment, aging effect and aging management program. Monitoring of spent fuel pool water level and level of fluid in the leak chase channel is not applicable inside the RB. Normally the Reactor Cavity Liner/Refueling Canal Liner/ Expansion Bellows are exposed to an Air-Indoor environment. No aging effects are (continued)

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-46 (contin-ued)	Group 5: Fuel pool liners	Cracking due to stress corrosion cracking; loss of material due to pitting and crevice corrosion	Water Chemistry and Monitoring of spent fuel pool water level and level of fluid in the leak chase channel.	No	anticipated for these stainless steel components in an air environment as indicated on Table 3.5.2-1 for the component/commodity Steel Components: Fuel Pool Liner. Refer to the discussion of SCC for the SFP Racks in Table line item 3.3.1-90.
3.5.1-47	Group 6: all metal structural members	Loss of material due to general (steel only), pitting and crevice corrosion	Inspection of Water-Control Structures Associated with Nuclear Power Plants. If protective coatings are relied upon to manage aging, protective coating monitoring and maintenance provisions should be included.	No	Consistent with NUREG-1801 for material, environment, and aging effect, but the Structures Monitoring Program is credited as the applicable aging management program as allowed by NUREG 1801, Section XI.S7. The Structures Monitoring Program meets the requirements of RG 1.127, Inspection of Water-Control Structures. The Structures Monitoring Program is used to manage the metal structural members (including the trash racks) at the Circulating Water Intake Structure for loss of material due to general, pitting and crevice corrosion.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-48	Group 6: earthen water control structures - dams, embankments, reservoirs, channels, canals, and ponds	Loss of material, loss of form due to erosion, settlement, sedimentation, frost action, waves, currents, surface runoff, seepage	Inspection of Water-Control Structures Associated with Nuclear Power Plants	No	Consistent with NUREG-1801 for material, environment, and aging effect, but the Structures Monitoring Program is credited as the applicable aging management program as allowed by NUREG 1801, Section XI.S7. The Structures Monitoring Program meets the requirements of RG 1.127, Inspection of Water-Control Structures. The Structures Monitoring Program is used to manage aging effects of loss of material and loss of form at the Intake Canal and the Wave Embankment Protection Structure.
3.5.1-49	BWR Only				
3.5.1-50	Groups B2, and B4: galvanized steel, aluminum, stainless steel support members; welds; bolted connections; support anchorage to building structure	Loss of material due to pitting and crevice corrosion	Structures Monitoring Program	No	Consistent with NUREG-1801. The Structures Monitoring Program is used to manage loss of material for galvanized steel, aluminum, and stainless steel due to corrosion for Groups B2 and B4 components. Other components are aligned with this group such as "Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and other (continued)

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-50 (continued)	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	Miscellaneous Structures," "Siding," "Steel Components: All Structural Steel," "Door (Non-fire)," or "Racks, Panels, Cabinets, and Enclosure for Electrical Equipment and Instrumentation," because they have the same material, environment, aging effect and aging management 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. The Bolting Integrity Program is used to manage cracking due to stress corrosion cracking and loss of material due to general corrosion for high strength (i.e., yield strength > 150 KSI) Reactor Coolant System anchor bolts in the RB. These high strength structural bolts are located in areas in the RB that are subject to wetting from RCS leakage or spray. Operating experience has not identified SCC or loss of material for these high strength bolts.
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	This NUREG-1801 item is not applicable. There are no NUREG-1801 Group III.B2 or III.B4 Lubrite or graphitic tool steel components used inside or outside of the RB at CR-3.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1-53	Groups B1.1, B1.2, and B1.3: support members: welds; bolted connections; support anchorage to building structure	Loss of material due to general and pitting corrosion	ISI (IWF)	No	Consistent with NUREG-1801. The ASME Section XI, Subsection IWF Program is used to manage loss of material for steel components due to corrosion for Groups B1.1 and B1.2 components. Group B1.3 for BWR Containment Supports is not applicable.
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	Consistent with NUREG-1801. The ASME Section XI, Subsection IWF Program is used to manage Loss of mechanical function due to corrosion, distortion, dirt, overload, and fatigue due to vibratory and cyclic thermal loads for Groups B1.1 and B1.2 components. Group B1.3 for BWR Containment Supports is not applicable.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	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. The Boric Acid Corrosion Program is used to manage loss of material for carbon steel, galvanized carbon steel, and aluminum components due to contact with boric acid. Group B1.3 for BWR Containment Supports is not applicable, and Groups B1.1-8, B1.2-6, B1.3-6 and B4-6 are not applicable because they are not used at CR-3.
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	This NUREG-1801 item is not applicable. Lubrite plates are not utilized in these applications for Groups B1.1 and B1.2. Group B1.3 for BWR Containment Supports is not applicable.
3.5.1-57	Groups B1.1, B1.2, and B1.3: Vibration isolation elements	Reduction or loss of isolation function/ radiation hardening, temperature, humidity, sustained vibratory loading	ISI (IWF)	No	This NUREG-1801 item is not applicable. Vibration isolation elements (non metallic) are not utilized in these applications for Groups B1.1 and B1.2. Also, Group B1.3 for BWR Containment Supports is not applicable.

TABLE 3.5.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTERS II AND III OF NUREG-1801 FOR CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Item Number	Component/ Commodity	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
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. Although they are not support members, some additional components (i.e., "Door (Non-Fire)," "Floor Drains," "Fire Hose Stations," "Steel Components: All structural steel," "Steel Components: Fuel Pool Liner," "Raised Floor," "Siding," and "Draft Stops") have been included in this item; based on use of the same material, environment, aging effect and aging management program.
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. Although they are not support members, some additional components (i.e., "Door (Non-Fire)," "Floor Drains," "Fire Hose Stations," "Steel Components: All structural steel," "Steel Components: Fuel Pool Liner," "Raised Floor," "Siding," and "Draft Stops") have been included in this item based on use of the same material, environment, aging effect and aging management program. Group B1.3 for BWR Containment Supports is not applicable.

TABLE 3.5.2-1 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Concrete	None	None			J, 501
		Stainless Steel	Concrete	None	None			J, 501
		Galvanized Steel	Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C- 2 C- 7	Aluminum	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A
			Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
		Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A
			Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A
			Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
		Stainless Steel	Borated Water Leakage	None	None	III.B2-9 (TP-4)	3.5.1-59	A
			Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Dome; Wall; Basemat; Ring Girder; Buttresses	C-1 C-2 C-3 C-4 C-6 C-7 C-8 C-12 C-14	Reinforced Concrete	Air - Outdoor	Change in Material Properties Cracking Loss of Material	ASME Section XI, Subsection IWL	II.A1-7 (C-05)	3.5.1-01	A
				Change in Material Properties Cracking Loss of Material	ASME Section XI, Subsection IWL	II.A1-4 (C-03)	3.5.1-01	A
				Cracking Loss of Material	ASME Section XI, Subsection IWL	VII.G-30 (A-92)	3.3.1-66	E
				Loss of Material	ASME Section XI, Subsection IWL	VII.G-31 (A-93)	3.3.1-67	E
			Air - Indoor	Change in Material Properties Cracking Loss of Material	ASME Section XI, Subsection IWL	II.A1-4 (C-03)	3.5.1-01	A
				Change in Material Properties Cracking Loss of Material	ASME Section XI, Subsection IWL	II.A1-7 (C-05)	3.5.1-01	A
				Cracking Loss of Material	ASME Section XI, Subsection IWL	VII.G-28 (A-90)	3.3.1-65	E
				Loss of Material	ASME Section XI, Subsection IWL	VII.G-29 (A-91)	3.3.1-67	E

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Basemat	C-1 C-2 C-3 C-7 C-12	Reinforced Concrete	Soil	Change in Material Properties Cracking Loss of Material	ASME Section XI, Subsection IWL	II.A1-4 (C-03)	3.5.1-01	A
				Change in Material Properties Cracking Loss of Material	ASME Section XI, Subsection IWL	II.A1-7 (C-05)	3.5.1-01	A, 514
				Cracking	Structures Monitoring	II.A1-5 (C-37)	3.5.1-02	A
				Change in Material Properties	ASME Section XI, Subsection IWL	II.A1-6 (C-02)	3.5.1-15	A
Concrete: Above Grade	C-2 C-3 C-6 C-7 C-12 C-13 C-14	Reinforced Concrete	Air-Indoor	Change in Material Properties Cracking Loss of Material	Structures Monitoring	III.A3-9 III.A4-3 III.A5-9 (T-04)	3.5.1-23	A
				Change in Material Properties Cracking Loss of Material	Structures Monitoring	III.A3-10 III.A4-4 III.A5-10 (T-06)	3.5.1-24	A

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-3 C-6 C-7 C-12 C-13 C-14	Reinforced Concrete	Air-Indoor	Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.1-1 III.B1.2-1 III.B2-1 III.B3-1 III.B4-1 III.B5-1 (T-29)	3.5.1-40	A
			Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
Concrete: Below Grade	C-2 C-3	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-2 C-3	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Cranes	C-7	Carbon Steel	Air - Indoor	Loss of Material	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-3 (A-07)	3.3.1-73	A, 515
				Loss of Material / Wear (of Rail)	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-1 (A-05)	3.3.1-74	A, 515
				None	None	VII.B-2 (A-06)	3.3.1-01	I
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 504

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Expansion Bellows	C-15	Stainless Steel	Air - Indoor	Fatigue damage	TLAA	II.A3-4 (C-13)	3.5.1-09	C, 503
				None	None	III.B5-5 (TP-5)	3.5.1-59	C, 504
			Treated Water	Loss of Material Cracking (SCC)	Water Chemistry	III.A5-13 (T-14)	3.5.1-46	C, 505
Fire Barrier Assemblies	C-4	Fire Proofing Materials (Thermo-Lag)	Air - Indoor	Loss of Material Cracking / Delamination Separation	Fire Protection			J, 502
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 504
Floor Drains	C-8	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	C, 508
			Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 516
Insulation	C-3	Unibestos	Air - Indoor	None	None			J, 506

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Penetration Sleeves	C-1 C-2 C-7	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (T-25)	3.5.1-55	C, 508
			Air - Indoor	Loss of Material	ASME Section XI Subsection IWE and 10 CFR Part 50, Appendix J	II.A3-1 (C-12)	3.5.1-18	A
				Cracking (Cyclic Loading)	ASME Section XI Subsection IWE and 10 CFR Part 50, Appendix J	II.A3-3 (C-14)	3.5.1-12	B, 513
		Stainless Steel	Borated Water Leakage	None	None	III.B5-6 (TP-4)	3.5.1-59	C, 504
			Air - Indoor	None	None	II.A3-2 (C-15)	3.5.1-10	I, 513
				Cracking (Cyclic Loading)	ASME Section XI Subsection IWE and 10 CFR Part 50, Appendix J	II.A3-3 (C-14)	3.5.1-12	B, 513

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Personnel Airlock; Equipment Hatch	C-1 C-3 C-8	Carbon Steel	Air - Indoor	Loss of Leak Tightness in Closed Condition	10 CFR Part 50, Appendix J and Plant Technical Specifications	II.A3-5 (C-17)	3.5.1-17	A
				Loss of Material	ASME Section XI Subsection IWE and 10 CFR Part 50, Appendix J	II.A3-6 (C-16)	3.5.1-18	A
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-2 C-7 C-11 C-12	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (T-25)	3.5.1-55	A
			Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (TP-30)	3.5.1-39	A
		Stainless Steel	Borated Water Leakage	None	None	III.B5-6 (TP-4)	3.5.1-59	A, 504
			Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	A, 504
	C-2 C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (TP-30)	3.5.1-39	A
		Galvanized Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 517

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-8 (T-25)	3.5.1-55	A
			Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Galvanized Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-4 (TP-3)	3.5.1-55	A
			Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A
		Stainless Steel	Borated Water Leakage	None	None	III.B3-6 (TP-4)	3.5.1-59	A
			Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
Seals and Gaskets	C-3 C-7	Elastomer	Air - Indoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	A, 509
			Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	A, 509
Seals, Gaskets, and Moisture Barriers	C-1 C-3	Elastomer	Air - Indoor	Cracking Change in Material Properties	ASME Section XI Subsection IWE and 10 CFR Part 50, Appendix J	II.A3-7 (C-18)	3.5.1-16	A, 507

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Steel Components: All Structural Steel	C-2 C-7 C-12	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (T-25)	3.5.1-55	C, 508
			Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 III.A4-5 (T-11)	3.5.1-25	A
		Fluorogold	Air - Indoor	Lock-up	Structures Monitoring			J, 510
				Change in Material Properties	Structures Monitoring			J, 510
Steel Components: Fuel Pool Liner	C-2 C-7 C-12	Stainless Steel	Borated Water Leakage	None	None	III.B5-6 (TP-4)	3.5.1-59	C, 504
			Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 504
			Treated Water	Loss of Material	Water Chemistry	III.A5-13 (T-14)	3.5.1-46	C, 512
				Cracking (SCC)	Water Chemistry	III.A5-13 (T-14)	3.5.1-46	A
Steel Elements: Liner; Liner Anchors; Integral Attachments	C-1 C-2 C-7 C-12	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (T-25)	3.5.1-55	C, 508
			Air - Indoor	Loss of Material	ASME Section XI Subsection IWE and 10 CFR Part 50, Appendix J	II.A1-11 (C-09)	3.5.1-06	A

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for ASME Class 1, 2, 3 Piping & Components	C-2 C-12	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B1.1-14 III.B1.2-11 (T-25)	3.5.1-55	A
			Air - Indoor	Loss of Material	ASME Section XI, Subsection IWF	III.B1.1-13 III.B1.2-10 (T-24)	3.5.1-53	A
				Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.1-2 III.B1.2-2 (T-28)	3.5.1-54	A
		Stainless Steel	Borated Water Leakage	None	None	III.B1.1-10 III.B1.2-8 (TP-4)	3.5.1-59	A
			Air - Indoor	None	None	III.B1.1-9 III.B1.2-7 (TP-5)	3.5.1-59	A
		Fluorogold	Air - Indoor	Loss of Mechanical Function	ASME Section XI, Subsection IWF			J, 511
				Change in Material Properties	ASME Section XI, Subsection IWF			J, 511
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-2 C-7 C-12	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A, 518

TABLE 3.5.2-1 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – REACTOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for Non-ASME Piping & Components	C-7 C-12	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A
			Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Stainless Steel	Borated Water Leakage	None	None	III.B2-9 (TP-4)	3.5.1-59	A
			Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A
Supports for Reactor Coolant System Primary Equipment	C-2 C-12	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B1.1-14 (T-25)	3.5.1-55	A
			Air - Indoor	Loss of Material	ASME Section XI, Subsection IWF	III.B1.1-13 (T-24)	3.5.1-53	A
				Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.1-2 (T-28)	3.5.1-54	A
				Cracking (SCC)	Bolting Integrity	III.B1.1-3 (T-27)	3.5.1-51	A, 548
				Loss of Material	Bolting Integrity	III.B1.1-4 (TP-9)	3.5.1-51	A, 548
Tendons	C-2	Carbon Steel	Air -Indoor	Loss of Material	ASME Section XI, Subsection IWL	II.A1-10 (C-10)	3.5.1-22	A
				Loss of Prestress	TLAA	II.A1-9 (C-11)	3.5.1-07	A
			Air - Outdoor	Loss of Material	ASME Section XI, Subsection IWL	II.A1-10 (C-10)	3.5.1-22	A
				Loss of Prestress	TLAA	II.A1-9 (C-11)	3.5.1-07	A

TABLE 3.5.2-2 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Battery Racks	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A
		Stainless Steel	Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A
			Borated Water Leakage	None	None	III.B2-9 (TP-4)	3.5.1-59	A
		Copper	Air - Indoor	None	None			J, 525
			Borated Water Leakage	None	None			J, 525

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade	C-2 C-3 C-4 C-6 C-7 C-8 C-14	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-30 (A-92)	3.3.1-66	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-31 (A-93)	3.3.1-67	A
			Air-Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 III.A5-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 III.A5-10 (T-06)	3.5.1-24	A

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-3 C-4 C-6 C-7 C-8 C-14	Reinforced Concrete	Air-Indoor	Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.1-1 III.B1.2-1 III.B2-1 III.B3-1 III.B4-1 III.B5-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-28 (A-90)	3.3.1-65	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-29 (A-91)	3.3.1-67	A
Concrete: Below Grade	C-2 C-3 C-7 C-8	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 III.A5-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 III.A5-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 III.A5-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 III.A5-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-2 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 III.A5-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 III.A5-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 III.A5-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 III.A5-7 (T-02)	3.5.1-32	A
Concrete: Submerged	C-2 C-7	Reinforced Concrete	Raw Water - Seawater	Loss of Material Cracking Change in Material Properties	Structures Monitoring			G, 543
				Change in Material Properties	Structures Monitoring	III.A3-7 III.A5-7 (T-02)	3.5.1-32	A
				Loss of material	Structures Monitoring	III.A6-7 (T-20)	3.5.1-45	A, 520

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Cranes	C-7	Carbon Steel	Air - Indoor	Loss of Material	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-3 (A-07)	3.3.1-73	A, 535
				Loss of Material / Wear (of Rail)	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-1 (A-05)	3.3.1-74	A, 535
				None	None	VII.B-2 (A-06)	3.3.1-01	I
Damper Mountings	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B4-5 (TP-11)	3.5.1-58	A
Door (Non-fire)	C-3 C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Door	C-3 C-4	Carbon Steel	Air - Indoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21)	3.3.1-63	E
			Air - Outdoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-4 (A-22)	3.3.1-63	E
Draft Stop	C-7	Galvanized Carbon Steel	Air - Indoor	None	None	III.B5-3 (TP-11)	3.5.1-58	C, 519, 522

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Expansion Bellows	C-15	Stainless Steel	Treated Water	Loss of Material Cracking	Water Chemistry	III.A5-13 (T-14)	3.5.1-46	C, 505
				Fatigue Damage	TLAA	II.A3-4 (C-13)	3.5.1-09	C, 542
Fire Barrier Assemblies	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation	Fire Protection			J, 526
Fire Barrier Penetration Seals	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-1 (A-19)	3.3.1-61	A
			Air - Outdoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-2 (A-19)	3.3.1-61	A
Fire Hose Stations	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	C, 522

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Floor Drains	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	C, 522
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 522
			Borated Water Leakage	None	None	III.B5-6 (TP-4)	3.5.1-59	C, 522
Masonry Walls	C-8	Concrete Block	Air - Indoor	Cracking	Masonry Wall	III.A3-11 III.A5-11 (T-12)	3.5.1-43	A
New Fuel Storage Racks	C-2	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	VII.A1-1 (A-94)	3.3.1-86	A
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-2 C-7 C-8 C-11	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B5-3 (TP-11)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-4 (A-94)	3.5.1-55	A

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures (continued)	C-2 C-7 C-8 C-11	Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	A
			Borated Water Leakage	None	None	III.B5-6 (TP-4)	3.5.1-59	A
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-8 (T-25)	3.5.1-55	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-4 (TP-3)	3.5.1-55	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-4 (TP-3)	3.5.1-55	A
		Stainless Steel	Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
			Borated Water Leakage	None	None	III.B3-6 (TP-4)	3.5.1-59	A

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Roof-Membrane / Built-up	C-3	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 529
Seals and Gaskets	C-2 C-3 C-7 C-8	Elastomer	Air - Indoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
			Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509, 536
Siding	C-3 C-7	Aluminum	Air - Indoor	None	None	III.B5-2 (TP-8)	3.5.1-58	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Spent Fuel Storage Racks	C-2 C-10	Stainless Steel	Treated Water	Cracking	Water Chemistry	VII.A2-7 (A-97)	3.3.1-90	A
		Boral	Treated Water	None	None	VII.A2-3 (A-89) VII.A2-5 (A-88)	3.3.1-13	I, 528
		Carborundum (B_4C)	Treated Water	Loss of Material	Carborundum (B_4C)			F, 540
Steel Components: All Structural Steel	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	C, 522

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Steel Components: Fuel Pool Liner	C-2 C-7	Stainless Steel	Treated Water	Loss of Material Cracking	Water Chemistry , Monitoring of spent fuel pool level, and Monitoring Leakage from the Leak Chase Channels	III.A5-13 (T-14)	3.5.1-46	A
			Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 522
			Borated Water Leakage	None	None	III.B5-6 (TP-4)	3.5.1-59	C, 522
Supports for ASME Class 1, 2, 3 Piping & Components	C-2	Carbon Steel	Air - Indoor	Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.1-2 III.B1.2-2 (T-28)	3.5.1-54	A
				Loss of Material	ASME Section XI, Subsection IWF	III.B1.1-13 III.B1.2-10 (T-24)	3.5.1-53	A
			Air - Outdoor	Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.1-2 III.B1.2-2 (T-28)	3.5.1-54	A
				Loss of Material	ASME Section XI, Subsection IWF	III.B1.1-13 III.B1.2-10 (T-24)	3.5.1-53	A

TABLE 3.5.2-2 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – AUXILIARY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for ASME Class 1, 2, 3 Piping & Components (continued)	C-2	Carbon Steel	Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B1.1-14 III.B1.2-11 (T-25)	3.5.1-55	A
		Fluorogold	Air - Indoor	Loss of Mechanical Function	ASME Section XI, Subsection IWF			J, 549
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A, 518
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A

TABLE 3.5.2-3 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – WAVE EMBANKMENT PROTECTION STRUCTURE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade	C-7	Reinforced Concrete (Includes Unreinforced Concrete, and Fabriform)	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A, 524, 550
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A, 550
Concrete: Below Grade	C-7	Reinforced Concrete (Includes Unreinforced Concrete and Fabriform)	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A, 550
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A, 524, 550
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A, 550
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A, 550

TABLE 3.5.2-3 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – WAVE EMBANKMENT PROTECTION STRUCTURE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-7	Reinforced Concrete (Includes Unreinforced Concrete)	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A, 550
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A, 524, 550
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A, 550
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A, 550
Earthen Berm	C-7 C-8	Earth	Air - Outdoor	Loss of Material	Structures Monitoring	III.A6-9	3.5.1-48	E, 531
				Loss of Form	Structures Monitoring	III.A6-9	3.5.1-48	E, 531

TABLE 3.5.2-4 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – BORATED WATER STORAGE TANK FOUNDATION AND SHIELD WALL

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A
		Galvanized Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A
Concrete: Above Grade	C-2 C-3 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A, 544
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A, 544

TABLE 3.5.2-4 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – BORATED WATER STORAGE TANK FOUNDATION AND SHIELD WALL

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-3 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 III.B5-1 (T-29)	3.5.1-40	A
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 III.B5-1 (T-29)	3.5.1-40	A
Door (Non-fire)	C-3 C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522

TABLE 3.5.2-4 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – BORATED WATER STORAGE TANK FOUNDATION AND SHIELD WALL

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	A
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-8 (T-25)	3.5.1-55	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-4 (TP-3)	3.5.1-55	A
Seals and Gaskets	C-3 C-7 C-8	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 521

TABLE 3.5.2-4 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – BORATED WATER STORAGE TANK FOUNDATION AND SHIELD WALL

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A

TABLE 3.5.2-5 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CABLE BRIDGE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Stainless Steel	Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
Concrete: Above Grade	C-7	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A

TABLE 3.5.2-5 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CABLE BRIDGE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade	C-7	Reinforced Concrete	Air - Outdoor	Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 (T-29)	3.5.1-40	A
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B5-1 (T-29)	3.5.1-40	A
Concrete: Below Grade	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-5 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CABLE BRIDGE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Concrete: Submerged	C-7	Reinforced Concrete	Raw Water - Seawater	Loss of Material Cracking Change in Material Properties	Structures Monitoring			G, 543
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
				Loss of Material	Structures Monitoring	III.A6-7 (T-20)	3.5.1-45	A, 520
Door (Non-fire)	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522

TABLE 3.5.2-5 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CABLE BRIDGE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-7	Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A
Steel Components: All Structural Steel	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
		Fluorogold	Air - Outdoor	Lock-up	Structures Monitoring			J, 551

TABLE 3.5.2-6 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CONTROL COMPLEX

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Battery Racks	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A
		Copper	Air - Indoor	None	None			J, 525
Concrete: Above Grade	C-1 C-2 C-3 C-4 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-10 (T-06)	3.5.1-24	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-30 (A-92)	3.3.1-66	A, 539

TABLE 3.5.2-6 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CONTROL COMPLEX

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-1 C-2 C-3 C-4 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-31 (A-93)	3.3.1-67	A, 539
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.2-1 III.B2-1 III.B3-1 III.B4-1 III.B5-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-28 (A-90)	3.3.1-65	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-29 (A-91)	3.3.1-67	A

TABLE 3.5.2-6 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CONTROL COMPLEX

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Below Grade	C-1 C-2 C-3 C-7 C-8	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A1-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A1-7 (T-02)	3.5.1-32	A
Concrete: Foundation	C-2 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A1-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A1-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A1-7 (T-02)	3.5.1-32	A
Control Room Ceiling	C-7	Willtec Foam	Air - Indoor	None	None			J, 530

TABLE 3.5.2-6 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CONTROL COMPLEX

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Damper Mountings	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B4-5 (TP-11)	3.5.1-58	A
Door	C-1 C-3 C-4	Carbon Steel	Air - Indoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21) VII.G-4 (A-22)	3.3.1-63	E
Fire Barrier Assemblies	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation	Fire Protection			J, 526
Fire Barrier Penetration Seals	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-1 (A-19)	3.3.1-61	A
Fire Hose Stations	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Floor Drains	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 522
Masonry Walls	C-7	Concrete Block	Air - Indoor	Cracking	Masonry Wall	III.A3-11 (T-12)	3.5.1-43	A

TABLE 3.5.2-6 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CONTROL COMPLEX

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Phase Bus Duct Enclosure Assemblies	C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	VI.A-13 (LP-06)	3.6.1-09	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
Raised Floor	C-2 C-4	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Roof-Membrane / Built-up	C-3	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 529
Steel Components: All Structural Steel	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A1-12 (T-11)	3.5.1-25	A

TABLE 3.5.2-6 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CONTROL COMPLEX

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for ASME Class 1, 2, 3 Piping & Components	C-2	Carbon Steel	Air - Indoor	Loss of Material	ASME Section XI, Subsection IWF	III.B1.2-10 (T-24)	3.5.1-53	A
				Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.2-2 (T-28)	3.5.1-54	A
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A, 518
		Elastomer	Air - Indoor	Reduction or Loss of Isolation Function	Structures Monitoring	III.B4-12 (T-31)	3.5.1-41	A, 538
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A

TABLE 3.5.2-7 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTAKE AND DISCHARGE CANALS

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Earthen Water-Control Structures: Dams, embankments, reservoirs, channels, canals and ponds	C-5 C-7	Earth	Air-Outdoor	Loss of Material Loss of Form	Structures Monitoring	III.A6-9 (T-22)	3.5.1-48	E, 531
			Raw Water - Seawater	Loss of Material Loss of Form	Structures Monitoring	III.A6-9 (T-22)	3.5.1-48	E, 531

TABLE 3.5.2-8 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CIRCULATING WATER DISCHARGE STRUCTURE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade	C-7	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-1 (T-18)	3.5.1-34	A, 531
Concrete: Below Grade	C-7	Reinforced Concrete	Soil	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-1 (T-18)	3.5.1-34	A, 531
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-3 (T-19)	3.5.1-34	A, 531
				Cracking	Structures Monitoring	III.A6-4 (T-08)	3.5.1-28	A
				Change in Material Properties	Structures Monitoring	III.A6-6 (T-16)	3.5.1-37	A, 531
Concrete: Foundation	C-7	Reinforced Concrete	Soil	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-1 (T-18)	3.5.1-34	A, 531
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-3 (T-19)	3.5.1-34	A, 531

TABLE 3.5.2-8 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CIRCULATING WATER DISCHARGE STRUCTURE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation (continued)	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A6-4 (T-08)	3.5.1-28	A
				Change in Material Properties	Structures Monitoring	III.A6-6 (T-16)	3.5.1-37	A, 531
Concrete: Submerged	C-7	Reinforced Concrete	Raw Water - Seawater	Loss of Material Cracking Change in Material Properties	Structures Monitoring			G, 543
				Change in Material Properties	Structures Monitoring	III.A6-6 (T-16)	3.5.1-37	A, 531
				Loss of Material	Structures Monitoring	III.A6-7 (T-20)	3.5.1-45	A, 531

TABLE 3.5.2-9 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CIRCULATING WATER INTAKE STRUCTURE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
Concrete: Above Grade	C-2 C-5 C-7	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-1 (T-18)	3.5.1-34	A, 531
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 III.B5-1 (T-29)	3.5.1-40	A
Concrete: Below Grade	C-2 C-5 C-7	Reinforced Concrete	Soil	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-3 (T-19)	3.5.1-34	A, 531
				Cracking	Structures Monitoring	III.A6-4 (T-08)	3.5.1-28	A
				Change in Material Properties	Structures Monitoring	III.A6-6 (T-16)	3.5.1-37	A, 531

TABLE 3.5.2-9 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CIRCULATING WATER INTAKE STRUCTURE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-2 C-5 C-7	Reinforced Concrete	Soil	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A6-3 (T-19)	3.5.1-34	A, 531
				Cracking	Structures Monitoring	III.A6-4 (T-08)	3.5.1-28	A
				Change in Material Properties	Structures Monitoring	III.A6-6 (T-16)	3.5.1-37	A, 531
Concrete: Submerged	C-2 C-5 C-7	Reinforced Concrete	Raw Water - Seawater	Loss of Material Cracking Change in Material Properties	Structures Monitoring			G, 543
				Change in Material Properties	Structures Monitoring	III.A6-6 (T-16)	3.5.1-37	A, 531
				Loss of Material	Structures Monitoring	III.A6-7 (T-20)	3.5.1-45	A, 531
Cranes	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-3 (A-07)	3.3.1-73	A, 535
				Loss of Material / Wear (of Rail)	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-1 (A-05)	3.3.1-74	A, 535
				None	None	VII.B-2 (A-06)	3.3.1-01	I

TABLE 3.5.2-9 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – CIRCULATING WATER INTAKE STRUCTURE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Raw Water - Seawater	Loss of Material	Structures Monitoring	III.A6-11 (T-21)	3.5.1-47	A, 531
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
			Raw Water - Seawater	Loss of Material	Structures Monitoring			J, 537
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A

TABLE 3.5.2-10 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DIESEL GENERATOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Battery Racks	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Stainless Steel	Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A

TABLE 3.5.2-10 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DIESEL GENERATOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade	C-2 C-3 C-4 C-6 C-7 C-8 C-13	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.2-1 III.B2-1 III.B5-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-30 (A-92)	3.3.1-66	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-31 (A-93)	3.3.1-67	A
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A

TABLE 3.5.2-10 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DIESEL GENERATOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-3 C-4 C-6 C-7 C-8 C-13	Reinforced Concrete	Air - Indoor	Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.2-1 III.B2-1 III.B3-1 III.B4-1 III.B5-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-28 (A-90)	3.3.1-65	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-29 (A-91)	3.3.1-67	A
Concrete: Below Grade	C-2 C-3 C-7 C-8	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-10 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DIESEL GENERATOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-2 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Damper Mountings	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B4-5 (TP-11)	3.5.1-58	A
Door	C-3 C-4	Carbon Steel	Air - Indoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21) VII.G-4 (A-22)	3.3.1-63	E

TABLE 3.5.2-10 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DIESEL GENERATOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Fire Barrier Penetration Seals	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-1 (A-19)	3.3.1-61	A
			Air - Outdoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-2 (A-20)	3.3.1-61	A
Floor Drains	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 522
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-2 C-7 C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532, 541

TABLE 3.5.2-10 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DIESEL GENERATOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Roof-Membrane / Built-up	C-3	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 529
Seals and Gaskets	C-8	Elastomer	Air - Indoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
			Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
Steel Components: All Structural Steel	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
Supports for ASME Class 1, 2, 3 Piping & Components	C-2	Carbon Steel	Air - Indoor	Loss of Material	ASME Section XI, Subsection IWF	III.B1.2-10 (T-24)	3.5.1-53	A
				Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.2-2 (T-28)	3.5.1-54	A
			Air - Outdoor	Loss of Material	ASME Section XI, Subsection IWF	III.B1.2-10 (T-24)	3.5.1-53	A
				Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.2-2 (T-28)	3.5.1-54	A

TABLE 3.5.2-10 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DIESEL GENERATOR BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A, 518
Supports for Non-ASME Piping & Components	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A

TABLE 3.5.2-11 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – EFW PUMP BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
		Stainless Steel	Reinforced Concrete	None	None			J, 501
Battery Racks	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
Concrete: Above Grade	C-2 C-3 C-4 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-30 (A-92)	3.3.1-66	A

TABLE 3.5.2-11 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – EFW PUMP BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-3 C-4 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-31 (A-93)	3.3.1-67	A
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.2-1 III.B2-1 III.B3-1 III.B5-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-28 (A-90)	3.3.1-65	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-29 (A-91)	3.3.1-67	A

TABLE 3.5.2-11 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – EFW PUMP BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Below Grade	C-2 C-3 C-7 C-8	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Concrete: Foundation	C-2 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-11 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – EFW PUMP BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Cranes	C-7	Carbon Steel	Air - Indoor	Loss of Material	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-3 (A-07)	3.3.1-73	A, 535
				Loss of Material / Wear (of Rail)	Inspection of Overhead Heavy Load and Light Load Handling Systems	VII.B-1 (A-05)	3.3.1-74	A, 535
				None	None	VII.B-2 (A-06)	3.3.1-01	I
Damper Mountings	C-2	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B4-5 (TP-11)	3.5.1-58	A
Door (Non-fire)	C-3 C-8	Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 522
Door	C-4	Carbon Steel	Air - Indoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21) VII.G-4 (A-22)	3.3.1-63	E
Fire Barrier Assemblies	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation	Fire Protection			J, 526
Floor Drains	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522

TABLE 3.5.2-11 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – EFW PUMP BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B5-3 (TP-11)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	A
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
Seals and Gaskets	C-3 C-8	Elastomer	Air - Indoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
Steel Components: All Structural Steel	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
Supports for ASME Class 1, 2, 3 Piping & Components	C-2	Carbon Steel	Air - Indoor	Loss of Material	ASME Section XI, Subsection IWF	III.B1.2-10 (T-24)	3.5.1-53	A
				Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.2-2 (T-28)	3.5.1-54	A
		Stainless Steel	Air - Indoor	None	None	III.B1.2-7 (TP-5)	3.5.1-59	A

TABLE 3.5.2-11 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – EFW PUMP BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A

TABLE 3.5.2-12 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DEDICATED EFW TANK ENCLOSURE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
		Stainless Steel	Reinforced Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A
			Treated Water	Loss of Material	Structures Monitoring			G, 527
Concrete: Above Grade	C-2 C-3 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A, 544
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A, 544
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.2-1 III.B2-1 III.B3-1 (T-29)	3.5.1-40	A

TABLE 3.5.2-12 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DEDICATED EFW TANK ENCLOSURE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-3 C-6 C-7 C-8	Reinforced Concrete	Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A, 544
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A, 544
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.2-1 III.B2-1 III.B3-1 (T-29)	3.5.1-40	A
			Treated Water	Loss of Material Cracking Change in Material Properties	Structures Monitoring			G, 545
Concrete: Below Grade	C-2 C-3 C-7 C-8	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A8-2 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A8-3 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A8-4 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A8-6 (T-02)	3.5.1-32	A

TABLE 3.5.2-12 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DEDICATED EFW TANK ENCLOSURE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-2 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A8-2 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A8-3 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A8-4 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A8-6 (T-02)	3.5.1-32	A
Damper Mountings	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B4-5 (TP-11)	3.5.1-58	A
Door (Non-fire)	C-3 C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Floor Drains	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A

TABLE 3.5.2-12 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – DEDICATED EFW TANK ENCLOSURE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Seals and Gaskets	C-3 C-8	Elastomer	Air - Indoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
Steel Components: All Structural Steel	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
Supports for ASME Class 1, 2, 3 Piping & Components	C-2	Carbon Steel	Air - Indoor	Loss of Material	ASME Section XI, Subsection IWF	III.B1.2-10 (T-24)	3.5.1-53	A
				Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.2-2 (T-28)	3.5.1-54	A
			Treated Water	Loss of Material	ASME Section XI, Subsection IWF			G, 546
				Loss of Mechanical Function	ASME Section XI, Subsection IWF			G, 547
		Stainless Steel	Air - Indoor	None	None	III.B1.2-7 (T-59)	3.5.1-59	A
			Treated Water	Loss of Material	ASME Section XI, Subsection IWF			G, 527
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Treated Water	Loss of Material	Structures Monitoring			G, 546

TABLE 3.5.2-13 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – FIRE SERVICE PUMPHOUSE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
Concrete: Above Grade	C-7	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 (T-29)	3.5.1-40	A

TABLE 3.5.2-13 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – FIRE SERVICE PUMPHOUSE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-7	Reinforced Concrete	Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 (T-29)	3.5.1-40	A
Concrete: Foundation	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Damper Mountings	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A

TABLE 3.5.2-13 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – FIRE SERVICE PUMPHOUSE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Door (Non-fire)	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Masonry Walls	C-7	Concrete Block	Air - Indoor	Cracking	Masonry Wall	III.A3-11 (T-12)	3.5.1-43	A
			Air - Outdoor	Cracking	Masonry Wall	III.A3-11 (T-12)	3.5.1-43	A
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A, 541
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532

TABLE 3.5.2-13 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – FIRE SERVICE PUMPHOUSE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Roof-Membrane / Built-Up	C-7	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 529
Steel Components: All Structural Steel	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A, 518
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A

TABLE 3.5.2-14 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
	C-7	PVC	Reinforced Concrete	None	None			J, 501
Battery Racks	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A, 533
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A, 533
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-6 (TP-3)	3.5.1-55	A, 533
		Stainless Steel	Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-59	A
			Borated Water Leakage	None	None	III.B2-9 (TP-4)	3.5.1-59	A

TABLE 3.5.2-14 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade	C-2 C-3 C-4 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-30 (A-92)	3.3.1-66	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-31 (A-93)	3.3.1-67	A

TABLE 3.5.2-14 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-3 C-4 C-6 C-7 C-8	Reinforced Concrete	Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B1.1-1 III.B1.2-1 III.B2-1 III.B3-1 III.B4-1 III.B5-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-28 (A-90)	3.3.1-65	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-29 (A-91)	3.3.1-67	A

TABLE 3.5.2-14 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Below Grade	C-2 C-3 C-7 C-8	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Concrete: Foundation	C-2 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-14 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Damper Mountings	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B4-5 (TP-11)	3.5.1-58	A
Door (Non-fire)	C-7	Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 522
Door	C-3 C-4	Carbon Steel	Air - Indoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21) VII.G-4 (A-22)	3.3.1-63	E
Fire Barrier Assemblies	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation	Fire Protection			J, 526
Fire Barrier Penetration Seals	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-1 (A-19)	3.3.1-61	A
			Air - Outdoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-2 (A-20)	3.3.1-61	A
Fire Hose Stations	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522

TABLE 3.5.2-14 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Floor Drains	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	C, 522, 533
Phase Bus Duct Enclosure Assemblies	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	VI.A-13 (T-30)	3.6.1-09	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (LP-06)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-2 C-7 C-11	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A, 541
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	A, 533
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B5-3 (TP-11)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	A
			Borated Water Leakage	None	None	III.B5-6 (TP-4)	3.5.1-59	A, 533

TABLE 3.5.2-14 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-2 C-3 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B3-8 (T-25)	3.5.1-55	A, 533
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Roof-Membrane / Built-Up	C-3	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 529
Seals and Gaskets	C-2 C-3 C-7 C-8	Elastomer	Air - Indoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
Steel Components: All Structural Steel	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B5-8 (TP-25)	3.5.1-55	C, 522, 533

TABLE 3.5.2-14 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – INTERMEDIATE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for ASME Class 1, 2, 3 Piping & Components	C-2	Carbon Steel	Air - Indoor	Loss of Mechanical Function	ASME Section XI, Subsection IWF	III.B1.1-2 III.B1.2-2 (T-28)	3.5.1-54	A
				Loss of Material	ASME Section XI, Subsection IWF	III.B1.1-13 III.B1.2-10 (T-24)	3.5.1-53	A
		Fluorogold	Air - Indoor	Loss of Mechanical Function	ASME Section XI, Subsection IWF			F, 549
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-2 C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A, 518
		Elastomer	Air - Indoor	Reduction or Loss of Isolation Function	Structures Monitoring	III.B4-12 (T-31)	3.5.1-41	A, 538
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Borated Water Leakage	Loss of Material	Boric Acid Corrosion	III.B2-11 (T-25)	3.5.1-55	A, 533

TABLE 3.5.2-15 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – MACHINE SHOP

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Aluminum	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
Door	C-4	Carbon Steel	Air - Indoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21) VII.G-4 (A-22)	3.3.1-63	E
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-7	Elastomer	Air - Outdoor	Reduction or Loss of Isolation Function	Structures Monitoring	III.B4-12 (T-31)	3.5.1-41	A, 538
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air-Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A

TABLE 3.5.2-16 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – MISCELLANEOUS STRUCTURES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-2 C-6 C-7 C-8	Carbon Steel	Reinforced Concrete	None	None			J, 501
		Stainless Steel	Reinforced Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
Concrete: Above Grade	C-2 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A

TABLE 3.5.2-16 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – MISCELLANEOUS STRUCTURES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-2 C-6 C-7 C-8	Reinforced Concrete	Air - Outdoor	Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 III.B5.1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-30 (A-92)	3.3.1-66	A, 534
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-31 (A-93)	3.3.1-67	A, 534
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 III.A5-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 III.A5-10 (T-06)	3.5.1-24	A

TABLE 3.5.2-16 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – MISCELLANEOUS STRUCTURES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Below Grade	C-2 C-6 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Concrete: Foundation	C-2 C-6 C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-16 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – MISCELLANEOUS STRUCTURES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Door (Non-fire)	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Fire Hose Stations	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-2 C-6 C-7 C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
			Soil	Loss of Material	One-Time Inspection			J, 523
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532

TABLE 3.5.2-16 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – MISCELLANEOUS STRUCTURES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
		Stainless Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Seals and Gaskets	C-8	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
Steel Components: All Structural Steel	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A

TABLE 3.5.2-17 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – SWITCHYARD FOR CRYSTAL RIVER SITE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Battery Racks	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
Concrete: Above Grade	C-7	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A

TABLE 3.5.2-17 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – SWITCHYARD FOR CRYSTAL RIVER SITE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-7	Reinforced Concrete	Air - Outdoor	Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 (T-29)	3.5.1-40	A
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 III.B5-1 (T-29)	3.5.1-40	A
Concrete: Below Grade	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-17 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – SWITCHYARD FOR CRYSTAL RIVER SITE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Foundation	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-7	Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A

TABLE 3.5.2-17 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – SWITCHYARD FOR CRYSTAL RIVER SITE

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Steel Components: All Structural Steel	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A
		Galvanized Carbon Steel	Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532

TABLE 3.5.2-18 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – SWITCHYARD RELAY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
Concrete: Above Grade	C-7	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 (T-29)	3.5.1-40	A

TABLE 3.5.2-18 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – SWITCHYARD RELAY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Below Grade	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Concrete: Foundation	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-18 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – SWITCHYARD RELAY BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Door (Non-fire)	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Masonry Walls	C-7	Concrete Block	Air - Indoor	Cracking	Masonry Wall	III.A3-11 (T-12)	3.5.1-43	A
			Air - Outdoor	Cracking	Masonry Wall	III.A3-11 (T-12)	3.5.1-43	A
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A
Roof-Membrane / Built-Up	C-7	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 529
Steel Components: All Structural Steel	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A

TABLE 3.5.2-19 CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – TURBINE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Anchorage / Embedment	C-7	Carbon Steel	Reinforced Concrete	None	None			J, 501
		PVC	Reinforced Concrete	None	None			J, 501
Battery Racks	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
Cable Tray, Conduit, HVAC Ducts, Tube Track	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B2-5 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B2-4 (TP-8)	3.5.1-58	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-7 (TP-6)	3.5.1-50	A
Concrete: Above Grade	C-4 C-7 C-8	Reinforced Concrete	Air - Outdoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 (T-29)	3.5.1-40	A

TABLE 3.5.2-19 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – TURBINE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Above Grade (continued)	C-4 C-7 C-8	Reinforced Concrete	Air - Outdoor	Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-30 (A-92)	3.3.1-66	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-31 (A-93)	3.3.1-67	A
			Air - Indoor	Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-9 (T-04)	3.5.1-23	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-10 (T-06)	3.5.1-24	A
				Reduction in concrete anchor capacity due to local concrete degradation	Structures Monitoring	III.B2-1 III.B3-1 III.B5-1 (T-29)	3.5.1-40	A
				Cracking Loss of Material	Fire Protection and Structures Monitoring	VII.G-28 (A-90)	3.3.1-65	A
				Loss of Material	Fire Protection and Structures Monitoring	VII.G-29 (A-91)	3.3.1-67	A

TABLE 3.5.2-19 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – TURBINE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Concrete: Below Grade	C-7 C-8	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A
Concrete: Foundation	C-7	Reinforced Concrete	Soil	Cracking	Structures Monitoring	III.A3-3 (T-08)	3.5.1-28	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-4 (T-05)	3.5.1-31	A
				Loss of Material Cracking Change in Material Properties	Structures Monitoring	III.A3-5 (T-07)	3.5.1-31	A
				Change in Material Properties	Structures Monitoring	III.A3-7 (T-02)	3.5.1-32	A

TABLE 3.5.2-19 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – TURBINE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Damper Mountings	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B4-5 (TP-11)	3.5.1-58	A
Door (Non-Fire)	C-3 C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Door	C-4	Carbon Steel	Air - Indoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21) VII.G-4 (A-22)	3.3.1-63	E
			Air - Outdoor	Loss of Material	Fire Protection and Structures Monitoring	VII.G-3 (A-21) VII.G-4 (A-22)	3.3.1-63	E
Fire Barrier Penetration Seals	C-4	Fire Proofing Materials	Air - Indoor	Loss of Material Cracking / Delamination Separation Change in Material Properties	Fire Protection	VII.G-1 (A-19)	3.3.1-61	A
			Air - Outdoor	Loss of Material Cracking Delamination Separation Change in Material Properties	Fire Protection	VII.G-2 (A-20)	3.3.1-61	A

TABLE 3.5.2-19 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – TURBINE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Fire Hose Stations	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
Floor Drains	C-8	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	C, 522
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	C, 522
Masonry Walls	C-7	Concrete Block	Air - Indoor	Cracking	Masonry Wall	III.A3-11 (T-12)	3.5.1-43	A
			Air - Outdoor	Cracking	Masonry Wall	III.A3-11 (T-12)	3.5.1-43	A
Phase Bus Duct Enclosure Assemblies	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	VI.A-13 (LP-06)	3.6.1-09	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and Other Miscellaneous Structures	C-7 C-8 C-11	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B5-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B5-3 (TP-11)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B5-5 (TP-5)	3.5.1-59	A

TABLE 3.5.2-19 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – TURBINE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B3-7 (T-30)	3.5.1-39	A
		Galvanized Carbon Steel	Air - Indoor	None	None	III.B3-3 (TP-11)	3.5.1-58	A
		Aluminum	Air - Indoor	None	None	III.B3-2 (TP-8)	3.5.1-58	A
		Stainless Steel	Air - Indoor	None	None	III.B3-5 (TP-5)	3.5.1-59	A
Roof-Membrane / Built-up	C-7	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 529
Seals and Gaskets	C-8	Elastomer	Air - Outdoor	Cracking Change in Material Properties	Structures Monitoring	III.A6-12 (TP-7)	3.5.1-44	C, 509
Siding	C-7	Aluminum	Air - Indoor	None	None	III.B5-2 (TP-8)	3.5.1-58	C, 522
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B4-7 (TP-6)	3.5.1-50	C, 532
Steel Components: All Structural Steel	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.A3-12 (T-11)	3.5.1-25	A

TABLE 3.5.2-19 (continued) CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORT - SUMMARY OF AGING MANAGEMENT EVALUATION – TURBINE BUILDING

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Supports for EDG, HVAC System Components, and Other Miscellaneous Mechanical Equipment	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B4-10 (T-30)	3.5.1-39	A, 518
		Elastomer	Air - Indoor	Reduction or Loss of Isolation Function	Structures Monitoring	III.B4-12 (T-31)	3.5.1-41	A, 538
Supports for Non-ASME Piping & Components	C-7	Carbon Steel	Air - Indoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
			Air - Outdoor	Loss of Material	Structures Monitoring	III.B2-10 (T-30)	3.5.1-39	A
		Stainless Steel	Air - Indoor	None	None	III.B2-8 (TP-5)	3.5.1-39	A

Notes for Tables 3.5.2-1 through 3.5.2-19:

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 AMP is credited or NUREG-1801 identifies a plant-specific AMP.
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:

501. The CR-3 aging management review methodology concluded that carbon/low alloy steel, galvanized carbon steel, stainless steel, and PVC completely encased in concrete has no aging effect. There is not a corresponding component/commodity in NUREG-180,1 Chapter II or III. However, NUREG-1801 Items RP-01, RP-06, EP-5, EP-20, SP-2, SP-13, and AP-19 which apply to carbon steel and stainless steel mechanical piping and components embedded in concrete validate there are no aging effects for stainless steel and carbon steel embedded in concrete.

502. The aging management review methodology concluded that Thermo-lag in this environment is susceptible to the aging effect of loss of material and cracking and/or delamination, and separation.

503. The expansion bellows is between the Fuel Transfer Tube and the transfer canal inside the Reactor Building. The expansion bellows is not a "Penetration sleeve; Penetration bellows" from NUREG-1801, II.A3-4, but is of the same material, in the same environments, has the same aging effects, and is a TLAA which is consistent with the aging management program column in NUREG-1801. This expansion bellows was determined to have a design cycle life and is evaluated as a TLAA. The expansion bellows is subject to an Air - Indoor environment normally but is subject to a treated water environment while the Reactor Building Refueling Canal is flooded during reactor fuel movements.

504. The following: Cranes; Expansion Bellows; Fire Barrier Assemblies (Wrap only), and Steel Components: Fuel Pool Liner are aligned with NUREG-1801, Item III.B5-5 or III.B5-6, as a "Miscellaneous Structure;" because they have the same material, environment, aging effect and aging management program, although they are not the same NUREG-1801 component "Support members; welds, bolted connections, support anchorage to building structure."

505. The Expansion Bellows is aligned with NUREG-1801, Item III.A5-13, as "Steel Components: Fuel Pool Liner" because it has the same material, environment, aging effect and aging management program; although it is not the same NUREG-1801 component.
506. The aging management review methodology concluded that the Insulation on piping in mechanical penetrations, in the Air-Indoor environment, has no aging effects.
507. The Moisture Barrier is the only part of the "Seals, Gaskets, and Moisture Barriers" commodity to which the ASME Section XI, Subsection IWE Program applies.
508. The components Floor Drains, or Steel Components: All structural steel, or Penetration Sleeves or Steel Elements: Liner; Liner Anchors; Integral Attachments are aligned with NUREG-1801, III.B5-8, as a "Miscellaneous Structure;" because they have the same material, environment, aging effect and aging management program, although they are not the same NUREG-1801 component "Support members; welds; bolted connections; support anchorage to building structure."
509. The Seals and Gaskets component is aligned with NUREG-1801, III.A6-12, because it has the same material, environment, aging effect and aging management program although it is not a NUREG-1801 Group 6 Water Control Structure. There is not a NUREG-1801 Group 3, 4, 5 or Group 8 Seals and Gaskets component provided in NUREG-1801. Cracking and change in material properties for elastomers results in loss of sealing and is considered an equivalent aging effect.
510. The same aging effect used for NUREG-1801, Items III.A4-6, for Lubrite plates (Lock-up) is assigned to Fluorogold slide bearing plates used on structural steel. In addition, CR-3 determined change in material properties due to radiation is an applicable aging effect. The Structures Monitoring Program is credited for inspecting the slide bearing plates which includes the Fluorogold plates.
511. The same aging effect used for NUREG-1801, Items III.B1.1-5 and III.B1.2-3, for Lubrite plates (loss of mechanical function) is assigned to Fluorogold slide bearing plates used on supports. In addition, CR-3 determined change in material properties due to radiation is an applicable aging effect. The ASME Section XI, Subsection IWF Program is credited for inspecting the slide bearing plates which includes the Fluorogold plates.
512. NUREG-1801 item III.A5-13 is for the Fuel Pool Liner. This NUREG-1801 line item was selected because the Reactor Cavity liner / Refueling Canal liner in the Reactor Building has the same material, environment (during refueling), aging effects, and aging management programs as the Fuel Pool Liner.
513. Component Type includes the exterior surface of the stainless steel fuel transfer tubes, blind flanges, bolting, and cover plates and the dissimilar metal welds at the stainless steel fuel transfer tube / carbon steel penetration sleeve interface located in the Reactor Building because the fuel transfer tube is examined by the ASME Section XI, Subsection IWE Program and the 10 CFR Part 50, Appendix J Program. The aging management review methodology concluded cracking due to SCC in the Air-indoor environment was not applicable because the stainless is not exposed to an aggressive environment.
514. NUREG-1801 item II.A1-7 discusses groundwater/soil environment but groundwater/soil environment is not listed in the environment column. A Soil environment has been applied in the environment column similar to Item III.A3-4.
515. Includes the RB Polar Crane, Reactor Vessel Tool Handling Jib Crane, 5-Ton Jib Crane, Control Rod Drive Mechanism Jib Crane, and the Main Fuel Handling Bridge Crane.
516. The component Floor Drains is aligned with NUREG-1801, Volume 2, Item III.B5-7, as "Miscellaneous Structures;" because it has the same material, environment, aging effect and aging management program, although it is not the same NUREG-1801 component, i.e., "Support members; welds; bolted connections; support anchorage to building structure."

517. The aging management review methodology concluded that galvanized steel in an Air-Outdoor environment is susceptible to the aging effect of Loss of Material. NUREG 1801, Groups B3, B5, and A3-12, do not have an equivalent material/environment combination that can be aligned to. Therefore, the miscellaneous structures noted are aligned with III.B4-7 to obtain a consistent aging effect and aging management program.
518. Includes only metallic vibration isolators for the ventilation equipment.
519. Draft Stops are sheet-metal curtains located around stairwell ceilings that have a 10 CFR 54.4(a)(2) intended function.
520. The CR-3 aging management review methodology concluded that loss of material (due to abrasion; cavitation) is an applicable aging effect/mechanism. This commodity is aligned with NUREG 1801, III.A6-7, because it has the same material, environment, aging effect, and aging management program, although it is not the same NUREG-1801 component.
521. The Seals and Gaskets component for the sealing/caulking of the BWST to its concrete shield wall is aligned with NUREG-1801, III.A6-12, because it has the same material, environment, aging effect, and aging management program although it is not a NUREG-1801 Group 6 Water Control Structure. Cracking and change in material properties for elastomers results in loss of sealing and is considered an equivalent aging effect.
522. The components "Door (Non-Fire)" or "Floor Drains" or "Fire Hose Stations" or "Steel Components: All structural steel" or "Steel Components: Fuel Pool Liner" or "Raised Floor" or "Siding" or "Draft Stops" are aligned with Other Miscellaneous Structures (Grouping III.B5); because they are the same materials, environments, aging effects and aging management programs, although they are not the same NUREG-1801 component.
523. The Diesel Fuel Oil Tank is held in place by steel hold-down straps that are buried in soil and, therefore, inaccessible. For the purpose of the aging management review, the hold-down straps were considered as a miscellaneous structure; and no credit was taken for the coal-tar epoxy equivalent to Bitumastic 300-M coating provided. The evaluation determined that the hold-down straps would have an aging effect of loss of material without credit for the coating. The coated hold-down straps were originally installed for the current 40-year licensing period, and it is concluded that a One-Time inspection be performed for the strap area at the top of the tank just prior to extended life (within two years of 2016). The One-Time inspection is used to determine that the hold-down straps do not have any deficiencies that could prevent the straps from performing their function.
524. Aging effects are not applicable for "Unreinforced Concrete or Fabriform."
525. The CR-3 aging management review methodology concluded that copper materials in an Air - Indoor or Borated Water Leakage environment have no aging effect. This applies only to straps for copper tubing.
526. Fire Barrier Assemblies include Thermo-lag or Mecatiss Fire Barriers.
527. The CR-3 methodology concluded that stainless steel conduits and support steel located in the Dedicated EFW Tank Enclosure Building northwest corner recessed area (similar to a sump) will have the aging effect of Loss of Material.
528. The CR-3 aging management review methodology determined that there are no aging effects for Boral. There has been no adverse operating experience recorded for CR-3 or Harris Nuclear Plant. Both the V. C. Summer Nuclear Station and the Brunswick Steam Electric Plant have been evaluated by the NRC staff for these aging effects, and the License Renewal Safety Evaluation Reports for those plants have determined the aging effects to be insignificant.
529. The Roof-Membrane / Built-up component is aligned with NUREG-1801, III.A6-12, because it has the same material, environment, aging effect and aging management program although it is not a NUREG-1801 Group 6 Water Control Structure. There is not a NUREG-1801 Group 3 or Group 4 Seals and Gaskets, and Moisture Barrier component provided in NUREG-1801. Cracking and change in material properties for elastomers results in loss of sealing and is considered an equivalent aging effect.

530. The CR-3 aging management review methodology concluded that there are negligible aging effects associated with the Control Room ceiling Willtec foam panels. Additionally, plant operating experience has identified no aging effects.
531. NUREG-1801 allows use of the "Structures Monitoring Program" instead of the "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Plants" aging management program if the plant is not committed to RG 1.127.
532. The components "Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Wall Supports, and other Miscellaneous Structures", "Siding", "Steel Components: All Structural Steel," "Door (Non-fire)," or "Racks, Panels, Cabinets, and Enclosure for Electrical Equipment and Instrumentation" are aligned with "Supports for EDG, HVAC System Components and Other Miscellaneous Mechanical Equipment;" because they have the same materials, environments, aging effects, and aging management programs, although they are not the same NUREG-1801 component.
533. The Borated Water Leakage environment applies only to the Nuclear Sampling Room.
534. The Fire Protection intended function only applies to the Transformer area 3-hour firewall.
535. For the Auxiliary Building, includes the 120-Ton Fuel Handling Area Crane, the Spent Fuel Pit Missile Shield Crane, and the Spent Fuel Pool Handling Bridge Crane. For the EFW Pump Building, includes the EFW Pump Building 3-Ton Crane. For the Circulating Water Intake Structure, includes the Intake Gantry Crane.
536. The Seals and Gaskets commodity group only applies to caulking for the concrete plugs on the Auxiliary Building roof (Elevation 119 ft.-0 in.) south of the BWST.
537. The CR-3 aging management review methodology concluded that stainless steel in Raw Water - Seawater has an aging effect of loss of material for the Trash Racks and associated support structure.
538. This line item includes only non-metallic vibration isolators for the ventilation system.
539. This line item applies to the Machine Room concrete walls at Control Complex elevation 181 ft.-4 in. only.
540. The CR-3 aging management review methodology determined that Carborundum (B_4C) has the aging effect Loss of Material, which will be managed by the Carborundum (B_4C) Monitoring Program.
541. This line item includes the Drain Trench Weir.
542. The expansion bellows is between the Fuel Transfer Tube and the Spent Fuel Pool liner plate located inside the Auxiliary Building. The expansion bellows is not a "Penetration sleeve; Penetration bellows" from NUREG-1801, II.A3-4, but is of the same material, in the same environments, has the same aging effects, and is a TLAA, which is consistent with the aging management program column in NUREG-1801. This expansion bellows was determined to have a design cycle life and is evaluated as a TLAA. The expansion bellows is subject to a treated water environment.
543. NUREG-1801 only addresses "corrosion of embedded steel" or "aggressive chemical attack" for a ground water/soil environment. In Group 3 and Group 6 structures, NUREG-1801 does not address "corrosion of embedded steel" or "aggressive chemical attack" in a Raw Water environment. CR-3 Group 3 and Group 6 structures have concrete components in a Raw Water environment. Raw Water - Seawater is aggressive at CR-3 because the sulfate content is greater than 1500 ppm and the chloride content is greater than 500 ppm. Concrete cracking, loss of material, and change in material properties are applicable aging effects for the submerged concrete. The Structures Monitoring Program is used to manage aging effects of submerged concrete for loss of material, cracking, and change in material properties.
544. NUREG-1801 Group 8 structures do not address corrosion of embedded steel and aggressive chemical attack in Air-Indoor or Air-Outdoor environments. The CR-3 methodology used NUREG-1801 Group 3 structures for potential aging effects to concrete.

545 Concrete in a treated water environment in the Dedicated EFW Tank Enclosure Building north-west corner recessed area has been conservatively evaluated as a groundwater environment to provide potential aging effects.

546. The CR-3 aging management review methodology concluded that carbon steel in a treated water environment has the aging effect of Loss of Material.

547. The CR-3 aging management review methodology conservatively applied Loss of Mechanical Function to a Treated Water environment to agree with Air-Indoor and Air-Outdoor environments.

548. This item includes the anchor bolts for Reactor Coolant System (Reactor Pressure Vessel, Reactor Coolant Pump, Pressurizer, and Steam Generator Lateral Supports) components.

549. The same aging effect used for NUREG-1801 Items III.B1.1-5 and III.B1.2-3 for Lubrite plates (loss of mechanical function) is assigned to Fluorogold slide bearing plates used on supports. The ASME Section XI, Subsection IWF or Structures Monitoring Program is credited for inspecting the slide bearing plates which includes the Fluorogold plates.

550 The Reinforced Concrete material for the Wave Embankment Structure includes any unreinforced concrete and the Fabriform used on the slope of the Berm.

551 The same aging effect used for NUREG-1801 Item III.A4-6 for Lubrite plates (i.e., Lock-up) is assigned to Fluorogold slide bearing plates used on structural steel. The Structures Monitoring Program is credited for inspecting the slide bearing plates which includes the Fluorogold plates.

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3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

3.6.1 INTRODUCTION

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

1. Non-Environmentally Qualified (EQ) Insulated Cables and Connections (Subsection 2.5.4.1)

   The commodity Non-EQ Insulated Cables and Connections was divided into the following groups in order to better align with the component types in NUREG-1801 and to include plant-specific commodities. These groups are used in the AMR summary table referenced in Subsection 3.6.2, Results, below:
   a. Cable connections-metallic parts, This commodity group corresponds to cable connections (metallic parts), item VI.A-1 from NUREG-1801,
   b. Insulated Cables and Connections. This commodity group corresponds to conductor insulation for electrical cables and connections, Item VI.A-2 from NUREG-1801,
   c. Cables and Connections Used in Instrumentation Circuits. This group aligns with conductor insulation for electrical cables and connections used in instrumentation circuits that are sensitive to reduction in conductor insulation resistance (IR), Item VI.A-3 from NUREG-1801,
   d. Medium-voltage Power Cables. This corresponds to conductor insulation for inaccessible medium-voltage cables, Item VI.A-4 from NUREG-1801,
   e. Electrical Connector Contacts Exposed to Borated Water Leakage, which corresponds to connector contacts for electrical connectors exposed to borated water leakage, Item VI.A-5 from NUREG-1801, and
   f. Fuse Holders. This commodity corresponds to those addressing insulation and metallic parts of fuse holders (not part of a larger assembly), Items VI.A-6, -7 and -8 from NUREG-1801.

2. Electrical Portions of Non-EQ Electrical/I&C Penetration Assemblies (Subsection 2.5.4.2)

3. Metal Enclosed Bus and Connections (Subsection 2.5.4.3)

   The commodity Metal Enclosed Bus (MEB) and Connections was divided into the following groups in order to better align with the component types in NUREG-1801.
   a. MEB - Bus/Connections, which align with NUREG-1801, Item VI.A-11,

b. MEB - Enclosure Assemblies (elastomers) which align with NUREG-1801, Item VI.A-12,
c. MEB - Enclosure Assemblies (metal), which align with NUREG-1801, Item VI.A-13 (Refer to Table 3.6.1, Item 3.6.1-09.).
d. MEB - Insulation/Insulators, which align with NUREG-1801, Item VI.A-14.

4. High-voltage Insulators (Subsection 2.5.4.4)

5. Switchyard Bus and Connections (Subsection 2.5.4.5)

6. Transmission Conductors and Connections (Subsection 2.5.4.6)

Table 3.6.1, Summary of Aging Management Evaluations in Chapter VI of NUREG-1801 for Electrical Components, provides the summary of the programs evaluated in NUREG-1801 that are applicable to component/commodity groups in this Section. Table 3.6.1 uses the format of Table 1 described in Section 3.0 above.

3.6.1.1 Operating Experience

The AMR methodology applied at CR-3 included use of operating experience (OE) to confirm the set of aging effects that had been identified through material/environment evaluations. Plant-specific and industry OE was identified and reviewed. The OE review consisted of the following:

Site: The review of site-specific, CR-3 OE included a review of the Action Tracking Database, Maintenance Rule documentation, Licensee Event Reports, the CR-3 System Notebooks, and interviews with CR-3 engineering personnel. The site-specific OE review identified no additional or unique aging effects requiring management.

Industry: Industry OE was obtained from SAND96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations," September 1996, which consolidates historical maintenance and industry OE for evaluation of aging mechanisms and effects. Additional generic OE was obtained in Revision 1 of NUREG-1801, "Generic Aging Lessons Learned (GALL)," U. S. Nuclear Regulatory Commission, September 2005. Draft Revision 1 of NUREG-1801 was issued in January 2005; more recent OE was captured by means of the Progress Energy OE review process and by a review of correspondence such as NRC Information Notices and Generic Letters, 10 CFR 21 reports, and vendor and INPO publications. The industry OE review identified no additional aging effects requiring management.

On-Going — On-going review of plant-specific and industry operating experience is continuing to be performed in accordance with the Corrective Action Program and the Progress Energy internal OE review process.

3.6.2 RESULTS

The following table summarizes the results of the aging management review for components/commodities in the Electrical and I&C Systems area.

Table 3.6.2-1 Electrical and I&C Systems – Summary of Aging Management Evaluation – Electrical/I&C Components/Commodities

This table uses the format of Table 2 described in Section 3.0.

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

The materials from which specific components/commodities are fabricated, the environments to which they are exposed, the aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each component/commodity in the following subsections.

3.6.2.1.1 Non-EQ Insulated Cables and Connections

Materials

The primary cable and connection insulation materials are:

I&C Cable Insulation
- PE
- EPR, EP
- SR
- FEP, Teflon
- XLP, XLPE, XLPO
- PVC
- EPDM
- ETFE
- Kerite FR, (HI-70)
- Kerite HTK
- Kerite FR3

Power Cable Insulation
- EPR, EP
- TRXLPE
- Kerite FR, (HI-70)
- Kerite HTK
- Kerite FR3

Connections
- EPR, EP
- XLP, XLPE, XLPO
- Melamine
- Nylon
- Phenolic
- Porcelain
- Kapton
- EPDM

The materials of construction for metallic parts of electrical connections are:
- Various metals
- Copper Alloy

Environment

The Non-EQ Insulated Cables and Connection components may be exposed to:

- Air - Indoor
- Air - Outdoor
- Adverse Localized Environment caused by Heat, Radiation, or Moisture in the Presence of Oxygen
- Adverse Localized Environment Caused by Exposure to Moisture and Voltage
- Air with Borated Water Leakage

An adverse, localized environment is defined as a condition in a limited plant area that is significantly more severe than the specified service condition for the equipment.

Aging Effects Requiring Management

The Non-EQ Insulated Cables and Connection components are subject to the following aging effects requiring management:

- Loosening of Bolted Connections
- Reduced Insulation Resistance
- Electrical Failure (breakdown of insulation)
- Corrosion of Connector Contact Surfaces
- Loss of Continuity

Aging Management Programs

The following AMPs manage the aging effects for the Non-EQ Insulated Cables and Connection components:

- Boric Acid Corrosion Program
- Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program
- Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program
- Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program
- Fuse Holder Program
- Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

3.6.2.1.2 Electrical Portions of Non-EQ Electrical/I&C Penetration Assemblies

Materials

The materials of construction for the Electrical Portions of Non-EQ Electrical/I&C Penetration Assemblies components are:

- XLPO
- SR
- Kapton
- EPDM
- CSPE
- EPR
- Kynar (PVDF)

Environment

The Electrical Portions of Non-EQ Electrical/I&C Penetration Assemblies are exposed to the following:

- Adverse Localized Environment caused by Heat, Radiation, or Moisture in the Presence of Oxygen

Aging Effects Requiring Management

The Electrical Portions of Non-EQ Electrical/I&C Penetration Assemblies components are subject to the following aging effects requiring management:

- Reduced Insulation Resistance
- Electrical Failure (breakdown of insulation)

Aging Management Programs

The following AMP manages the aging effects for the Electrical Portions of Non-EQ Electrical/I&C Penetration Assemblies components:

- Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

3.6.2.1.3 Metal Enclosed Bus and Connections

Materials

The materials of construction for the Metal Enclosed Bus and Connections components are:

- Aluminum
- Copper
- Elastomers

- Fiberglass
- Phenolic
- Porcelain
- Organic Polymers
- Silver-Plated Aluminum
- Steel

Environment

The Metal Enclosed Bus and Connections components are exposed to the following:
- Air - Indoor
- Air - Outdoor

Aging Effects Requiring Management

The Metal Enclosed Bus and Connections components are subject to the following aging effects requiring management:
- Loosening of Bolted Connections
- Hardening and Loss of Strength
- Reduced Insulation Resistance
- Electrical Failure

Aging Management Programs

The following AMP manages the aging effects for the Metal Enclosed Bus and Connections components:
- Metal Enclosed Bus Program

3.6.2.1.4 High-Voltage Insulators

Materials

The materials of construction for the High-Voltage Insulators are:
- Porcelain
- Galvanized metals
- Cement

Environment

The High-Voltage Insulators are exposed to the following:
- Air - Outdoor

Aging Effects Requiring Management

The High-Voltage Insulator components are subject to the following aging effects requiring management:

- Degradation of Insulation Quality
- Loss of Material

Aging Management Programs

The following AMP manages the aging effects for the High-Voltage Insulator components:

- High-Voltage Insulators in the 230KV Switchyard Program

3.6.2.1.5 Switchyard Bus and Connections

Materials

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

- Aluminum
- Stainless Steel
- Galvanized Steel

Environment

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

- Air - Outdoor

Aging Effects Requiring Management

The AMR determined that the aging effects for the Switchyard Bus and Connections components are not significant and require no aging management activities.

Aging Management Programs

The AMR determined that no aging management activities are required for the Switchyard Bus and Connections components.

3.6.2.1.6 Transmission Conductors and Connections

Materials

The materials of construction for the Transmission Conductors and Connections components are:

- Aluminum
- Steel

Environment

The Transmission Conductors and Connections components are exposed to the following:

- Air - Outdoor

Aging Effects Requiring Management

The AMR determined that the aging effects for the Transmission Conductors and Connections components are not significant and require no aging management activities.

Aging Management Programs

The AMR determined that no aging management activities are required for the Transmission Conductors and Connections components.

3.6.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 provides the basis for identifying those programs that warrant further evaluation by the reviewer in the LRA. For the Electrical and I&C Systems, those programs are addressed in the following subsections.

3.6.2.2.1 Electrical Equipment Subject to Environmental Qualification

As discussed in Section X.E1 of NUREG-1801, aging evaluations performed in accordance with the Environmental Qualification (EQ) Program may involve a time-limited aging analysis (TLAA) as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of EQ TLAAs is addressed separately in Section 4.4.

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

Salt and Surface Contamination

In accordance with NUREG-1801, degradation of insulator quality due to the presence of any salt deposits and surface contamination could occur in high-voltage insulators. Various airborne materials such as dust, salt and industrial effluents can contaminate insulator surfaces. Surface contamination can be a problem in areas where there are greater concentrations of airborne particles due to proximity to facilities that discharge soot or proximity to the ocean where salt spray is prevalent. A large buildup of contamination enables the conductor voltage to track along the surface more easily and can lead to insulator flashover. The buildup of surface contamination is typically a slow, gradual process that is even slower for rural areas with generally less suspended particles and SO_2 concentrations in the air than urban areas. Although CR3 is located in a rural area, it is in close proximity to the Gulf of Mexico. Site OE has shown that flashover of overhead transmission line insulators due to contamination from salt spray is an applicable aging mechanism that requires management. This aging mechanism is not applicable to the station post insulator in the 230KV Switchyard. Flashover of station post insulators has not been experienced at CR3. This is attributed to the fact that station post insulators are oriented vertically whereas the overhead transmission line insulators may be angled to form various "string" configurations depending on the design application making them more susceptible to surface contamination. Also, the overall length of a station post insulator is often longer than that of overhead transmission line insulators to meet the necessary clearance requirements for personnel safety in the switchyard. The longer overall length of a station post insulator increases the "creepage distance" of the insulator making it less susceptible to surface contamination.

Therefore, a plant-specific High-Voltage Insulators in the 230KV Switchyard Program will be implemented to preclude the buildup of surface contamination on overhead transmission line insulators in the 230KV Switchyard.

Mechanical Wear

As stated in NUREG-1801, loss of material due to mechanical wear caused by wind could occur in high-voltage insulators. Loss of material due to mechanical wear is an aging effect for strain and suspension insulators if they are subject to significant movement. Movement of the insulators can be caused by wind blowing the supported transmission conductor, causing it to swing from side to side. If this swinging is frequent enough, it could cause wear in the metal contact points of the insulator string and between an insulator and the supporting hardware. Site OE has shown that mechanical wear resulting in loss of material to the steel pins connecting the insulators to one another is an applicable aging effect that requires management for the overhead transmission line insulators in the 230KV Switchyard. This aging mechanism is not applicable to the station post insulator in the 230KV Switchyard. Station post insulators do not have steel swivel points like overhead transmission line insulators and are not susceptible to mechanical wear due to their mounting configuration. Therefore, a plant-specific High-Voltage Insulators in the 230KV Switchyard Program will be implemented to mitigate this aging mechanism on overhead transmission line high-voltage insulators in the 230KV Switchyard.

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 Preload

Switchyard Bus and Connections

The switchyard buses within the scope of this review consist of 4 in. integral web channel bus (IWCB) constructed of rectangular aluminum. The switchyard buses are connected to short lengths of flexible conductors that do not normally vibrate and are supported by station post insulators mounted to static, structural components such as cement footings and structural steel. Based on this design configuration, wind induced vibration is not an applicable aging mechanism.

With no connections to moving or vibrating equipment, loss of material due to vibration is not an aging effect requiring management. Aluminum bus exposed to the service conditions of the 230KV Switchyard does not experience any appreciable aging effects, except for minor oxidation, which does not impact the ability of the switchyard bus to perform its intended function. Therefore, it is concluded that general corrosion resulting in the oxidation of the switchyard bus is not an aging effect requiring management.

The bolted connections associated with the switchyard bus are for the connections to station post insulators used to support the bus. Other connections to the bus are welded. The components involved in switchyard bus connections are constructed from cast aluminum, galvanized steel and stainless steel. No organic materials are involved. The station post insulators used to support the switchyard bus are bolted to the channel on the underside of the IWCB utilizing either galvanized or stainless steel bolts. Components in the 230KV Switchyard are exposed to precipitation. Connection materials exposed to the service conditions of the 230KV Switchyard do not experience any appreciable aging effects, except for minor oxidation, which does not impact the ability of the switchyard bus to perform its intended function. The steel bolting hardware used in this application has been selected because of its ability to inhibit rust. Based on operating experience, corrosion of the structural bolting used in this application is not significant enough to cause a loss of intended function.

Transmission Conductors and Connections

Transmission conductor mounting hardware loss of material due to wind induced abrasion and fatigue is an applicable aging mechanism but is not significant enough to cause a loss of intended function. Wind induced abrasion and fatigue could be caused by transmission conductor vibration resulting from wind loading. However, a high wind loading factor of 135 mph (with an additional safety factor for wind gusts) has been considered in the design and installation of transmission conductors in the CR-3 transmission and distribution network. Strong winds could cause the transmission conductors to sway from side to side. If this swinging is frequent enough, it could cause the transmission conductor's mounting hardware to wear. Although this mechanism is possible, experience has shown that the transmission conductors do not normally swing and when they do, because of strong winds, they dampen quickly once the wind has subsided. Therefore, it is concluded that mounting hardware loss of material caused by transmission conductor vibration (sway) and fatigue is not an aging effect requiring management.

Loss of transmission conductor strength due to corrosion is an applicable aging effect but ample design margin ensures that it is not significant enough to cause a loss of intended function. All CR-3 transmission conductors are Type ACSR (aluminum conductor steel reinforced). They are constructed of strand aluminum conductors wound around a steel core. No organic materials are involved. The most prevalent mechanism contributing to loss of conductor strength of an ACSR transmission conductor is corrosion, which includes corrosion of the steel core and aluminum strand pitting. For ACSR transmission conductors, degradation begins as a loss of zinc from the galvanized steel core wires. Corrosion rates depend largely on air quality, which includes suspended particles chemistry, SO_2 concentration in air, precipitation, fog chemistry, and meteorological conditions. Corrosion of ACSR transmission conductors is a very slow process that is even slower for rural areas with generally less suspended particles and SO_2 concentrations in the air than urban areas. CR-3 is located in a rural area where airborne particle concentrations are comparatively low. Consequently, this is not considered a significant contributor to this aging mechanism. There is a set percentage of composite conductor strength established at which a transmission conductor is replaced. 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 a conductor must be designed to withstand under heavy load requirements, which includes consideration of ice, wind, and temperature. Tests performed by Ontario Hydroelectric showed a 30% loss of composite conductor strength of an 80-year-old transmission conductor due to corrosion. Assuming a 30% loss of strength, there would still be significant margin between what is required by the NESC and actual conductor strength. These requirements were evaluated for applicability to the specific transmission conductors used at CR-3. CR-3 is in the light loading zone; therefore, the Ontario Hydroelectric heavy loading zone study is conservative. A typical 954 MCM ACSR transmission conductor used in the 230KV Switchyard will be used as an illustration. The ultimate strength of a 954 MCM (24/7 strand) ACSR conductor is 33,500 lbs and the maximum design tension for this conductor is 15,000 lbs. The margin between the maximum design tension and the ultimate strength is 18,500 lbs.; i.e., there is a 55.2% ultimate strength margin (18,500/33,500). The Ontario Hydroelectric study showed a 30% loss of composite conductor strength in an 80-year-old conductor. In the case of the CR-3 954 MCM ACSR transmission conductor, a 30% loss of ultimate strength would mean there would still be 25.2% ultimate strength margin between what is required by the NESC and the actual conductor strength in an 80-year old conductor. The CR-3 transmission conductors within the scope of this review have relatively short spans. Therefore, the tension exerted on the conductors in the 230KV Switchyard is less than would be experienced in typical transmission applications, which could be up to 1000 feet in length. This evaluation shows that there is ample design margin in the transmission conductors at CR-3. This analysis shows that the Ontario Hydroelectric test envelops the transmission conductors at CR-3; and, based on the conservatism in ultimate strength margin, demonstrates that loss of conductor strength is not an aging effect requiring management for the ACSR transmission conductors within the scope of this review. Therefore, no aging management activities are required for the period of extended operation.

Regarding the aging effect of increased electrical resistance, Switchyard Bus conductor connections are generally of the compression bolted category. No organic materials are involved. Components in the 230KV Switchyard are exposed to precipitation. Connection materials exposed to the service conditions of the CR-3 230KV Switchyard do not experience any appreciable aging effects, except for minor oxidation, which does not impact the ability of the switchyard bus to perform its intended function. CR-3 transmission conductor connection surfaces are coated with an anti-oxidant compound (a grease-type sealant) prior to tightening the connection to prevent the formation of oxides on the metal surface and to prevent moisture from entering the connection, thus reducing the chances of corrosion. Based on operating experience, this method of installation has been shown to provide a corrosion resistant low electrical resistance connection. Therefore, it is concluded that general corrosion resulting in the oxidation of switchyard connection surface metals is not an aging effect requiring management. The only bolted connections associated with the transmission conductors are for the connections to the switchyard bus and for the connections to the high voltage bushings on the Backup Engineered Safeguards Transformer (BEST). The aluminum bolting hardware used for the connections to the switchyard bus was selected to be compatible with the aluminum connector/conductor coefficient of thermal expansion. This ensures that the contact pressure of the bolt and washer combination used in the connector is maintained to the initial vendor specified torque value. CR-3 design incorporates the use of stainless steel "Belleville" washers on the bolted electrical connections to the main power transformers to compensate for temperature changes, maintain the proper torque, and prevent loosening of dissimilar metal connection hardware. This method of assembly is consistent with the good bolting practices recommended in EPRI Technical Report 1003471, "Bolted Joint Maintenance and Applications Guide," December 2002. Connection materials exposed to the service conditions of the CR-3 230KV Switchyard may experience minor oxidation but it is not significant enough to cause a loss of intended function.

3.6.2.2.4 Quality Assurance for Aging Management of Non-Safety Related Components

QA provisions applicable to License Renewal are discussed in Section B.1.3.

3.6.2.3 Time-Limited Aging Analysis

The Time-Limited Aging Analyses (TLAA) identified below are associated with Electrical and I&C System components.

1. Environmental Qualification Vendor Qualification Packages. Vendor Qualification Packages are aging analyses for components in the Environmental Qualification (EQ) Program. EQ of electrical components is a TLAA; refer to the discussion of Item 3.6.1-01 on Table 3.6.1 below.

2. Non-Environmental Qualification Electrical and I&C Components. Several non-EQ components that are in the scope of License Renewal have aging analyses that meet the definition of a TLAA. These consist of surge capacitors, control relays, overvoltage relays, and frequency relays for the Emergency Diesel Generators. These components are to be refurbished, or replaced, or have their qualification extended prior to reaching the end of their service lives. These activities are managed in accordance with the Corrective Action Program. Therefore, the effects of aging are being managed in accordance with 10 CFR 54.21(c)(1)(iii).

3. Non-Environmental Qualification Electrical and I&C Cables. Several non-EQ electrical cables that are in the scope of License Renewal have aging analyses that meet the definition of a TLAA. These consist of safety related and non-safety related electrical and I&C cables. These cables are to be replaced or have their qualification extended prior to reaching the end of their service lives. These activities are managed in accordance with the Corrective Action Program. Therefore, the effects of aging are being managed in accordance with 10 CFR 54.21(c)(1)(iii).

3.6.3 CONCLUSIONS

The Electrical and I&C System components/commodities having aging effects requiring management have been evaluated, and aging management programs have been selected to manage the aging effects. A description of the aging management programs is provided in Appendix B, along with a demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstration provided in Appendix B, the effects of aging will be adequately managed so that there is reasonable assurance that the intended functions of Electrical and I&C Systems components/commodities will be maintained consistent with the current licensing basis during the period of extended operation.

TABLE 3.6.1 SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTER VI OF NUREG-1801 FOR ELECTRICAL COMPONENTS

Item Number	Component/ Commodity	Aging Effect/Mechanism	Aging Management Program	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 electric components	Yes, TLAA	Further evaluation of EQ TLAAs is documented in Subsection 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 Not Subject to 10 CFR 50.49 EQ Requirements Used In Instrumentation Circuits	No	Consistent with NUREG-1801. This AMP applies to cable systems in the Nuclear Instrumentation and Radiation Monitoring Systems.
3.6.1-04	Conductor insulation for inaccessible medium voltage (2KV to 35KV) 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 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTER VI OF NUREG-1801 FOR ELECTRICAL COMPONENTS

Item Number	Component/ Commodity	Aging Effect/Mechanism	Aging Management Program	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.
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	Fatigue due to ohmic heating, thermal cycling, electrical transients, frequent manipulation, vibration, and chemical contamination are not applicable aging effects. This is discussed in plant-specific Note 603 of Table 3.6.2-1. Loss of continuity due to corrosion and oxidation will be managed by the Fuse Holder Program.
3.6.1-07	Metal enclosed bus – Bus/connections	Loosening of bolted connections due to thermal cycling and ohmic heating	Metal Enclosed Bus	No	Consistent with NUREG-1801.
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	Consistent with NUREG-1801.
3.6.1-09	Metal enclosed bus – Enclosure assemblies	Loss of material due to general corrosion	Structures Monitoring Program	No	Consistent with NUREG-1801. CR-3 manages the aging effect with the *Structures Monitoring Program*. The AMR for this item is performed in Tables 3.5.2-6, 3.5.2-14, and 3.5.2-19.

TABLE 3.6.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTER VI OF NUREG-1801 FOR ELECTRICAL COMPONENTS

Item Number	Component/ Commodity	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.6.1-10	Metal enclosed bus – Enclosure assemblies	Hardening and loss of strength due to elastomers degradation	Structures Monitoring Program	No	The Metal Enclosed Bus Program, XI.E4, is credited for the aging management of elastomer seals associated with the MEB Enclosure Assemblies. The MEB Program performs internal inspections of the enclosure assembly for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion which may indicate degradation of the elastomer seal.
3.6.1-11	High-voltage insulators	Degradation of insulation quality due to presence of any salt deposits and surface contamination; Loss of material caused by mechanical wear due to wind blowing on transmission conductors	Plant specific	Yes, plant specific	Consistent with NUREG-1801. The plant-specific High-Voltage Insulators in the 230KV Switchyard Program will be used to manage the applicable aging effects. This is discussed in Subsection 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 preload	Plant specific	Yes, plant specific	The aging effects specified in NUREG-1801 are negligible. As discussed in plant-specific Notes 606 and 607 of Table 3.6.2-1, no AMP is required. Further evaluation of this item is provided in Subsection 3.6.2.2.3.

TABLE 3.6.1 (continued) SUMMARY OF AGING MANAGEMENT EVALUATIONS IN CHAPTER VI OF NUREG-1801 FOR ELECTRICAL COMPONENTS

Item Number	Component/ Commodity	Aging Effect/Mechanism	Aging Management Program	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.
3.6.1-14	Fuse Holders (Not Part of a Larger Assembly) Insulation material	None	None	NA - No AEM or AMP	Consistent with NUREG-1801.

TABLE 3.6.2-1 ELECTRICAL AND I&C SYSTEMS – SUMMARY OF AGING MANAGEMENT EVALUATION – ELECTRICAL/I&C COMPONENTS/COMMODITIES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Electrical Equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Various	Various Polymeric and Metallic Materials	Adverse localized environment caused by heat, radiation, oxygen, moisture or voltage	Various degradation effects/ Various mechanisms	Environmental Qualification of Electric Equipment	VI.B-1 (L-05)	3.6.1-01	A
Cable Connections – Metallic Parts	E-1	Various Metals	Air - Indoor 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 Environmental Qualification Requirements	VI.A-1 (LP-12)	3.6.1-13	A
Non-EQ Insulated Cables and Connections	E-1	Various Organic Polymers (e.g., EPR, SR, EPDM, XLPE)	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure/ 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 Environmental Qualification Requirements	VI.A-2 (L-01)	3.6.1-02	A

TABLE 3.6.2-1 (continued) ELECTRICAL AND I&C SYSTEMS – SUMMARY OF AGING MANAGEMENT EVALUATION – ELECTRICAL/I&C COMPONENTS/COMMODITIES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Non-EQ Cables and Connections Used in Instrumentation Circuits Sensitive to a Reduction in Insulation Resistance (IR)	E-1	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 Environmental Qualification Requirements Used in Instrumentation Circuits	VI.A-3 (L-02)	3.6.1-03	A, 601
Medium-Voltage Power Cables	E-1	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 Environmental Qualification Requirements	VI.A-4 (L-03)	3.6.1-04	A
Electrical connector contacts exposed to borated water leakage	E-1	Various Metals	Air with borated water leakage	Corrosion of connector contact surfaces/ intrusion of borated water	Boric Acid Corrosion	VI.A-5 (L-04)	3.6.1-05	A

TABLE 3.6.2-1 (continued) ELECTRICAL AND I&C SYSTEMS – SUMMARY OF AGING MANAGEMENT EVALUATION – ELECTRICAL/I&C COMPONENTS/COMMODITIES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Fuse Holders (Not Part of a Larger Assembly); Insulation	E-2	Insulation Material-Melamine, Phenolic	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 (Thermal/ thermoxidative), of organics/ thermoplastics, radiation-induced oxidation, moisture intrusion and ohmic heating	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	VI.A-6 (LP-03)	3.6.1-02	A, 602
			Air - Indoor	None	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	VI.A-7 (LP-02)	3.6.1-14	A, 602
Fuse Holders (Not Part of a Larger Assembly); Metallic Clamp	E-1	Copper Alloy	Air - Indoor	Loss of Continuity/ Corrosion, Oxidation	Fuse Holder	VI.A-8 (LP-01)	3.6.1-06	I, 603

TABLE 3.6.2-1 (continued) ELECTRICAL AND I&C SYSTEMS – SUMMARY OF AGING MANAGEMENT EVALUATION – ELECTRICAL/I&C COMPONENTS/COMMODITIES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Non-EQ Electrical/ I&C Penetration Assemblies	E-1	XLPO, SR, Kapton, EPDM, CSPE, EPR, Kynar (PVDF)	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen	None	None			J, 604
Non-EQ Electrical/ I&C Penetration Assembly Pigtails	E-1	EPDM	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 Environmental Qualification Requirements			J, 605
Metal Enclosed Bus-Bus/ Connections	E-1	Aluminum/ Silver Plated Aluminum, Copper, Steel	Air - Indoor Air - Outdoor	Loosening of bolted connections/thermal cycling and ohmic heating	Metal Enclosed Bus	VI.A-11 (LP-04)	3.6.1-07	A

TABLE 3.6.2-1 (continued) ELECTRICAL AND I&C SYSTEMS – SUMMARY OF AGING MANAGEMENT EVALUATION – ELECTRICAL/I&C COMPONENTS/COMMODITIES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Metal Enclosed Bus-Enclosure assemblies	E-2	Elastomers	Air - Indoor Air - Outdoor	Hardening and loss of strength/ elastomer degradation	Metal Enclosed Bus	VI.A-12 (LP-10)	3.6.1-10	E
Metal Enclosed Bus-Insulation/ insulators	E-2	Fiberglass, Organic Polymers, Phenolic, Porcelain	Air - Indoor Air - Outdoor	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure/ thermal/ thermoxidative degradation of organics/ thermoplastics, radiation-induced oxidation, moisture/ debris intrusion, and ohmic heating	Metal Enclosed Bus	VI.A-14 (LP-05)	3.6.1-08	A
High-Voltage Insulators	E-2	Cement, Galvanized Metals, Porcelain	Air - Outdoor	Degradation of insulation quality/presence of any salt deposits or surface contamination	High-Voltage Insulators in the 230KV Switchyard	VI.A-9 (LP-07)	3.6.1-11	A
				Loss of material/ mechanical wear due to wind blowing on transmission conductors	High-Voltage Insulators in the 230KV Switchyard	VI.A-10 (LP-11)	3.6.1-11	A

TABLE 3.6.2-1 (continued) ELECTRICAL AND I&C SYSTEMS – SUMMARY OF AGING MANAGEMENT EVALUATION – ELECTRICAL/I&C COMPONENTS/COMMODITIES

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Switchyard Bus and Connections	E-1	Aluminum, Galvanized Steel, Stainless Steel,	Air - Outdoor	None	None	VI.A-15 (LP-09)	3.6.1-12	I, 606
Transmission Conductors and Connections	E-1	Aluminum, Steel	Air - Outdoor	None	None	VI.A-16 (LP-08)	3.6.1-12	I, 607

Notes for Table 3.6.2-1:

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 AMP is credited or NUREG-1801 identifies a plant-specific AMP.
F. Material not in NUREG-1801 for this component.
G. Environment not in NUREG-1801 for this component and material.
H. Aging effect not in NUREG-1801 for this component, material and environment combination.
I. Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.
J. Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific Notes:

601. The scope of this program applies to the non-EQ cable systems in the Nuclear Instrumentation and Radiation Monitoring systems that are sensitive to a reduction in insulation resistance.

602. Evaluation has shown that the materials used for the fuse holder base (or block) experience no applicable aging effects in their service environment. Therefore, no aging management program is warranted for this item. However, since fuse holders represent another type of electrical connection similar to terminal blocks, fuse holders are included in the Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Program .

603. CR-3 fuse holders subject to AMR are used in control valve and/or intermittent instrumentation and control (I&C) applications. The only fuses that could potentially be exposed to thermal cycling and ohmic heating are those that carry significant current in power supply applications. I&C circuits characteristically operate at such low currents that no appreciable thermal cycling or ohmic heating occurs. Since thermal cycling and ohmic heating apply to power supply applications, they are not considered applicable aging mechanisms for CR-3 fuse holders. CR-3 electrical design ensures that stresses due to forces associated with electrical faults and transients are mitigated by the fast action of circuit protective devices at high currents. Mechanical stress due to electrical faults is not considered a credible aging mechanism since such faults are infrequent and random in nature. CR-3 fuses are not routinely pulled and/or manipulated to facilitate plant testing. Therefore, frequent manipulation is not considered an applicable aging mechanism. Vibration is induced in fuse holders by the operation of external equipment, such as compressors, fans, and pumps. Plant walk-down has verified that there are no direct sources of vibration for the fuse holder panels, and the panels are mounted separately to their own unistrut support structure on a concrete wall or column. Therefore, vibration is not considered an applicable aging mechanism. Plant walk-down has verified that there are no potential sources of chemical contamination in the area and that the fuse holders are totally enclosed in a protective junction box which would provide protection even if chemical contamination were possible. Therefore, based on their installed location and design configuration, chemical contamination is not considered an applicable aging mechanism. Plant walk-down has verified that corrosion and oxidation are credible aging mechanisms for fuse holders located within the Auxiliary Building due to moisture. The moisture required to produce corrosion and oxidation is not present in other non-condensing areas of the plant. The Fuse Holder Aging Management Program will confirm the absence of corrosion and oxidation resulting from moisture on the metallic clamp. The scope of this program applies to fuse holders located in stand-alone junction boxes within the Auxiliary Building.

604. Evaluation has shown that the insulation materials for this commodity group are aptly suited for their service conditions and acceptable for the period of extended operation.

605. The Electrical Cables and Connections Not Subject To 10 CFR 50.49 EQ Requirements Program is applicable to non-EQ Namco conduit seal assembly pigtails.

606. Section 3.6 of NUREG-1800 indicates that further evaluation Switchyard Bus and Connections should be provided to address 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 Preload. Refer to the further evaluation of these aging effects in Subsection 3.6.2.2.3.

607. Section 3.6 of NUREG-1800 indicates that further evaluation Transmission Conductors and Connections should be provided to address 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 Preload. Refer to the further evaluation of these aging effects in Subsection 3.6.2.2.3.

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4.0 TIME-LIMITED AGING ANALYSES

Two areas of technical review are required to support an application for a renewed operating license. The first area of technical review is the Integrated Plant Assessment, described in Chapters 2.0 and 3.0. The second area of technical review is the identification and evaluation of plant-specific time-limited aging analyses and exemptions, provided in this chapter. The evaluations included in this chapter meet the requirements contained in10 CFR 54.21(c).

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) be provided as part of the application for a renewed license. 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 10 CFR 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 10 CFR 54.4(b); and
6. Are contained or incorporated by reference in the current licensing basis.

4.1.1 TIME-LIMITED AGING ANALYSES IDENTIFICATION PROCESS

The process used to identify the CR-3-specific TLAAs is consistent with the guidance provided in NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule." Calculations and evaluations that could potentially meet the six criteria of 10 CFR 54.3 were identified by searching CLB and other documents including:

- Technical Specifications,
- The CR-3 FSAR,
- Docketed licensing correspondence,
- Design Basis Documents,
- CR-3 calculations, and
- Applicable AREVA analyses and reports.

Industry- and NRC-prepared documents that list generic TLAAs also were reviewed to provide additional assurance of the completeness of the plant-specific list. These documents included NEI 95-10; NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Rev. 1, U. S. Nuclear Regulatory Commission, September 2005, NUREG-1801, "Generic Aging Lessons Learned (GALL) Report;" Rev. 1, U. S. Nuclear Regulatory Commission, September 2005, and License Renewal Applications submitted by other PWR licensees.

The potential TLAAs were evaluated by screening against the six criteria in the definition of TLAA in 10 CFR 54.3. The analyses and evaluations that meet all six criteria of 10 CFR 54.3 are the TLAAs for CR-3 and are listed in Table 4.1-1.

Table 4.1-2 summarizes the results of reviewing the generic list of TLAAs provided in Tables 4.1-2 and 4.1-3 of NUREG-1800.

4.1.2 EVALUATION OF TIME-LIMITED AGING ANALYSES

As required by10 CFR 54.21(c)(1), an evaluation of CR-3-specific TLAAs must be performed to 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 functions(s) will be adequately managed for the period of extended operation.

The results of these evaluations are summarized in Table 4.1-1 and discussed in Sections 4.2 through 4.7.

4.1.3 IDENTIFICATION OF EXEMPTIONS

10 CFR 54.21(c) also requires that the application for a renewed license include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on TLAAs as defined in 10 CFR 54.3. This was performed by evaluating the basis for each active exemption, granted pursuant to 10 CFR 50.12, to determine whether the exemption was based on a TLAA.

As a result of this review, one exemption was identified as meeting the definition of a TLAA. This is a partial exemption from the provisions to 10 CFR 50, Appendix A, General Design Criterion 4, to permit revision of the design of reactor coolant pump supports. Specifically the exemption permitted replacing 32 large bore piping snubbers with four smaller snubbers and four struts. The analysis used leak-before-break (LBB) technology that relies on fracture mechanics to demonstrate the capability to detect leakage well before any cracks in the pipe wall could become unstable and grow to failure. The fracture mechanics analysis is contained in report BAW-1847, "The B&W Owner's Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," Revision 1. This evaluation is a TLAA for CR-3 and is listed on Table 4.1-1.

TABLE 4.1-1 TIME-LIMITED AGING ANALYSES

NUREG-1800 TLAA Category	Analysis	10 CFR 54.21(c)(1) Paragraph	Section
1.	**Reactor Vessel Neutron Embrittlement**	----------	4.2
	Neutron Fluence	(ii)	4.2.1
	Upper Shelf Energy Analysis	(ii)	4.2.2
	Pressurized Thermal Shock Analysis	(ii)	4.2.3
	Operating Pressure-Temperature Limits Analysis	(iii)	4.2.4
	Low Temperature Overpressure Limits Analysis	(iii)	4.2.5
	Reactor Vessel Underclad Cracking	(ii)	4.2.6
	Reduction in Fracture Toughness of Reactor Vessel Internals	(iii)	4.2.7
2.	**Metal Fatigue**	----------	4.3
	Fatigue Analyses (NSSS Components)	----------	4.3.1
	Reactor Vessel	(iii)	4.3.1.1
	Reactor Vessel Internals	(ii) and (iii)	4.3.1.2
	Control Rod Drive Mechanism	(i)	4.3.1.3
	Reactor Coolant Pumps	(i) and (ii)	4.3.1.4
	Steam Generators	(iii)	4.3.1.5
	Pressurizer	(iii)	4.3.1.6
	Reactor Coolant Pressure Boundary Piping (USAS B31.7)	(iii)	4.3.1.7
	Implicit Fatigue Analysis (B31.1 Piping)	----------	4.3.2
	USAS B31.1.0 Piping - RCPB Class 1	(i) and (iii)	4.3.2.1
	USAS B31.1.0 Piping - Non-Class 1	(i) and (ii)	4.3.2.2
	Environmentally-Assisted Fatigue Analysis	(iii)	4.3.3
	RCS Loop Piping Leak-Before-Break Analysis	(i)	4.3.4
3.	**Environmental Qualification of Electrical Equipment**	----------	4.4
	10 CFR 50.49 Thermal, Radiation, and Cyclical Aging Analyses	(iii)	4.4.1
4.	**Concrete Containment Tendon Prestress**	----------	4.5
	Tendon Stress Relaxation Analysis	(ii)	4.5.1
5.	**Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis**	----------	4.6
	Fuel Transfer Tube Expansion Bellows Cycles	(i)	4.6.1
6.	**Other Plant-Specific Time-Limited Aging Analyses**	----------	4.7
	Analysis of Bedrock Dissolution from Groundwater	(ii)	4.7.1

TABLE 4.1-2 REVIEW OF GENERIC TLAAs LISTED ON TABLES 4.1-2 AND 4.1-3 OF NUREG-1800

NUREG-1800 Generic TLAA Examples	Applicability to CR-3	Section
NUREG-1800, Table 4.1-2		
Reactor vessel neutron embrittlement	Yes	4.2
Concrete containment tendon prestress	Yes	4.5
Metal fatigue	Yes	4.3
Environmental qualification of electrical equipment	Yes	4.4
Metal corrosion allowance	No - No potential TLAA identified.	-
Inservice flaw growth analyses that demonstrate structure stability for 40 years	No - No potential TLAA identified.	-
Inservice local metal containment corrosion analyses	No - Did not meet TLAA criteria.	-
High-energy line-break postulation based on fatigue cumulative usage factor	No - Did not meet TLAA criteria.	-
NUREG-1800, Table 4.1-3		
Intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic SS 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 pumps	Yes	4.3.2.2
Fatigue analysis for the reactor coolant pump flywheel	No - Did not meet TLAA criteria.	-
Fatigue analysis of polar crane	No - Did not meet TLAA criteria.	-
Flow-induced vibration endurance limit for the reactor vessel internals	Yes	4.3.1.2
Transient cycle count assumptions for the reactor vessel internals	Yes	4.3.1.2
Ductility reduction of fracture toughness for the reactor vessel internals	Yes	4.2.7
Leak before break	Yes	4.3.4
Fatigue analysis for the containment liner plate	No - Did not meet TLAA criteria.	-
Containment penetration pressurization cycles	No - No potential TLAA identified.	-
Reactor vessel circumferential weld inspection relief (BWR)	Not applicable.	-

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

Neutron embrittlement is the term used to describe changes in mechanical properties of reactor vessel (RV) materials that result from exposure to fast neutron flux (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 defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence.

Fracture toughness of ferritic materials is not only dependent upon fluence, but is also dependent upon temperature. The reference temperature for nil-ductility transition, RT_{NDT}, is an indicator of the transition temperature range above which the material behaves in a ductile manner, and below which it behaves in a brittle manner. As fluence increases, the nil-ductility reference temperature increases. This means higher temperatures are required for the material to continue to behave in a ductile manner. This shift in reference temperature is the ΔRT_{NDT} plus a margin term which is added to account for uncertainties associated with the limited amount of data available for making the projections. Determining the projected reduction in fracture toughness as a function of fluence affects several analyses used to support operation of CR-3:

- Reactor Pressure Vessel (RPV) Fluence
- RPV Material Upper-Shelf Energy (USE)
- RPV Pressurized Thermal Shock (PTS)
- RPV Operating Pressure-Temperature (P-T) Limits
- RPV Low-Temperature Overpressure Protection (LTOP) Setpoints
- Reactor Vessel Underclad Cracking
- Reduction in Fracture Toughness of Reactor Vessel Internals

Since extending the operating period from 40 years to 60 years will further increase the fluence levels, the 60-year fluence value must be determined and used to determine its impact upon the analyses used to support operation. The approach taken was that, if the existing analyses could not be demonstrated to remain valid, new analyses were prepared. If a revised analysis was not feasible, the aging effect identified within the time-limited aging analysis (TLAA) will be managed during the period of extended operation.

4.2.1 NEUTRON FLUENCE

Summary Description

Loss of fracture toughness is an aging effect caused by the neutron embrittlement aging mechanism that results from prolonged exposure to neutron irradiation. This process results in increased tensile strength and hardness of the material with reduced toughness. The rate of neutron exposure is defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence. As neutron embrittlement progresses, the toughness/temperature curve shifts down (lower fracture toughness), and the curve shifts to the right (brittle/ductile transition temperature increases).. NRC regulations require projections to be made showing the degree of shift expected using end-of-life (EOL) fluence values. A minimum upper shelf energy (USE) value limits the degree of downward shift, and a pressurized thermal shock (PTS) screening criteria (maximum reference temperature) limits the shifting of the ductile/brittle transition temperature to the right. If a projection indicates a shift exceeding these limits could occur in the future, changes must be implemented to either prevent this from occurring, such as improved operational practices, fluence reduction strategies, or additional evaluations must demonstrate that equivalent margins of safety exist even with the projected shift.

End-of-life fluence is based on a projected value of effective full power years (EFPY) over the licensed life of the plant. For the current term of operation, end-of-life for CR-3 is 40 years and reactor vessel embrittlement calculations for pressurized thermal shock and upper shelf energy are based on fluence projections at 32 EFPY. The plant began operation in December 1976, and the plant lifetime capacity factor through 2005 is 68.2%. Assuming a plant capacity factor of 98.5% beyond 2005, CR-3 will accrue approximately 50.3 EFPY by December 2036. Therefore, a 54 EFPY fluence estimate used for calculating reactor vessel embrittlement for 60 years of operation is bounding for the period of extended operation.

Analysis

AREVA NP (previously Framatome) developed a fluence analysis methodology that can be used to accurately predict the fast neutron fluence in the reactor vessel using surveillance capsule dosimetry and/or cavity dosimetry to verify the fluence predictions. This methodology was developed through a full-scale benchmark experiment that was performed at the Davis-Besse Unit 1 reactor. The benchmark experiment demonstrated that the AREVA NP methodology was unbiased and was accurate well within the NRC suggested standard deviation of 20%. The AREVA NP fluence analysis methodology is compliant with NRC Regulatory Guide (RG) 1.190, as described in topical report BAW-2241NP-A, Revision 1, "Fluence and Uncertainty Methodologies," December 1999.

The NRC reviewed the AREVA NP methodology and concluded that the proposed methodology is acceptable for determining the pressure vessel fluence of B&W designed reactors. The NRC determined that the AREVA NP methodology could be referenced in B&W designed reactor licensing actions with three limitations. The applicability of those limitations to CR-3 license renewal are discussed below:

1. The dosimetry calculation-to-measurement database includes an extensive set of PWR core/internals/vessel configurations. However, the dosimetry set is not complete and certain designs are not included in the data-base. CR-3 is a B&W-designed reactor, and all applicable dosimetry is included in the database reported in BAW-2241NP-A, Revision 1.
2. Should there be changes in the input cross section of this methodology the licensee will evaluate the changes for their impact and if necessary will modify the methodology accordingly. (There have been no changes to the cross section methodology.)
3. The licensee will provide the staff with a record of future modifications of the methodology. (Note that there have been no changes to the methodology.)

The AREVA NP methodology was used to calculate the neutron fluence exposure to the CR-3 reactor vessel. The fast neutron fluence (neutron energy (E) > 1.0 MeV) at the reactor vessel upper and lower plates, as well as specific welds, was calculated in accordance with the requirements of RG 1.190.

The 54 EFPY fluence values include ex-vessel cavity dosimetry data from Cycles 11 and 12 and plant operation through Cycle 14. To account for a measurement uncertainty recapture, the Cycle 14 fluxes were used for Cycle 15 and increased by a factor of 1.02 for Cycles 16 and 17. The Cycle 16 and 17 flux was increased by a factor of 1.25 for Cycle 18 through the period of extended operation. The 54 EFPY fluence projections at the reactor vessel inside wetted surface are presented in Table 4.2-1.

Reactor Vessel Beltline

Reactor pressure vessel boundary components outside the beltline region have been evaluated to determine whether additional materials should be considered "beltline" material for the period of extended operation. The beltline, as defined by 10 CFR 50.61(a)(3), is the region of the reactor pressure vessel that directly surrounds the effective height of the active core and adjacent regions of the reactor pressure vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection for the most limiting material with regard to radiation damage. The threshold fluence for potential beltline material is 1.0E+17 n/cm^2, E > 1.0 MeV. The beltline materials for CR-3 for 60 years (i.e., 54 EFPY) include items 2 through 12 in Table 4.2-1. Items 1 and 13 are not considered beltline material since the fluence is less than the threshold fluence specified in 10 CFR 50, Appendix H.

Reactor Vessel Surveillance Program

The limiting beltline circumferential weld based on fluence and RT_{PTS}, as discussed in Subsection 4.2.3, for CR-3 at 54 EFPY is WF-70, heat number 72105. As indicated on Table 4.2-1, the fluence at 54 EFPY for weld WF-70, is 1.56E+19 n/cm^2. In the Master Integrated Reactor Vessel Material Surveillance Program (MIRVP), two capsules with weld wire heat number 72105 have been irradiated to fluence values equal to or greater than 1.56E+19 n/cm^2 and tested. Therefore, the MIRVP program covers the fluence at 54 EFPY for CR-3 weld WF-70, and no additional surveillance material or testing is required for 60 years of operation.

The limiting beltline axial weld based on fluence and RT_{PTS} for CR-3 at 54 EFPY is WF-8, heat number 8T1762. This heat of material is not in the MIRVP, and there is no need to add this material since the CR-3 Linde 80 beltline weld materials, including WF-8, are adequately represented by the eight heats of material in the MIRVP program.

The limiting shell plate material for CR-3 is C4344-1, which was included in CR-3-specific capsules, and all specimens have been removed and tested. As indicated on Table 4.2-1, the 54 EFPY fluence at plate C4344-1 is predicted to be 1.60E+19 n/cm^2. Capsule CR3-F, which contained C4344-1 material, received a fluence of 1.08E+19 n/cm^2 and was removed and tested. The MIRVP has determined that no further testing is required for material C4344-1 since the plate material is not the limiting material for the CR-3 vessel and the MIRVP meets the requirements of 10 CFR 50, Appendix H. For further information on the CR-3 Reactor Vessel Surveillance Program see Appendix B, Subsection B.2.17.

Therefore, the neutron fluence has been projected to the end of the period of extended operation using a methodology previously approved by the NRC. These fluence projections will be used for evaluating fluence-based TLAAs for CR-3 License Renewal.

Disposition: 10 CFR 54.21(c)(1)(ii) – The neutron fluence analyses have been projected to the end of the period of extended operation.

4.2.2 UPPER SHELF ENERGY ANALYSIS

Summary Description

Fracture toughness is a property which describes the ability of a material to resist fracture. In reactor vessel ferritic materials, toughness increases as a function of temperature. At low temperatures, material toughness is relatively low and changes very little with temperature, and the material is said to exhibit brittle behavior. As the temperature increases, a transition region is eventually reached in which toughness increases rapidly with an increase in temperature. At temperatures above the transition region, the material toughness is relatively high and changes very little with temperature, and the material is said to exhibit ductile behavior. The Charpy impact test is used to estimate fracture toughness by measuring the amount of energy absorbed during the fracture of a notched test specimen. Upper-shelf energy (USE) is a measure of the average energy absorbed by Charpy impact specimens tested at a temperature above the upper end of the transition region. 10 CFR 50, Appendix G, states that reactor vessel beltline materials must have Charpy upper-shelf energy (C_VUSE) in the transverse direction for base metal and along the weld for weld metal of no less than 75 ft-lb in the unirradiated condition, and must maintain C_VUSE of no less than 50 ft-lb throughout the licensed life of the vessel, unless it can be demonstrated that lower values of energy will provide margins of safety against fracture equivalent to those required by the ASME Code, Section XI, Appendix G.

NRC RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides two methods for determining C_VUSE. Position 1.2 applies for material that does not have surveillance data available, and Position 2.2 applies for material that does have surveillance data. For Position 1.2, the percent drop in C_VUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of RG 1.99, Revision 2. This percentage drop is applied to the initial C_VUSE to obtain the adjusted C_VUSE. For Position 2.2, the percent drop in C_VUSE is determined by plotting the available data on Figure 2, and fitting the data with a line drawn parallel to the existing lines that represent upper bounds of all the plotted points.

Analysis

<u>USE for Beltline Plates and Forgings</u>

Initial upper shelf energy and copper content for beltline plates and forgings are listed in Table 4.2-2. Fluence at the 1/4T location is based on attenuation of the inside wetted surface fluences presented in Table 4.2-1 using RG 1.99, Revision 2, Equation (3) with a base metal thickness of 8.44 in. and cladding thickness of 0.125 in. Upper shelf energies for these beltline plates and forgings at 54 EFPY, using Position 1.2, are reported in Table 4.2-2 and are all above 50 ft-lb, which is acceptable. Percentage reduction in USE is obtained from Figure 2 of Regulatory Guide 1.99, Revision 2. Position 2.2 could be applied to plate C4344-1, but Position 1.2 is bounding.

<u>USE for Beltline Welds</u>

As is the case for the current term of operation, the C_VUSE values for all beltline welds are below 50 ft-lb, requiring an equivalent margin analysis (EMA) for the period of extended operation. The methodology used to evaluate CR-3 beltline welds at 60 years is consistent with the EMA methods reported in BAW-2192PA, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A & B Service Loads," April 1994; BAW-2178PA, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads," April 1994; and BAW-2275A, "Low Upper Shelf Toughness Fracture Mechanics Analysis of B&W

Designed Reactor Vessels for 48 EFPY," August 1999. BAW-2275A comprises Appendix B of BAW-2251A, "Demonstration of Management of Aging Effects for the Reactor Vessel," January 2002.

An updated EMA was performed on CR-3 limiting beltline welds WF-70, WF-8, and WF-18 to consider the effect of increased fluence on the J-integral, which is a function of fluence. The applied J-integral, which is due to loading, is not a function of fluence and remains unchanged from earlier analyses. The results of the updated analysis are provided in Table 4.2-3 and Table 4.2-4.

The results in Table 4.2-3 show that the first acceptance criterion of $J_{0.1} / J_1 > 1.0$ from ASME Section XI, Article K-2200(a)(1) for Level A and B service loading is met. The results in Table 4.2-4 show that the acceptance criterion of $J_{0.1} / J_1 > 1.0$ for Level C and D service loading is also met.

Therefore, the limiting CR-3 welds provide margins of safety equivalent to those of Appendix G of the Section XI of the ASME Code and have adequate upper-shelf toughness, and satisfy the requirements of Appendix G to 10 CFR 50 for operation through 54 EFPY.

<u>USE Summary for Beltline Plates, Forgings, and Welds</u>

An evaluation of the USE for the CR-3 RPV beltline materials was performed for the 54-EFPY License Renewal period using the guidelines in RG 1.99, Revision 2. The evaluations for the decreases in USE of the RPV were performed at the 1/4T wall location of each beltline material using the respective copper contents and Figure 2 of RG 1.99, Revision 2. The results of the evaluations are provided in Table 4.2-2. All shell plate and forgings remain above the 50 ft-lb limit. However, all the CR-3 RPV beltline welds have projected USE less than 50 ft-lb, and equivalent margins analyses have been performed to show the acceptability of these welds to the end of the 60-year period of extended operation. The results of these evaluations are shown in Table 4.2-3 and Table 4.2-4.

Disposition: 10 CFR 54.21(c)(1)(ii) – The analysis of Upper Shelf Energy has been projected to the end of the period of extended operation.

4.2.3 PRESSURIZED THERMAL SHOCK ANALYSIS

Summary Description

10 CFR 50.61 defines screening criteria for embrittlement of reactor pressure vessel materials in pressurized-water reactors, as well as actions that are required if these screening criteria are exceeded. The screening criteria limit the degree that a vessel material may increase in its reference temperature for pressurized thermal shock - RT_{PTS}, following neutron irradiation of the reactor pressure vessel. For circumferential

welds, the pressurized thermal shock (PTS) screening criterion is 300°F maximum. For plates, forgings, and axial weld materials, the screening criterion is 270°F maximum. The projected EOL RT_{PTS} values must be shown to remain below the applicable screening temperature.

Analysis

A PTS evaluation for the CR-3 RV beltline materials was performed in accordance with 10 CFR 50.61. The PTS reference temperature, RT_{PTS}, values are calculated by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT} and the margin term to cover the uncertainties in the values of initial RT_{NDT} copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective RV beltline materials copper and nickel contents and the neutron fluence applicable to the CR-3 RV for License Renewal at 54 EFPY.

The evaluations for the CR-3 RT_{PTS} values were performed for each CR-3 reactor vessel beltline material with chemistry factors determined from Tables 1 and 2 in 10 CFR 50.61. In addition, the chemistry factors for the upper shell plate, heat number C4344-1 was recalculated using the available CR-3 surveillance data in accordance with RG 1.99, Revision 2.

The CR-3 RT_{PTS} values for the reactor vessel beltline materials for the period of extended operation are found in Table 4.2-5, calculated using 54 EFPY inside wetted surface fluence projections. The limiting longitudinal welds are WF-8 and WF-18 with an RT_{PTS} of 231.3°F, which is below the screening criterion of 270°F. The limiting circumferential weld is WF-70 with an RT_{PTS} of 253.8°F, which is below the screening criterion of 300°F.

Disposition: **10 CFR 54.21(c)(1)(ii) – The analyses for the shift in PTS reference temperature have been projected to the end of the period of extended operation.**

4.2.4 OPERATING PRESSURE-TEMPERATURE LIMITS ANALYSIS

Summary Description

The adjusted reference temperature (ART) is the value of Initial RT_{NDT} + ΔRT_{NDT} + margins for uncertainties at a specific reactor vessel location. Neutron embrittlement increases the ART. Thus, the minimum temperature at which a reactor vessel is allowed to be pressurized increases over the licensed period. The ART of the limiting beltline material is used to adjust the beltline pressure-temperature (P-T) limits to account for radiation effects. 10 CFR Part 50, Appendix G requires reactor vessel thermal limit analyses to determine operating P-T limits for boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences. Operating limits for pressure and temperature are required for three categories of operation: 1) hydrostatic pressure tests and leak tests, 2) non-nuclear heat-up/cooldown and low level physics tests, and 3) core critical operation.

10 CFR 50, Appendix G, provides P-T limits and minimum temperature requirements for the reactor vessel. The P-T limits and minimum temperature requirements are defined by operating condition, vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. The P-T limits must be at least as restrictive as limits obtained by following the methods of analysis and margins of safety of Appendix G of Section XI of the ASME Code.

The P-T limits are established by calculations that utilize the materials and fluence data obtained through the reactor surveillance capsule program. Normally, the P-T limits are calculated for several years into the future and remain valid for an established period of time.

Analysis

The ART values for the CR-3 reactor vessel beltline region materials are calculated in accordance with RG 1.99, Revision 2, by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT}, and a margin term to cover the uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 54 EFPY. The evaluations for the CR-3 ART were performed at the 1/4T and 3/4T wall location of each beltline material with chemistry factors determined from Tables 1 and 2 in RG 1.99, Revision 2. In addition, the chemistry factors for the Upper Shell Plate, heat number C4344-1, were recalculated using the available CR-3 surveillance data.

In this manner, ART results for the CR-3 reactor vessel beltline region materials applicable to 54 EFPY were determined, and are presented in Table 4.2-6. Based on the results in Table 4.2-6, the controlling beltline material for the CR-3 reactor vessel with respect to P-T limits are the Upper Shell Circumferential Weld (Inside 40%) SA-1769 (at 1/4T) and the Upper/Lower Shell Circumferential Weld WF-70 (at 3/4T).

The pressure-temperature operating limits were developed in accordance with the requirements of 10 CFR Part 50, Appendix G, utilizing the analytical methods and flaw acceptance criteria of topical report BAW-10046A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," Revision 2, June 1996, and ASME Code Section XI, Appendix G, 2001 edition through 2003 Addenda. CR-3 has implemented changes in the P-T limit curves throughout the current period of operation. ASME Code Cases N-588 and N-640 are incorporated in ASME Section XI, Appendix G, 2001 edition through 2003 Addenda. With the incorporation of the new methodology from ASME Code Section XI, Appendix G, 2001 edition through 2003 Addenda, and the improved replacement RV head, the 54 EFPY uncorrected P-T limits provide more operating room than the 32 EFPY uncorrected P-T curves.

CR-3 Technical Specifications refer to the P-T limit curves in the Pressure-Temperature Limits Report (PTLR), and those P-T limit curves are valid through 32 EFPY. Although new P-T limits for CR-3 for the period of extended operation have been calculated, it is not intended to implement these new curves at this time. CR-3 will continue to implement changes in the P-T limit curves in the PTLR, as required by Appendix G of 10 CFR part 50, for the remainder of the current period of operation and for the extended period of operation. The P-T limits for the remainder of the current period of operation and for the extended period of operation will be managed by using approved fluence calculations when there are changes in power or core design, and with surveillance capsule results. Updating the P-T limit curves using the described approach will assure that the operational limits remain valid for the remainder of the current period of operation and for the extended period of operation. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

Disposition: **10 CFR 54.21(c)(1)(iii) – The P-T limits analysis will be managed through the end of the period of extended operation.**

4.2.5 LOW-TEMPERATURE OVERPRESSURE LIMITS ANALYSIS

Summary Description

ASME Section XI, Appendix G, establishes procedures and limits for RCS pressure and temperature primarily for low temperature conditions to provide protection against non-ductile failure of the RV. The Low Temperature Overpressure Protection System (LTOPS) assures that these limits are not exceeded when it is enabled at low temperatures.

Analysis

The LTOP setpoints for CR-3 have been reanalyzed to support operation to the end of the period of extended operation. The LTOP setpoint analysis included the fluence projections from Subsection 4.2.1 and available surveillance data. The revised LTOP setpoints will be implemented when the revised P-T limit curves are implemented, prior to exceeding 32 EFPY. Maintaining the LTOP setpoints in accordance with Appendix G of 10 CFR 50 and 10 CFR 50.60 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

Disposition: **10 CFR 54.21(c)(1)(iii) – The LTOP setpoints will be managed through the end of the period of extended operation.**

4.2.6 REACTOR VESSEL UNDERCLAD CRACKING

Summary Description

Underclad cracking (UCC) refers to intergranular separations in the heat affected zones of low alloy base metal under austenitic stainless steel cladding. B&W conducted an intensive investigation of UCC in the 1970s, consisting of testing and analysis. Results of the investigation showed the subject flaws are present only in A-508, Class 2, forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc process such as the six-wire, strip, and the two-wire series arc. The investigations also noted that no anomalies were observed in SA-533 Grade B, Class 1 plate materials clad by any of the high-heat-input processes.

The maximum discontinuity depth observed throughout the industry (0.156 in.) was used in the fracture mechanics analysis summarized in BAW-10013-A, "Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones under Austenitic Stainless Steel Weld Cladding," October, 1972. The results of the fracture mechanics analysis demonstrated that the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted growth due to design fatigue cycles.

Analysis

The fracture mechanics analysis for underclad cracking was updated in BAW-2274A, "Fracture Mechanics Analysis of Postulated Underclad Cracks in B&W Designed Reactor Vessels for the Period of Extended Operation," August 1999, to include the period of extended operation. The revised analysis concluded that postulated underclad cracking in the RVI meets the acceptance criteria of the ASME Code, Section XI, IWB-3612. The maximum crack growth and applied stress intensity factor for normal and upset conditions occur in the nozzle belt region. The fracture toughness margin for normal and upset conditions was determined to be 3.63, which is greater than the required toughness margin of 3.16. The maximum applied stress intensity for the emergency and faulted condition occurs in the closure head to head flange regions. The fracture toughness margin for emergency and faulted condition was 2.42, which is greater than the required toughness margin of 1.41.

The revised analysis was based on fracture toughness properties associated with 60-year fluences and was intended to bound the B&W fleet. While CR-3 is not specifically listed as a participant in BAW-2274A, the generic evaluation used bounding loads from the entire fleet of B&W 177 FA lowered loop operating plants. The loads used in the analysis are bounding for CR-3, provided that the material properties of applicable CR-3 vessel are bounded by those presented in BAW-2274A. Three vessel regions were evaluated: (1) nozzle belt, (2) closure flange, and (3) beltline.

Nozzle Belt

The ART at the inside surface of CR-3 Lower Nozzle Belt Forging AZJ 94 is 3.0°F higher than the ART evaluated for the previously limiting forging. Therefore, the CR-3 nozzle belt forging is not bounded and was re-analyzed for 54 EFPY. The results show that the postulated 0.353 in.-deep flaw on the inside surface of the CR-3 Lower Nozzle Belt Forging satisfies the IWB-3612 acceptance criteria for fracture toughness margin. Considering 54 EFPY of fatigue crack growth, the final flaw size is 0.487 in., and the fracture toughness margin of 3.49 for Level A and B Service Loadings is greater than the required value of 3.16. The available fracture toughness margin for Level C and D Service Loadings is 2.50 which exceeds the required value of 1.41. The results demonstrate that a postulated underclad crack in the CR-3 Lower Nozzle Belt Forging would satisfy the flaw acceptance criteria of the ASME Code for 54 EFPY of operation over a period of 60 years.

Closure Flange

Evaluation of the closure flange in BAW-2274A identified limiting closure flange material based on an inside surface fluence of 7.78E+16 n/cm^2. For CR-3, the fluence at 54 EFPY at the closure flange is 4.38E+13 n/cm^2 (Refer to Item 1 of Table 4.2-1) and thus remains bounded.

Beltline (Upper and Lower Shells)

CR-3 beltline upper and lower shell plates are fabricated from SA-533 Grade B, Class 1 and are not susceptible to underclad cracking. Since CR-3 does not have A-508, Class 2 forgings in the upper and lower shell region, the increase in ART due to increased fluence at 54 EFPY is not relevant for the evaluation of underclad cracking.

Disposition: 10 CFR 54.21(c)(1)(ii) – The underclad cracking analysis has been projected through the period of extended operation.

4.2.7 REDUCTION IN FRACTURE TOUGHNESS OF REACTOR VESSEL INTERNALS

Summary Description

Reduction of fracture toughness of 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 both the irradiation temperature and neutron fluence. The reactor vessel internals components most susceptible to reduction in fracture toughness are those nearest to the reactor core.

The effect of irradiation on the mechanical properties and deformation limits for the reactor vessel internals was evaluated for the current term of operation in Appendix E of topical report BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," June, 1970. The 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. This analysis is a TLAA for the current term of operation.

Analysis

In accordance with the guidance of NUREG-1801, Revision 1, regarding the aging management of reactor vessel internals components, CR-3 will:

1. Participate in the industry programs for investigating and managing aging effects on reactor internals,

2. Evaluate and implement the results of the industry programs as applicable to the reactor internals, and

3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor vessel internals to the NRC for review and approval.

This CR-3 commitment is documented in the FSAR supplement.

Disposition: **10 CFR 54.21(c)(1)(iii) – Reduction in fracture toughness of reactor vessel internals will be managed, consistent with the commitment to participate in industry programs related to the reactor vessel internals, through the end of the period of extended operation.**

TABLE 4.2-1 PROJECTED 60-YEAR (54 EFPY) FLUENCE VALUES

Reactor Vessel Location (At inside wetted surface)	Material ID	Fluence (n/cm^2)
Plates & Forgings		
1. Nozzle Belt Closure Flange (Note 1)	Not applicable	4.38E+13
2. Nozzle Belt Forging - Lower	AZJ 94	1.48E+19
3. Upper Shell Plate	C4344-1	1.60E+19
4. Upper Shell Plate	C4344-2	1.60E+19
5. Lower Shell Plate	C4347-1	1.62E+19
6. Lower Shell Plate	C4347-2	1.62E+19
Welds		
7. Upper Shell Circumferential (Circ.) Weld (Inside 40%)	SA-1769	1.48E+19
8. Upper Shell Circ. Weld (Outside 60%)	WF-169-1	(Note 2)
9. Upper Shell Axial Weld	WF-8	1.44E+19
10. Upper Shell Axial Weld	WF-18	1.44E+19
11. Upper Shell to Lower Shell Circ. Weld	WF-70	1.56E+19
12. Lower Shell Axial Welds	SA-1580	1.35E+19
13. Lower Nozzle Belt Forging to Outlet Nozzle Forging Weld (Note 1)	WF-70	6.67E+16

Notes:

1. Items 1 and 13 are not considered beltline material since the fluence is less than the threshold fluence specified in 10 CFR 50, Appendix H.
2. The Upper Shell Circumferential Weld (Outside 60%) does not contact the inside surface of the reactor vessel; therefore, inside wetted surface fluence is not applicable.

TABLE 4.2-2 PROJECTED 54 EFPY CHARPY V-NOTCH UPPER SHELF ENERGY (C_VUSE)

Material Description				Cu wt%	Initial C_VUSE ft-lbs	54 EFPY[1] Fluence 1/4T Location, n/cm[2]	54 EFPY % Drop at 1/4T	54 EFPY C_VUSE at 1/4T
Reactor Vessel Beltline Region Location	Material ID	Heat Number	Type					
Regulatory Guide 1.99, Revision 2, Position 1.2								
Nozzle Belt Forging - Lower	AZJ 94	123V190	A508-64 Cl. 2	0.13	109	8.66E+18	21.3	86
Upper Shell Plate	C4344-1	C4344-1	SA-533 Gr B1	0.2	88	9.36E+18	28.5	63
Upper Shell Plate	C4344-2	C4344-2	SA-533 Gr B1	0.2	88	9.36E+18	28.5	63
Lower Shell Plate	C4347-1	C4347-1	SA-533 Gr B1	0.12	119	9.47E+18	20.7	94
Lower Shell Plate	C4347-2	C4347-2	SA-533 Gr B1	0.12	86	9.47E+18	20.7	68
Upper Shell Circ. Weld (Inside 40%)	SA-1769	71249	ASA/Linde 80	0.23	70	8.66E+18	EMA[2]	EMA[2]
Upper Shell Circ. Weld (Outside 60%)	WF-169-1	8T1554	ASA/Linde 80	0.16	70	Note 3	Note 3	Note 3
Upper Shell Axial Weld	WF-8	8T1762	ASA/Linde 80	0.19	70	8.42E+18	EMA[2]	EMA[2]
Upper Shell Axial Weld	WF-18	8T1762	ASA/Linde 80	0.19	70	8.42E+18	EMA[2]	EMA[2]
Upper Shell to Lower Shell Circ. Welds	WF-70	72105	ASA/Linde 80	0.32	70	9.12E+18	EMA[2]	EMA[2]
Lower Shell Axial Welds	SA-1580	8T1762	ASA/Linde 80	0.19	70	7.90E+18	EMA[2]	EMA[2]

Notes:

1. ¼ T Fluence values are calculated using RG 1.99, Revision 2, Equation (3) using inside wetted surface fluence values in Table 4.2-1. The base metal thickness is 8.44 in. and cladding thickness is 0.125 in.
2. Equivalent Margins Analyses (EMA) required because C_VUSE is less than 50 ft-lb.
3. The Upper Shell Circumferential Weld (Outside 60%) WF-169-1 is not located at the ¼ T location; therefore, C_VUSE projections are not applicable.

TABLE 4.2-3 EQUIVALENT MARGINS ANALYSIS FOR LEVEL A AND B SERVICE LOADS – J-INTEGRAL RESISTANCE AT A FLAW DEPTH OF ¼ T AT 54 EFPY

Beltline Weld ID	Surface Fluence (n/cm^2)	¼ T Fluence (n/cm^2)	$J_{0.1}$ material, Lower Bound ($in\text{-}lb/in^2$)	J_1 applied ($in\text{-}lb/in^2$)	$J_{0.1} / J_1$	Acceptance Criterion for $J_{0.1} / J_1$	Conclusion
54 EFPY Values							
WF-70	Not applicable (N/A)	9.12E+18	534	169	3.16	>1.0	Acceptable
WF-8, 18	N/A	8.42E+18	661	506	1.31	>1.0	Acceptable

TABLE 4.2-4 EQUIVALENT MARGINS ANALYSIS FOR LEVEL C AND D SERVICE LOADS - J-INTEGRAL RESISTANCE AT A FLAW DEPTH OF 1/10T AT 54 EFPY

Beltline Weld ID	Surface Fluence (n/cm^2)	$^1/_{10}$T Fluence (n/cm^2)	$J_{0.1}$ material, Lower Bound ($in\text{-}lb/in^2$)	J_1 applied ($in\text{-}lb/in^2$)	$J_{0.1} / J_1$	Acceptance Criterion for $J_{0.1} / J_1$	Conclusion
54 EFPY Values							
WF-70	1.56E19	1.27E+19	523	65	8.05	>1.0	Acceptable
WF-8, 18	1.44E19	1.18E+19	653	165	3.96	>1.0	Acceptable

TABLE 4.2-5 PTS REFERENCE TEMPERATURE EVALUATION THROUGH YEAR 60 (54 EFPY)

Reactor Vessel Beltline Region Material	Material ID	Heat Number	Type	Cu wt%	Ni wt%	Chem. Factor	Initial RT_{NDT} (°F)	54 EFPY Fluence at Inside Wetted Surface (n/cm^2)	Fluence Factor	ΔRT_{NDT} (°F)	Margin (°F)	RT_{PTS}	Screening Criteria (°F)
Nozzle Belt Forging - Lower	AZJ 94	123V190	A-508-64, Cl. 2	0.13	0.72	94.0	+3	1.48E+19	1.109	104.2	70.7	177.9	270
Upper Shell Plate	C4344-1	C4344-1	SA-533, Gr. B, Cl 1	0.20	0.54	115.8(1)	+20	1.60E+19	1.130	130.8	17.0	167.8	270
Upper Shell Plate	C4344-2	C4344-2	SA-533, Gr. B, Cl 1	0.20	.054	141.8	+20	1.60E+19	1.130	160.2	34.0	214.2	270
Lower Shell Plate	C4347-1	C4347-1	SA-533, Gr. B, Cl 1	0.12	0.58	82.6	-10	1.62E+19	1.133	93.6	34.0	117.6	270
Lower Shell Plate	C4347-2	C4347-2	SA-533, Gr. B, Cl 1	0.12	0.58	82.6	+45	1.62E+19	1.133	93.6	34.0	172.6	270
Upper Shell Circ. Weld (Inside 40%)	SA-1769	71249	ASA/ Linde 80	0.23	0.59	167.6	+10	1.48E+19	1.109	185.8	56.0	251.8	300
Upper Shell Circ. Weld (Outside 60%)	WF-169-1	8T1554	ASA/ Linde 80	0.16	0.57	143.9	-5	N/A(2)	N/A	N/A	N/A	N/A	300
Upper Shell Axial Weld	WF-8	8T1762	ASA/ Linde 80	0.19	0.57	152.4	-5	1.44E+19	1.101	167.8	68.5	**[231.3]**	270
Upper Shell Axial Weld	WF-18	8T1762	ASA/ Linde 80	0.19	0.57	152.4	-5	1.44E+19	1.101	167.8	68.5	**[231.3]**	270
Upper Shell to Lower Shell Circ Weld	WF-70	72105	ASA/ Linde 80	0.32	0.58	199.3	-26	1.56E+19	1.123	223.8	56.0	253.8	300
Lower Shell Axial Welds	SA-1580	8T1762	ASA/ Linde 80	0.19	0.57	152.4	-5	1.35E+19	1.083	165.1	68.5	228.6	270

Notes:

1. The chemistry factor was determined from surveillance data.
2. The Upper Shell Circumferential Weld (Outside 60%), WF-169-1, does not contact the inside surface, so there is no inside surface fluence calculated for this weld.

[xxx] - Limiting reactor vessel beltline region materials in accordance with 10 CFR 50.61.

TABLE 4.2-6 ADJUSTED REFERENCE TEMPERATURE PROJECTIONS AT 54 EFPY

Material Description				Initial RT_{NDT}	Chemistry Factor	54 EFPY Fluence 10^{19} n/cm^2			ΔRT_{NDT}, °F at 54 EFPY		Margin		ART, °F at 54 EFPY	
Reactor Vessel Beltline Region Location	Matl. ID	Heat Number	Type			Inside surface	¼T Location	¾T Location	¼T Location	¾T Location	¼T Location	¾T Location	¼T Location	¾T Location
Nozzle Belt Forging - Lower	AZJ 94	123V 190	SA-508, Cl. 2	+3	94.0	1.48	0.866	0.314	90.2	64.1	70.7	70.7	163.9	137.9
Upper Shell Plate	C4344-1	C4344-1	SA-533, Gr. B, Cl. 1	+20	115.8	1.60	0.936	0.340	113.6	81.4	17.0	17.0	150.6	118.4
Upper Shell Plate	C4344-2	C4344-2	SA-533, Gr. B, Cl. 1	+20	141.8	1.60	0.936	0.340	139.2	99.6	34.0	34.0	193.2	153.6
Lower Shell Plate	C4347-1	C4347-1	SA-533, Gr. B, Cl. 1	-10	82.6	1.62	0.947	0.344	81.4	58.3	34.0	34.0	105.4	82.3
Lower Shell Plate	C4347-2	C4347-2	SA-533, Gr. B, Cl. 1	+45	82.6	1.62	0.947	0.344	81.4	58.3	34.0	34.0	160.4	137.3
Upper Shell Circ. Weld (inside 40%)	SA-1769	71249	Linde 80	+10	167.6	1.48	0.866	N/A(1)	160.8	N/A	56.0	N/A	**[226.8]**	N/A
Upper Shell Circ. Weld (Outside 60%)	WF-169-1	8T1554	Linde 80	-5	143.9	1.48	N/A(2)	0.314	N/A	98.2	N/A	68.5	N/A	161.7
Upper Shell Axial Weld	WF-8	8T1762	Linde 80	-5	152.4	1.44	0.842	0.306	145.1	102.9	68.5	68.5	208.5	166.4
Upper Shell Axial Weld	WF-18	8T1762	Linde 80	-5	152.4	1.44	0.842	0.306	145.1	102.9	68.5	68.5	208.5	166.4
Upper Shell to Lower Shell Circ. Weld	WF-70	72105	Linde 80	-26	199.3	1.56	0.912	0.331	194.2	138.7	56.0	56.0	224.2	**[168.7]**
Lower Shell Axial Welds	SA-1580	8T1762	Linde 80	-5	152.4	1.35	0.790	0.287	142.3	100.4	68.5	68.5	205.8	163.9

Notes:

1. The Upper Shell Circumferential Weld (Inside 40%), SA-1769, is not located at the ¾T location, so there is no ¾T fluence calculated for this weld.
2. The Upper Shell Circumferential Weld (Outside 60%), WF-169-1, is not located at the ¼T location, so there is no ¼T fluence calculated for this weld.

[xxx] – Controlling values of adjusted reference temperature.

4.3 METAL FATIGUE

Several thermal and mechanical fatigue analyses of plant mechanical components have been identified as time-limited aging analyses (TLAAs) for CR-3. These are discussed in the following Subsections.

The evaluation of components is used to demonstrate compliance with 10 CFR 54.21(c)(1) by using a combination of the methods of 54.21(c)(1)(i) for analyses that remain valid for the period of extended operation, 54.21(c)(1)(ii) for analyses that have been projected to the end of the period of extended operation, and 54.21(c)(1)(iii) for monitoring of design transients and managing the effects of aging for the period of extended operation. The following sections provide a summary of the evaluation results for each of the major components evaluated.

4.3.1 FATIGUE ANALYSES (NSSS COMPONENTS)

The CR-3 approach is to identify the latest design fatigue analyses associated with each NSSS component within the reactor coolant pressure boundary (RCPB) in order to demonstrate that the design analyses will remain bounding through the period of extended operation. Components within the scope of this review include non-pressure boundary reactor internals components.

Original fatigue design calculations assumed a large number of design transients corresponding to relatively severe system dynamics over the original 40-year design life. In general, actual plant operations have resulted in only a fraction of the originally expected fatigue duty. An assessment of the number of NSSS design transients that have occurred through December 2007 was compiled to determine the margin between the number of accrued cycles and the original 40-year design cycles.

The first step in the evaluation was to establish the current fatigue design bases for the major NSSS components. This was done by reviewing component design reports, amendments to those reports, and the assessment of the impact of the NRC approved measurement uncertainty recapture 1.6% power uprate to identify the full set of NSSS design transients used in the fatigue evaluations. The governing NSSS Design Transients are those identified in the CR-3 FSAR, Table 4-8, and listed in Table 4.3-1. Cumulative Usage Factor (CUF) values were compiled from CR-3 component design documents and are presented in Table 4.3-2.

The second step in the evaluation was to gather and review plant design information, actual plant transient data from the RCS and other sources, and archived RCS operational parametric data. This information was used to develop actual operational transients experienced from plant startup through December 2007. The transient data was obtained from the CR-3 Cycle and Transient Monitoring Program, input from plant personnel, and historical data obtained from CR-3 records.

There is considerable margin after 30 years of operation to the NSSS design transient cycles originally defined for 40 years, and CR-3 has determined there is no need to increase the number of NSSS design transients for the period of extended operation. The RCS CUFs may be conservatively projected to 60 years of operation by multiplying the 40-year CUFs by a factor of 1.5; this is equivalent to multiplying the NSSS design transient cycles by a factor of 1.5. Therefore, 40-year usage factors in excess of 0.67 (i.e., 1.0/1.5) may be assumed to exceed the ASME Code, Section III limit of 1.0 at 60-years. This method of usage factor projection is conservative since CR-3 has determined that it is unlikely that the NSSS design transients for 40 years will be exceeded at 60 years of operation.

The final step in the evaluation was to consider the effects of the reactor water environment on 40-year fatigue usage factors at selected NSSS locations as identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Design Curves to Selected Nuclear Power Plant Components," as required by NUREG-1801, Revision 1. This assessment is provided in Subsection 4.3.3.

The following subsections provide a summary of the fatigue analyses evaluation results for each of the major NSSS components evaluated.

4.3.1.1 Reactor Vessel

Summary Description

The reactor vessel (RV) was designed in accordance with Section III of the ASME Code – Class 1, for the replacement closure head, and Class A, for the remaining vessel items; therefore, metal fatigue was considered in the design of the RV components. CUF analyses for the RV are applicable TLAAs, since they are based on NSSS design transient cycles originally defined for 40 years. The NSSS Design Transients are those identified in Table 4.3-1. Forty-year design CUF values for the RV items are identified in Table 4.3-2.

Analysis

For the components that are part of the RV, one pressure-retaining item has a 40-year CUF that exceeds 0.67: the Lower Service Support Structure attachment weld with a CUF of 0.72. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation using the RCPB Fatigue Monitoring Program.

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.1.2 Reactor Vessel Internals

Summary Description

The CR-3 reactor vessel internals (RVI) were designed and constructed prior to the development of ASME Code requirements for core support structures. Therefore, existing industry structural practice was used in the design of the internals structural members; and the only specific fatigue analyses performed in the original design were those that addressed high cycle fatigue reported in BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibration," September 1, 1972. In modifications following original design, plant-specific fatigue analyses were performed for the reactor vessel internals replacement bolts as presented in BAW-1843PA, "The B&WOG Evaluation of Internals Bolting Concerns in 177 FA Plants," January 1986, and BAW-1789P, "The B&WOG Evaluation of Internals Bolting Concerns in 177 FA Plants," August 1984. These topical reports summarize fatigue analyses performed to the ASME Code, Section III, Subsection NG, including both high-cycle fatigue from flow induced vibrations (FIV) and low-cycle fatigue from NSSS design transients. The NSSS Design Transients are those identified in Table 4.3-1. Forty-year design CUF values for the replacement RVI bolts are identified in Table 4.3-2.

Analysis

FIV Endurance Limit Assumptions

BAW-10051 calculated stress values for the redesigned RVI and compared them to endurance limit stress values. These endurance limit values were based on an assumed value of 1012 cycles for 40 years of operation. Since the fatigue curves at the time of design only went up to 106 cycles, these curves were extrapolated to 1012 cycles. The methodology used in BAW-10051 was extended from 40 years to 60 years by multiplying the assumed endurance limit cycles by 1.5 and then using 1013 cycles to determine the endurance limit based on more recent ASME fatigue curves which extend now to 1011 cycles. The component item stress values in BAW-10051 were compared to the recalculated endurance limit values and were shown to be acceptable. Therefore, the FIV analysis has been projected to the end of the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(ii) – The analysis has been projected to the end of the period of extended operation.

Cumulative Usage Factors for RV Internals Replacement Bolts

The RV internals bolts that were replaced at CR-3 include 120 Upper Core Barrel bolts made from A-286, 60 Lower Core Barrel bolts made from X-750, 96 Lower Thermal Shield bolts made from X-750, and 72 Surveillance Specimen Holder Tube (SSHT) bolts made from X-750. Two of these sets of replacement bolts have 40-year CUFs that exceed 0.67. These are the lower core barrel bolts with CUF of 0.759 and the lower thermal shield bolts with CUF of 0.84. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation using the CR-3 RCPB Fatigue Monitoring Program.

Disposition: 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.3.1.3 Control Rod Drive Mechanism

Summary Description

The "Type C" control rod drive mechanism (CRDM) motor tube was designed in accordance with ASME Code, Section III, Class A, 1968 Edition with Addenda through Summer 1970, and metal fatigue was considered in the design of the component. CUFs of the CRDM motor were not calculated as it was shown that the motor tube did not require analysis for cyclic operation in accordance with ASME Section III, paragraph

N-415.1. The calculations performed to comply with N-415.1 are applicable TLAAs since they are based on NSSS design transient cycles originally defined for 40 years of operation. The NSSS design transients are those identified in Table 4.3-1.

Analysis

Calculations performed in accordance with N-415.1(a) through N-415.1(f) of the ASME Code, Section III for the CRDM motor tube are based on NSSS design transients. The NSSS design transients for CR-3 have not been increased for the period of extended operation. Therefore, the analyses performed in accordance with N-415.1(a) through N-415.1(f) of the ASME Code, Section III are acceptable for the period of extended operation since the NSSS design transients have not been revised.

Disposition: 10 CFR 54.21(c)(1)(i) – The analyses remain valid for the period of extended operation.

4.3.1.4 Reactor Coolant Pumps

Summary Description

The reactor coolant pumps (RCPs) were designed in accordance with the ASME Code, Section III, Class A, but were not code stamped, and metal fatigue was considered in the design of the component. CUFs of the RCPs are applicable TLAAs since the CUFs are based on NSSS design transient cycles originally defined for 40-years of operation. The NSSS Design Transients are those identified in Table 4.3-1. Forty-year design CUF values for the RCP items are identified in Table 4.3-2.

Analysis

The RCP items listed in Table 4.3-2 have CUFs below 0.67. The RCP pump cover has the largest 40-year design usage factor at 0.65.

Calculations performed in accordance with N-415.1(a) through N-415.1(f) of the ASME Code, Section III, for the RCP seal and heat exchanger are based on NSSS design transients. The NSSS design transients for CR-3 have not been increased for the period of extended operation.

Based on the above, the analyses for the RCP casing, cover, and shaft have been projected to the end of the period of extended operation, and the analyses of the RCP seal and heat exchanger performed in accordance with N-415.1(a) through N-415.1(f) of the ASME Code, Section III, are acceptable for the period of extended operation since the NSSS design transients have not been revised.

Disposition: **10 CFR 54.21(c)(1)(i) – The analyses remain valid for the period of extended operation, and**
10 CFR 54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation since the maximum CUF for RCP items is less than 0.67.

4.3.1.5 Steam Generators

Summary Description

The Once-Through Steam Generators (OTSGs) were designed in accordance with the ASME Code, Section III, Class A, and metal fatigue was considered in the design of the components. CUFs of the OTSG components are applicable TLAAs since the CUFs are based on NSSS design transient cycles originally defined for 40 years of operation. The NSSS Design Transients are those identified in Table 4.3-1. Forty-year design CUF values for the OTSG components are identified in Table 4.3-2.

Analysis

For the components that are part of the OTSG, five items have 40-year CUFs that exceed 0.67: the Emergency Feedwater (EFW) Nozzle Studs, Main Feedwater (MFW) Nozzle, Mechanical Sleeves, Remote Welded Plug, and the Support Skirt. The CUF values for these components are:

EFW Nozzle Studs	0.97
MFW Nozzle	0.92
Mechanical Sleeve	0.904
Remote Welded Plug	0.90
Support Skirt	0.89

Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation by means of the CR-3 RCPB Fatigue Monitoring Program.

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.1.6 Pressurizer

Summary Description

The Pressurizer was designed in accordance with the ASME Code, Section III, Class A, and metal fatigue was considered in the design of the component. The Pressurizer surge nozzle was modified in 2007 to include a weld overlay over the Alloy 600 weld that connects the surge nozzle to a stainless steel safe end. The weld overlay was designed in accordance with the 1989 Edition of ASME Code, Section III, Subsection NB. CUFs for the Pressurizer are applicable TLAAs since they are based on NSSS design transient cycles originally defined for 40 years. The NSSS Design Transients are those identified in Table 4.3-1. Forty-year design CUF values for the Pressurizer are identified in Table 4.3-2.

Analysis

For the components that are part of the Pressurizer, three items have 40-year CUFs that exceed 0.67: the Surge Nozzle with weld overlay, the Heater Bundle closure seal weld, and the Thermowell Nozzle. The CUF for these components are:

Surge Nozzle with weld overlay	0.81
Heater Bundle closure seal weld	0.86
Thermowell Nozzle	0.71

Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation by the CR-3 RCPB Fatigue Monitoring Program.

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.1.7 Reactor Coolant Pressure Boundary Piping (USAS B31.7)

Summary Description

RCPB piping includes all piping within the ASME, Section XI, Subsection IWB inspection boundary at CR-3. The IWB inspection boundary includes B&W-supplied main coolant piping and portions of Architect/Engineer-supplied ancillary systems, e.g., Decay Heat Removal, Core Flood, and Make Up & Purification Systems, including Low Pressure Injection, High Pressure Injection, and Makeup/Letdown piping, attached to the Reactor Coolant System piping. The IWB inspection boundary within the ancillary systems typically extends to the first or second isolation valve or to a flow restricting orifice. The B&W-supplied main coolant piping was designed in accordance with USAS B31.7, and the ancillary systems connected to the main coolant piping were designed in accordance with USAS B31.1. TLAAs for the RCPB piping include CUFs for B31.7 designed piping, which are addressed in this subsection, and stress range reduction factors for B31.1 designed piping, which are addressed in Subsection 4.3.2.

The scope of USAS B31.7 piping at CR-3 includes the 36 in. hot leg piping, including attached branch connections and safe ends; 28 in. cold leg piping, including attached branch connections and safe ends; Pressurizer surge line piping; and Pressurizer spray line piping. CUFs of USAS B31.7 RCPB piping are applicable TLAAs since they are based on NSSS design transient cycles originally defined for 40 years of operation. The NSSS Design Transients are those identified in Table 4.3-1. Forty-year design CUF values for the RCPB piping are identified in Table 4.3-2.

Analysis

For the components that are part of the RCPB piping, the Pressurizer spray line piping and High Pressure Injection/Makeup (HPI/MU) Nozzle safe end CUFs exceed 0.67 at 40 years. The CUF of the Pressurizer spray line is 0.70, and the CUF of the HPI/MU Nozzle safe end is 0.95.

In accordance with NRC letter (H. Silver) to FPC (P. Beard), "Crystal River Unit 3 - NRC Bulletin 88-08 'Thermal Stress in Piping Connected to Reactor Coolant Systems,' (TAC No. M69621)," dated June 18, 1992, the piping items within the scope of NRC Bulletin 88-08 at CR-3 include the HPI/MU nozzle, safe end, and thermal sleeve. Fatigue of the HPI/MU nozzle, safe end, and thermal sleeve is evaluated above for the period of extended operation.

Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation by means of the CR-3 RCPB Fatigue Monitoring Program.

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.2 IMPLICIT FATIGUE ANALYSIS (B31.1 PIPING)

The RCPB piping evaluated in Subsection 4.3.1.7 includes the original B&W scope of supply that was designed in accordance with USAS B31.7. RCPB piping within ancillary systems attached to the main coolant piping and designed in accordance with USAS B31.1.0 are discussed in Subsection 4.3.2.1. Fatigue of Non-Class 1 piping designed to USAS B31.1.0 is discussed in Subsection 4.3.2.2.

4.3.2.1 USAS B31.1.0 Piping - RCPB Class 1

Summary Description

RCPB Class 1 piping designed in accordance with USAS B31.1.0 Piping Code includes piping in ancillary systems connected to the B&W-supplied main coolant piping. These systems include Decay Heat Removal, Core Flood, and Makeup & Purification Systems, including Low Pressure Injection, High Pressure Injection, and Makeup/Letdown piping.

The USAS B31.1.0 design does not require analyses of cumulative fatigue usage, but cyclic loading was considered in a simplified manner in the design process. The overall number of thermal cycles expected during the 40-year lifetime of these components was compared to limits (7,000 cycles or more), above which stress range reduction factors had to be applied to the allowable stress range for secondary stresses (expansion and displacement) to account for thermal cycling. These components are considered to have implicit fatigue analyses. Since the overall number of cycles could potentially increase during the period of extended operation, these implicit fatigue analyses are also considered to be TLAAs requiring evaluation for the period of extended operation.

For piping designed in accordance with the USAS B31.1.0-1967 Code rules, the designer was required to determine the overall number of thermal cycles anticipated for the component in 40 years, and was required to apply stress range reduction factors if this number exceeded 7,000. Power piping at CR-3 complies with USAS B31.1.0-1967. Since these analyses were based upon the number of cycles expected to occur during the original license period, these analyses are also considered to be TLAAs.

All RCPB piping attached to the B&W scope of supply was designed in accordance with USAS B31.1.0. The spool piece that is connected to the HPI/MU safe end was designed to USAS B31.1.0 but was analyzed for fatigue using USAS B31.7 in response to NRC Bulletin 88-08.

Analysis

USAS B31.1.0 Piping: RCPB Class 1 Transient Assumptions

The applicable transient cycles for piping systems designed in accordance with USAS B31.1.0-1967 rules were originally determined by summing the individual transients to which the component would be exposed in 40 years. In order to evaluate these TLAAs for 60 years, the numbers of cycles now expected to occur in 60 years should be compared to the numbers of design cycles that were considered in these analyses. For the RCPB systems, the number of thermal cycles correlates with plant heatups and cooldowns, which are limited to 240 cycles per Table 4.3-1. Since the transient set (and associated cycles) in the RCS Functional Specification is being maintained, the analytical basis for these components remains unchanged. Therefore, the analyses for these components remain valid for the period of extended operation in accordance with10 CFR 54.21(c)(1)(i).

Disposition: **10 CFR 54.21(c)(1)(i) – The analyses remain valid for the period of extended operation.**

Cumulative Usage Factor for HPI/MU Safe End Spool Piece

The HPI/MU safe end is welded to a stainless steel spool piece that was analyzed for fatigue analysis in accordance with USAS B31.7 to support NRC Bulletin 88-08. The 40-year CUF for the spool piece is 0.94. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation by means of the RCPB Fatigue Monitoring Program.

Disposition: **10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.**

4.3.2.2 USAS B31.1.0 Piping - Non-Class 1

Summary Description

Piping designed in accordance with USAS B31.1.0 Piping Code was not required to have analyses of cumulative fatigue usage, but cyclic loading was considered in a simplified manner in the design process. The overall number of thermal cycles expected during the 40-year lifetime of these components was compared to limits (7,000 cycles or more), above which stress range reduction factors had to be applied to the allowable stress range for secondary stresses (expansion and displacement) to account for thermal cycling. These Non-Class 1 components are considered to have implicit fatigue analyses. Since the overall number of cycles could potentially increase during the period of extended operations, these implicit fatigue analyses are also considered to be TLAAs requiring evaluation for the period of extended operation.

For piping designed in accordance with the USAS B31.1.0-1967 code rules, the designer was required to determine the overall number of thermal cycles anticipated for the component in 40 years, and was required to apply stress range reduction factors if this number exceeded 7,000. Power piping at CR-3 complies with USAS B31.1.0-1967. Since these analyses were based upon the number of cycles expected to occur during the original license period, these analyses are also considered to be TLAAs.

Analysis

Components with Cycles Related to RCS Heatups and Cooldowns

The applicable transient cycles for piping systems designed in accordance with USAS B31.1.0-1967 rules were originally determined by summing the individual transients to which the component would be exposed in 40 years. In order to evaluate these TLAAs for 60 years, the numbers of cycles now expected to occur in 60 years should be compared to the numbers of design cycles that were considered in these analyses. For most systems, the number of thermal cycles correlates with plant heatups and cooldowns, which are limited to 240 cycles per Table 4.3-1. The applicable systems include:

- Steam and power conversion systems and components, and
- ESF Systems connected to the RCS.

Since the transient set (and associated cycles) in the RCS Functional Specification is being maintained, the analytical basis for these components remains unchanged. Therefore, the analyses for these components 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 analyses remain valid for the period of extended operation.

<u>Components with Cycles Unrelated to RCS Heatups and Cooldowns</u>

For components in systems whose cycles do not track plant heatups and cooldowns, a specific evaluation of the components operating history was performed. Examples of components in this group include:

- Engine exhaust components for diesel engines in the Emergency Diesel Generator, Emergency Feedwater and Fire Protection Systems,
- Sampling piping and components in the Liquid and Post-Accident Liquid Sampling Systems, and the
- Turbine-Driven Emergency Feedwater Pump Turbine.

Evaluations were performed that projected the number of expected cycles in 60 years. The evaluations concluded that the components remain qualified through the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation.

4.3.3 ENVIRONMENTALLY-ASSISTED FATIGUE ANALYSIS

The effects of reactor water environment on fatigue were evaluated for a subset of representative components. The representative components selected were based upon the evaluations in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The representative components evaluated are as follows:

- Reactor Vessel Shell and Lower Head (including incore instrumentation nozzles)
- Reactor Vessel Inlet and Outlet Nozzles
- Pressurizer Surge Line (including hot leg and Pressurizer surge nozzles)

- HPI/MU Nozzle
- Core Flood Nozzle
- Decay Heat Removal System Class 1 Piping

The methods used to evaluate environmental effects on fatigue were based on NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels," NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," and NUREG/CR-6717, "Environmental Effects of Fatigue Crack Initiation in Piping and Pressure Vessel Steels." In addition, the method used to obtain environmental effects for nickel-based alloy was obtained from H. S. Metha and S. R. Goeeslin, "Environmental Factor Approach to Account for Water Effects in Pressure Vessel and Piping Fatigue Evaluations," Nuclear Engineering and Design, 1998. Environmental fatigue life correction factors (F_{en}) were used to obtain adjusted cumulative fatigue usage (U_{en}) which includes the effects of reactor water environments.

Environmentally-adjusted U_{en} factors are summarized in Table 4.3-3. Evaluations at all locations are based on application of environmental penalty factors to the ASME 40-year CUF values. Bounding F_{en} values of 2.45 for low-alloy steel, 15.35 for stainless steel, and 1.49 for Alloy 600 were applied to the 40-year design CUFs with the exception of surge line piping and decay heat injection piping.

For surge line piping, the ASME Section III analysis of record for CR-3 was revised to include the effects of environmentally assisted fatigue. The environmental correction factor F_{en} from NUREG/CR-5704 was used to determine the number of allowable cycles for each load pair. The F_{en} correction factor was obtained by integration from peak to valley considering transformed metal temperature, transformed strain rate, and transformed dissolved oxygen. The strain rate was assumed to be at 0.0004%/sec or less, and transformed strain rate was held constant at Ln (0.001). Based on historical data, dissolved oxygen is 0.05 ppm or less, and transformed oxygen was held constant at 0.026. Transformed metal service temperature was determined by integration of metal temperature for the load pair analyzed. Therefore, the F_{en} varies from 2.55 (when metal temperature is less than 392 °F) to a maximum of 15.35 (when metal temperature equals or exceeds 392 °F). Thermal striping, which was considered separately, was assigned an F_{en} of 1.0 as the maximum calculated strain amplitude is less than the threshold strain amplitude of 0.097% listed in NUREG/CR-5704.

The Decay Heat Injection piping at CR-3 was designed in accordance with USAS B31.1 and therefore did not receive an explicit CUF evaluation. A fatigue evaluation of the Decay Heat Injection piping was performed specifically for License Renewal using USAS B31.7, 1969 Edition.. The CUF was multiplied by the bounding F_{en} value of 2.55.

Based on the results of this evaluation, and in accordance with 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function(s) will be adequately managed for the period of extended operation using the CR-3 RCPB Fatigue Monitoring Program.

Disposition: 10 CFR 54.21(c)(1)(iii) – The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.3.4 RCS LOOP PIPING LEAK-BEFORE-BREAK ANALYSIS

Summary Description

The successful application of leak-before-break (LBB) to the CR-3 RCS main coolant piping is described in Topical Report BAW-1847, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," Revision 1, September 1985. This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping (36 in. hot leg piping and 28 in. cold leg piping) under normal plus faulted loading conditions and was approved by the NRC for the current term of operation. The TLAA in report BAW-1847, Revision 1, addresses fatigue flaw growth. In addition, Section 3.3.4.3 of the report includes a qualitative assessment of thermal aging of cast austenitic stainless steel (CASS) RCP inlet and exit nozzles; this assessment is not considered a TLAA. However, reduction of fracture toughness by thermal aging of the RCP inlet and exit nozzles was evaluated for License Renewal to ensure that the conclusions of the LBB evaluation reported in BAW-1847, Revision 1, remain valid for the period of extended operation.

Analysis

<u>Fatigue Flaw Growth</u>

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." Specifically, a surface flaw was postulated at selected locations of the piping system (i.e., highest stress coincident with the lower bound of the material properties for base metal, weldments, and safe ends); and a fatigue crack growth analysis for postulated flaws was then performed to demonstrate that the surface flaws are likely to propagate in the through-wall direction and develop leakage before they will propagate circumferentially around the pipe. Flaw growth calculations are reported in Section 4.3, Table 4-3, of BAW-1847, Revision 1, and are based on 240 heatup and cooldown cycles and 22 cycles of safe shutdown earthquake.

The original transient cycles that were defined for 40 years of operation for the RCS components have not been revised for License Renewal and are being monitored by the CR-3 Reactor Coolant Pressure Boundary Fatigue Monitoring Program. If a transient cycle count approaches or exceeds the allowable design limit, corrective actions are taken. Therefore, the flaw growth evaluation reported in BAW-1847, Revision 1, remains valid for the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(i) since CR-3 has not revised the transients defined in the RCS design specification for License Renewal.

Thermal Aging of CASS RCP Suction and Discharge Nozzles

The susceptibility of the RCS main coolant piping to thermal aging was qualitatively addressed in Section 3.3.4.3 of BAW-1847, Revision 1. As described in BAW-2243A, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," The B&W Owners Group Generic License Renewal Program, June 1996, there are no RCS main coolant piping segments fabricated from CASS. However, the heat affected zone of the welded joint that connects the wrought austenitic stainless steel 28 in. pump transition piece to the CASS RCP inlet and exit nozzles may be susceptible to thermal embrittlement. Limited data regarding thermal aging of CASS material was available at the time of the preparation of BAW-1847, Revision 1. In the report, the values of fracture toughness for aged CASS were assumed to be bounded by the ferritic piping and ferritic weldments. Since the publication of BAW-1847, Revision 1, a significant amount of data has been obtained regarding thermal aging of CASS materials. Test data obtained from an Argonne National Laboratory Report by Chopra and Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, U.S. Nuclear Regulatory Commission, Washington DC, May 1994, indicate that prolonged exposure of CASS to reactor coolant operating temperatures can lead to reduction of fracture toughness by thermal embrittlement. The fracture toughness curves for the ferritic base metal and ferritic weld metals used in the RCS piping LBB analysis were compared to the lower-bound fracture toughness curves of CR-3 RCP CASS materials (i.e., statically cast CF8M) from the Argonne report. The fracture toughness curve of the lower-bound CASS material is below the fracture toughness curves used in the RCS piping LBB analysis. Therefore, the assumption in BAW-1847, Revision 1, that the fracture toughness of the ferritic piping and ferritic weldments bounds the fracture toughness of CASS required further evaluation for License Renewal.

A flaw stability analysis was performed using the lower-bound CASS fracture toughness curves from the Argonne report cited above to show acceptability of LBB for the RCS main coolant piping for the period of extended operation. The most limiting material and location used in the RCS piping LBB analysis (i.e., BAW-1847, Revision 1) was determined to be the base metal material of the straight section of the 28 in. cold leg pipe. Both the suction and discharge nozzles of the RCP casings are attached to the 28 in. cold leg pipes and have similar geometries and applied loads as the limiting location used for the LBB analysis. The discharge and suction nozzles of the RCP casings were evaluated for LBB using lower-bound CASS fracture toughness properties.

Bounding 10 gpm leakage crack sizes (i.e., a margin of 10 on the plant's leak detection capability) for the RCP suction and discharge nozzle were determined using a method that is consistent with that reported in BAW-1847, Revision 1. In the revised analysis, the applied loadings were considered using the absolute sum load combination method. Therefore, in accordance with NUREG-0800, Standard Review Plan (SRP) 3.6.3, a margin of 1.0 on load was used. The leakage flaw size for the suction nozzle was determined to be 4.31 in. and the leakage flaw size for the discharge nozzle was determined to be 4.43 in. In addition, a crack extension value of 0.6 in. was considered in the flaw stability analysis. A flaw stability analysis was performed for the RCP inlet (suction) and exit (discharge) nozzles, and the discharge nozzle was found to be limiting. The maximum applied J value at the discharge nozzle, for the 10 gpm leakage flaw size, was determined to be 0.510 kips/in. The critical crack size was determined to be 10.8 in. Therefore, the margin on flaw size was determined to be 2.4 (i.e., 10.8/4.43). This is greater than the required margin of 2.0 in accordance with SRP 3.6.3. Based on the results of this analysis, it is concluded that the required margins for LBB per SRP 3.6.3 are met, even with consideration of the lower-bound CASS fracture toughness properties for the suction and discharge nozzles.

<u>Summary: Leak-Before-Break for the Period of Extended Operation</u>

In summary, it has been demonstrated that the fatigue flaw growth analysis reported in BAW-1847, Revision 1, remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) since the number of NSSS design transients will not be revised for License Renewal. The remainder of the generic LBB analysis for the B&W operating plants reported in BAW-1847, Revision 1, remains valid for the period of extended operation with the exception of the original qualitative assessment of reduction of fracture toughness by thermal aging of CASS. The assessment of reduction of fracture toughness by thermal aging of CASS is not considered a TLAA. Reduction of fracture toughness of the RCP nozzles was determined to be acceptable for the period of extended operation through the flaw stability analysis described above. In addition, recent NRC concerns related to Alloy 82/182 and LBB analyses are addressed in the industry's submittal MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," EPRI, Palo Alto, CA: 2005, 1011808. The Alloy 82/182 welds within the scope of BAW-1847, Revision 1, are the welds that connect the 28 in. stainless steel carbon steel cold leg piping to the stainless safe pump transition pieces. Based on the above, the flaw growth analysis remains valid for the period of extended operation.

Disposition: 10 CFR 54.21(c)(1)(i) – The RCS loop LBB analysis remains valid for the period of extended operation.

TABLE 4.3-1 NSSS TRANSIENT CYCLES

ID. No.	ASME Transient Classification	Transient Description[1]	40-Year Design Cycles
1A	Normal	RCS Heatup 70°F to 557°F at 100 °F/hr	240
1B	Normal	RCS Cooldown 557°F to 70°F at 100 °F/hr	240
7		Step Load Reduction (100% to 8% Power)	
	Upset	Resulting from Turbine Trip	160
	Upset	Resulting from Electrical Load Rejection	150
8		Reactor Trips	
	Upset	Resulting from Loss of All Reactor Coolant Pumps	40
	Upset	Due to Turbine Trip Without Automatic Control Action	160
	Upset	Resulting from Complete Loss of All Main Feedwater	88
	Upset & Emerg.	Included in Transients 11, 15, 17A, and 17B	110
9	Upset	Rapid Depressurization (2,200 psi to 300 psi in 1 hr)	40
10	Upset	Change of Flow (Loss of One or More Reactor Coolant Pumps)	20
11	Upset	Rod Withdrawal Accident	40
12		Hydrostatic Tests at 3,125 psig	
	Test	RCS Components (Primary Side)	20
	Test	OTSG A (Secondary Side)	35
	Test	OTSG B (Secondary Side)	35
15	Upset	Loss of Station Power	40
17A	Upset	Loss of Feedwater to One OTSG	20
17B	Emergency	Stuck Open Turbine Bypass Valve	10
22		High Pressure Injection Valve Test	
	Normal	Actuation of Makeup Valve MUV-23	40
	Normal	Actuation of Makeup Valve MUV-24	40
	Normal	Actuation of Makeup Valve MUV-25	40
	Normal	Actuation of Makeup Valve MUV-26	40
22		Core Flooding Check Valve Test	
	Normal	Core Flood Tank CFT-1A	240
	Normal	Core Flood Tank CFT-1B	240
8		High Pressure Injection Actuations	
	Upset	MUV-23	11
	Upset	MUV-24	11
	Upset	MUV-25	11
	Upset	MUV-26	11
No ID		High Pressure Auxiliary Pressurizer Spray	15
14	Upset	Control Rod Drop	40
25	Upset	Refill of Hot, Dry Depressurized OTSG	
		OTSG A	50
		OTSG B	50
26	Upset	Emergency Feedwater Actuation	
		Flow Initiation to OTSG A Upper Feed Nozzles	1510
		Flow Initiation to OTSG B Upper Feed Nozzles	1510

Note:

1. Consists of the transients tracked as part of the RCPB Fatigue Monitoring Program.

TABLE 4.3-2 DESIGN FATIGUE USAGE FACTORS

List No.	Component	Location	40-Year Fatigue Usage
1	Reactor Vessel		
2		Control Rod Drive Nozzle (J-Groove Weld)	0.65
3		Closure Head Dome-to-Flange	0.03
4		RV Flange-to-Shell Transition	0.02
5		Closure Head Studs	0.43
6		RV Inlet Nozzle	0.11
7		RV Outlet Nozzle	0.46
8		RV Lower Head	0.022
9		Bottom-Mounted Nuclear Instrument Nozzle	0.58
10		Core Flood Nozzle-Base of Nozzle	0.26
11		RV Support Skirt	0.085
12		Lower Service Support Structure Attachment Weld	0.72
13	Reactor Vessel Internals		
14		Upper Core Barrel Replacement Bolts	0.0
15		Lower Core Barrel Replacement Bolts	0.759
16		Thermal Shield (Lower) Replacement Bolts	0.84
17		Surveillance Specimen Holder Tube Replacement Bolts	<0.001
18	Control Rod Drive Mechanism		
19		Type C-Motor Tube Housings and Extension	Exempt
20	Reactor Coolant Pumps		
21		RCP Casing (Volute/Upper Flange)	0.32
22		RCP Cover	0.65
23		RCP Lower Shaft	0.007
24		RCP Seal & Heat Exchanger	Exempt
25	Steam Generators		
26		Upper and Lower Tubesheet	0.13
27		Primary Inlet Nozzle	0.03
28		Primary Outlet Nozzle	0.03
29		Emergency Feedwater Nozzle	0.57
30		Emergency Feedwater Nozzle Studs	0.97
31		Emergency Feedwater Sleeve Retainer Bar	0.67
32		OTSG Shell	0.00
33		Main Feedwater Nozzle	0.92
34		Mechanical Sleeve	0.904

TABLE 4.3-2 (continued) DESIGN FATIGUE USAGE FACTORS

List No.	Component	Location	40-Year Fatigue Usage
35	Steam Generators (continued)	Remote Weld Plug (plug-to-tubesheet weld)	0.90
36		Steam Generator Support Skirt	0.89
37	Pressurizer		
38		External Supports (Shell)	0.02
39		Shell	0.10
40		Surge Nozzle (weld overlay)	0.8136
41		Spray Nozzle	0.143
42		Heater Bundle Closure (seal weld)	0.86
43		Heater Stud	0.30
44		Thermowell Nozzle	0.71
45		Heater Bundle Closure Diaphragm	0.60
46	Reactor Coolant Pressure Boundary Piping		
47		Cold Leg (28 in.) Hot Leg (36 in.)	0.362 0.351
48		Surge Line Elbows	0.37
49		Surge Line-Non Elbows	0.40
50		Hot Leg Surge Line Nozzle	0.143 (CS nozzle) 0.118 (SS pipe)
51		Spray line piping	0.70
52		High Pressure Injection/Makeup Safe End Spool piece-to-safe end weld (B31.1 piping)	0.95 0.94
53		Cold Leg Spray Nozzle	0.63

TABLE 4.3-3 ENVIRONMENTALLY-ADJUSTED CUF VALUES

Component	Environmentally Adjusted CUF	F_{en}	A	B	C	54.21 (c) (1)
Reactor Vessel Shell and Lower Head (LAS)	0.053	2.45	NA	NA	NA	(iii)
Incore Instrumentation Nozzle (Ni-Cr-Fe)	0.86	1.49	NA	NA	NA	(iii)
Reactor vessel inlet nozzle (LAS)	0.27	2.45	NA	NA	NA	(iii)
Reactor vessel outlet nozzle (LAS)	0.76 (Note 1)	2.45	X (Note 1)	NA	NA	(iii)
Surge line piping up to but not including weld piping next to weld overlays (SS)	1.54	$2.55<F_{en}<15.35$ (Note 2)	X (Note 2)	X (Note 2)	NA	(iii)
Surge line hot leg nozzle and stainless steel piping adjacent to weld overlay (SS)	0.29	15.35	NA	NA)	NA	(iii)
Surge line Pressurizer nozzle and stainless steel safe end adjacent to weld overlay (SS)	0.95	15.35	NA	X (Note 3)	NA	(iii)
Core flood nozzle (LAS)	0.64	2.45	NA	NA	NA	(iii)
HPI/MU nozzle (SS safe end)	1.89	15.35	NA	NA	NA	(iii)
Decay heat injection Class 1 piping (Stainless Steel Tee)	0.011	2.55	NA	NA	NA	(iii)

A. Reduced cycles used in the calculation.
B. Refined calculations performed.
C. Redefined transients used in the evaluation. Redefinition means that transient thermal-hydraulic definitions (e.g., temperature and pressure) are redefined.

Notes:

1. In accordance with Table 4.3-2, the RV outlet nozzle CUF is 0.46 which includes 30,000 power loading and unloading transients in excess of those permitted by the CR-3 design basis. Removing this conservatism results in a reduction of the RV outlet nozzle CUF from 0.46 to 0.31.
2. The full set of NSSS design transients are used with the exception of power loading and unloading. These were reduced from 48,000 to 2,600 based on a review of operating data. Transient regrouping was performed for heatups and cooldowns based on CR-3 operating experience. Transient 22 (HPI test) was revised based on revision to CR-3 procedures that eliminate the thermal transient. F_{en} was based on NUREG/CR-5704 considering transformed strain rate, transformed dissolved oxygen, and transformed metal service temperature.
3. Considers weld overlay and is the CUF at inside surface of original pipe adjacent to weld overlay.

4.4 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

4.4.1 10 CFR 50.49 THERMAL, RADIATION, AND CYCLICAL AGING ANALYSES

Thermal, radiation, and cyclical aging analyses of plant electrical and I&C components required to meet 10 CFR 50.49 have been identified as time-limited aging analyses (TLAAs) for CR-3.

Summary Description

The NRC has established nuclear station environmental qualification (EQ) requirements in 10 CFR 50, Appendix A, Criterion 4, and in 10 CFR 50.49. Section 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (that is, those areas of the plant that could be subject to the harsh environmental effects of a loss-of-coolant accident (LOCA), high energy line breaks (HELBs), or post-LOCA radiation) are qualified to perform their safety function in those harsh environments after the effects of in-service aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

<u>EQ Program Background</u>

The CR-3 EQ Program meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics and the environmental conditions to which the components could be subjected. Section 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. Section 50.49(e) also requires replacement or refurbishment of components not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. Section 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions.. Sections 50.49(k) and (l) permit different qualification criteria to apply based on plant and component vintage. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," July 1981; and RG 1.89, Rev. 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," June 1984. Compliance with 10 CFR 50.49 provides reasonable assurance that the component can perform its intended functions during accident conditions after experiencing the effects of in-service aging.

The CR-3 EQ Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for electrical components in the CR-3 EQ Program that specify a qualification of at least 40 years are TLAAs for license renewal because all of the criteria contained in 10 CFR 54.3 are met.

Under 10 CFR 54.21(c)(1)(iii), the CR-3 EQ Program, which implements the requirements of 10 CFR 50.49 (as further defined and clarified by DOR Guidelines, NUREG-0588, and RG 1.89, Rev. 1), is viewed as an aging management program for License Renewal. Reanalysis of an aging evaluation to extend the qualifications of components is performed on a routine basis as part of the CR-3 EQ Program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). TLAA demonstration option (iii), which states that the effects of aging will be adequately managed for the period of extended operation, is chosen; and the CR-3 EQ Program will manage the aging effects of the components associated with the environmental qualification TLAA. Section 4.4.2.1.3 of NUREG-1800 states that the staff evaluated the EQ program (10 CFR 50.49) and determined that it is an acceptable aging management program to address environmental qualification according to 10 CFR 54.21(c)(1)(iii). The evaluation referred to in the SRP-LR contains sections on "EQ Component Reanalysis Attributes" and "Evaluation and Technical Basis," which is the basis of the description provided below.

EQ Component Reanalysis Attributes

The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the CR-3 EQ Program. While a component life-limiting condition may be due to thermal, radiation or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed peak ambient temperature of the component, an activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to CR-3 quality assurance program requirements, which require the verification of assumptions and conclusions. As already noted, important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed below.

Analytical Methods

The CR-3 EQ Program uses the same analytical models in the reanalysis of an aging evaluation as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging a similar approach may be used.

Data Collection & Reduction Methods

Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis per the CR-3 EQ Program. Temperature data used in an aging evaluation should be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors. 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 must be justified. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.

Underlying Assumptions

CR-3 EQ Program component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Action

Under the CR-3 EQ Program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner (that is, sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).

The CR-3 EQ Program has been demonstrated to be capable of programmatically managing the qualified lives of the components falling within the scope of the program for License Renewal. Based on the above review, the continued implementation of the CR-3 EQ Program provides reasonable assurance that the aging effects will be managed and that EQ components will continue to perform their intended functions for the period of extended operation. This result meets the requirements of 10 CFR 54.21(c)(iii). A comparison of the CR-3 Environmental Qualification Program to the corresponding program in NUREG-1801 is provided in Appendix B, Subsection B.3.2.

Disposition: **10 CFR 54.21(c)(1)(iii) – Aging of components within the EQ Program will be adequately managed for the period of extended operation.**

4.5 CONCRETE CONTAINMENT TENDON PRESTRESS

4.5.1 TENDON STRESS RELAXATION ANALYSIS

Summary Description

The CR-3 Reactor Building consists of a prestressed reinforced concrete cylinder and hemispherical dome. The cylinder wall and dome roof have been provided with a post-tensioning system. The cylinder wall is prestressed utilizing a two-way post-tensioning system. The dome roof is prestressed utilizing a three-way post-tensioning system. The prestressing tendons tend to lose their prestressing forces with time due to creep and shrinkage of concrete and relaxation of the prestressing steel. Loss of tendon prestress is a TLAA; therefore, the adequacy of the prestressing forces is reviewed for the period of extended operation.

There have been eight tendon surveillance tests since CR-3 plant startup in December 1976. Since 1997, these tests have been performed under the ASME Section XI, Subsection IWL Program. The IWL program inspects a sample of tendons from each category (i.e., dome, vertical, and hoop). The program calculates the regression analysis trend lines of these three groups based on individual tendon forces consistent with NRC Information Notice 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," that is, using individual-tendon data rather than averages and using all prior test data. It confirms that the acceptance criteria have been met and, therefore, that tendon prestresses will remain above minimum required values for the succeeding inspection interval.

Analysis

For the purposes of extending the CR-3 plant operating license, regression analysis was used to extrapolate the tendon prestress forces to the end of the extended period of operation. Figures 4.5-1, 4.5-3, and 4.5-5 illustrate the overall results of the regression analysis for the three groups of tendons. Figures 4.5-2, 4.5-4, and 4.5-6 show the results for the individual control tendons.

The resulting trend line for control tendon 61V08 shown in Figure 4.5-4 has not been projected forward to the end of the extended period of operation. The original vertical control tendon, 12V01, required retensioning in the 7th interval surveillance. Tendon 61V08 was selected as the new control tendon for subsequent surveillances. As a result, only two data points are available for this tendon, from surveillance intervals 7 and 8. The measurement from surveillance interval 8 was 2% higher than that for surveillance interval 7, a difference that is within the overall accuracy of the testing system. Since the trend line generated from the two points has a positive slope, more data from future surveillances will be needed before an accurate trend line for tendon 61V08 can be established.

The values computed demonstrated that prestress in all three groups of tendons should remain above the applicable minimum required values for the extended period of operation and that the tendons should maintain their design basis function. The following tables and figures document the results of these analyses:

The TLAA evaluation addressed tendon loss of preload, using 10 CFR 54.21(c)(1)(ii) to project the tendon preload to the end of the 60-year service period for each group of tendons. The projected "average" preload values at the end of the 60-year service period are then compared with the required minimum average tendon preload. For each group of tendons, the projected preload value exceeds the required minimum average tendon preload. Therefore, prestress in all three groups of tendons will remain above the applicable minimum required values for the period of extended operation; and the tendons will perform their intended function.

Disposition: **10 CFR 54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation; and**

TABLE 4.5-1 SUMMARY OF TENDON DATA

Tendon Type	Total Number of Tendons	Minimum Required Average Values (Kips/Tendon)	Value Extrapolated to End of Period of Extended Operation (Kips/Tendon)	Conclusion
Dome	123	1215	1255	Note 1
Vertical	144	1149	1478	Note 1
Hoop	282	1252	1329	Note 1

Note:

1. The value at the end of the period of extended operation is greater than the minimum required value.

TABLE 4.5-2 DOME TENDON DATA

Surveillance No.	Tendon	Years Since Initial Tensioning	Measured Force/ Tendon	Computed Value
1	D139	3.22	1590	1553
	D215	3.25	1644	1553
	D221	3.14	1511	1555
	D228	3.11	1524	1556
	D234	3.1	1513	1556
	D340	3.11	1562	1556
2	D122	5.41	1647	1513
	D140	5.41	1587	1513
	D208	5.4	1594	1513
	D323	5.47	1526	1512
	D331	5.5	1461	1512
3	D123	6.99	1304	1493
	D212*	6.9	1338	1494
	D215	6.98	1594	1494 (Note 1)
	D322	6.91	1494	1494
	D329	6.92	1506	1494
4	D105	12.92	1453	1445
	D212*	12.9	1276	1445
	D328	12.89	1619	1445
5	D215	19.01	1518	1415 (Note 1)
	D242	18.96	1425	1415
	D231	19.06	1335	1414
6	D113	22.99	1427	1400
	D115	22.7	1380	1401
	D212*	22.9	1335	1400
	D304	23	1598	1400
	D311	23.03	1408	1400

TABLE 4.5-2 (continued) DOME TENDON DATA

Surveillance No.	Tendon	Years Since Initial Tensioning	Measured Force/ Tendon	Computed Value
7	D126	26.81	1377	1388
	D212*	26.6	1292	1388
	D339	26.47	1507	1389
8	D129	32.96	1289	1372
	D212*	32.92	1277	1372
	D238	32.89	1511	1372
Extrapolated		63 (Note 2)		1321

* Indicates Control Tendon

Notes:

1. This value was not used in the regression analysis as the tendon was retensioned in an earlier surveillance.
2. The extended period of operation will end in the 63rd year from the date of initial tensioning.

TABLE 4.5-3 VERTICAL TENDON DATA

Surveillance No.	Tendon	Years Since Initial Tensioning	Measured Force/ Tendon	Computed Value
1	12V19	3.26	1589.5	1671
	12V20	2.87	1785	1680
	12V21	3.26	1632.5	1671
	23V15	3.22	1590	1672
	34V06	3.16	1678	1673
	45V03	3.19	1678	1673
	56V01	3.28	1718.5	1671
2	12V12	5.47	1718	1638
	12V20	5.2	1740	1641
	23V05	5.57	1580	1637
	34V01	5.54	1569	1637
	45V06	5.53	1685	1637
	56V01	5.59	1707	1636
	56V20	5.48	1630	1638
3	12V01*	7.07	1315	1621 (Note 1)
	34V06	6.99	1600	1622 (Note 2)
	34V19	7.02	1640	1622
	45V16	7.01	1575	1622
	56V11	7.04	1565	1621
	61V05	7.07	1519	1621
4	12V01*	13.07	1535	1581
	34V04	13.07	1623	1581
	56V02	13.04	1648	1581
5	34V06	19.08	1590	1557
	61V14	19.08	1587	1557
	56V15	19.18	1541	1556
6	12V01*	23.15	1471	1544
	23V02	23.15	1609	1544
	61V21	23.25	1525	1544

TABLE 4.5-3 (continued) VERTICAL TENDON DATA

Surveillance No.	Tendon	Years Since Initial Tensioning	Measured Force/ Tendon	Computed Value
7	12V01*	27.028	1446	1534
	12V02	27.12	1546	1534
	23V24	27.14	1521.8	1534
	45V14	26.94	1552	1534
	61V08*	26.91	1476	1534
8	12V01	33.14	1559.95	1521 (Note 2)
	45V20	33.03	1456.8	1521
	61V08*	33.05	1505.98	1521
	61V17	33.11	1580.18	1521
Extrapolated		63 (Note 3)		1478

*Indicates control tendon -The original control tendon, 12V01, required retensioning in the 7th interval surveillance. Tendon 61V08 was selected as the new control tendon for subsequent surveillances.

Notes:

1. Data for this tendon for this surveillance was considered erroneous and not used in constructing the trend line. The force value was substantially lower than the value from subsequent surveillance tests.
2. This value was not used in the regression analysis as the tendon was retensioned in an earlier surveillance.
3. The extended period of operation will end in the 63rd year from the date of initial tensioning.

TABLE 4.5-4 HOOP TENDON DATA

Surveillance No.	Tendon	Years Since Initial Tensioning	Measured Force/ Tendon	Computed Value
1	13H10	3.17	1524	1549
	13H19	3.17	1485	1549
	13H37	3.17	1606	1549
	13H47	3.17	1606	1549
	62H9	3.17	1574	1549
	46H21*	3.17	1502	1549
	46H29	3.17	1463	1549
	46H31	3.17	1457	1549
	46H46	3.17	1464	1549
	51H11	3.17	1474	1549
2	13H22	5.46	1572	1508
	13H32	5.46	1611	1508
	13H43	5.46	1583	1508
	35H24	5.46	1533	1508
	35H28	5.46	1430	1508
	35H44	5.46	1622	1508
	46H42	5.46	1548	1508
	51H10	5.46	1572	1508
	51H23	5.46	1528	1508
	51H37	5.46	1567	1508
3	13H19	6.739	1424	1493 (Note 1)
	13H46	6.592	1546	1495
	35H35	6.786	1328	1493
	35H40	6.608	1458	1495
	42H20	6.614	1544	1494
	42H40	6.636	1466	1494
	46H10	6.678	1478	1494
	51H26*	6.569	1424	1495
	51H45	6.792	1492	1492
	62H34	6.617	1546	1494
4	13H20	12.575	1456	1447
	13H40	12.606	1471	1447
	51H26*	12.608	1411	1447
	51H41	12.817	1362	1446
	64H19	12.728	1470	1446

TABLE 4.5-4 (continued) HOOP TENDON DATA

Surveillance No.	Tendon	Years since Initial Tensioning	Measured Force/ Tendon	Extrapolated Value
5	35H01	18.942	1572	1417
	42H01	18.928	1560	1417
	46H21*	18.833	1425	1417 (Note 1)
	46H28	18.728	1375	1418
	46H29	18.769	1300	1418 (Note 1)
	46H30	18.781	1382	1418
	46H47	18.817	1468	1418
	62H08	18.814	1435	1418
6	42H18	22.692	1476	1404
	42H29	22.786	1448	1403
	42H30	22.733	1389	1404
	42H31	22.775	1338	1404
	42H32	22.703	1356	1404
	42H33	22.772	1361	1404
	42H35	22.733	1296.5	1404
	42H36	22.781	1408	1403
	42H37	22.744	1401.5	1404
	42H44	22.711	1471.5	1404
	51H25	22.822	1363	1403
	51H26*	22.628	1320	1404
	51H27	22.836	1265.5	1403
	51H28	22.647	1450.5	1404
	53H02	22.797	1611	1403
	53H46	22.667	1560	1404
	62H41	22.797	1426	1403
	62H46	22.736	1485	1404
7	46H21*	26.656	1388	1392
	46H30	26.694	1356	1392
	46H31	26.667	1343	1392
	46H32	26.628	1367	1392
	46H33	26.664	1358	1392
	46H34	26.619	1425	1392
	46H35	26.653	1377	1392
	46H36	26.608	1344	1392
	46H37	26.644	1293	1392

TABLE 4.5-4 (continued) HOOP TENDON DATA

Surveillance No.	Tendon	Years since Initial Tensioning	Measured Force/ Tendon	Extrapolated Value
7 (continued)	46H38	26.625	1353	1392
	46H39	26.647	1356	1392
	53H16	26.628	1475	1392
	63H02	26.672	1552	1392
	63H09	26.814	1432	1391
8	13H33	32.817	1306	1377
	13H34	32.636	1368	1377
	13H35	32.814	1244	1377
	13H36	32.622	1385	1377
	13H37	32.825	1289	1377
	13H38	32.639	1395	1377
	42H46	32.692	1558	1377
	46H19	32.711	1358	1377
	46H20	32.619	1298	1377
	46H21*	32.694	1330	1377
	46H22	32.622	1311	1377
	46H23	32.708	1329	1377
	46H24	32.636	1425	1377
	51H34	32.644	1464	1377
	62H29	32.736	1369	1377
	62H30	32.681	1290	1377
	62H31	32.739	1269	1377 (Note 2)
	62H32	32.689	1332	1377 (Note 2)
	62H33	32.733	1313	1377
	62H34	32.686	1378	1377
Extrapolated		63 (Note 3)		1329

*Control Tendon - Tendon 51H26 was used as the control tendon when testing was performed during outages for surveillances 3, 4 and 6. Tendon 46H21 was used during online testing for surveillances 5, 7 and 8.

Notes:

1. This value was not used in the regression analysis as the tendon was retensioned in an earlier surveillance.
2. This value was not used in the regression analysis as the tendon was only tested on one end.
3. The extended period of operation will end in the 63rd year from the date of initial tensioning.

FIGURE 4.5-1 PROJECTED FORCE IN DOME TENDONS

Based on Regression Analysis Using Surveillance Data

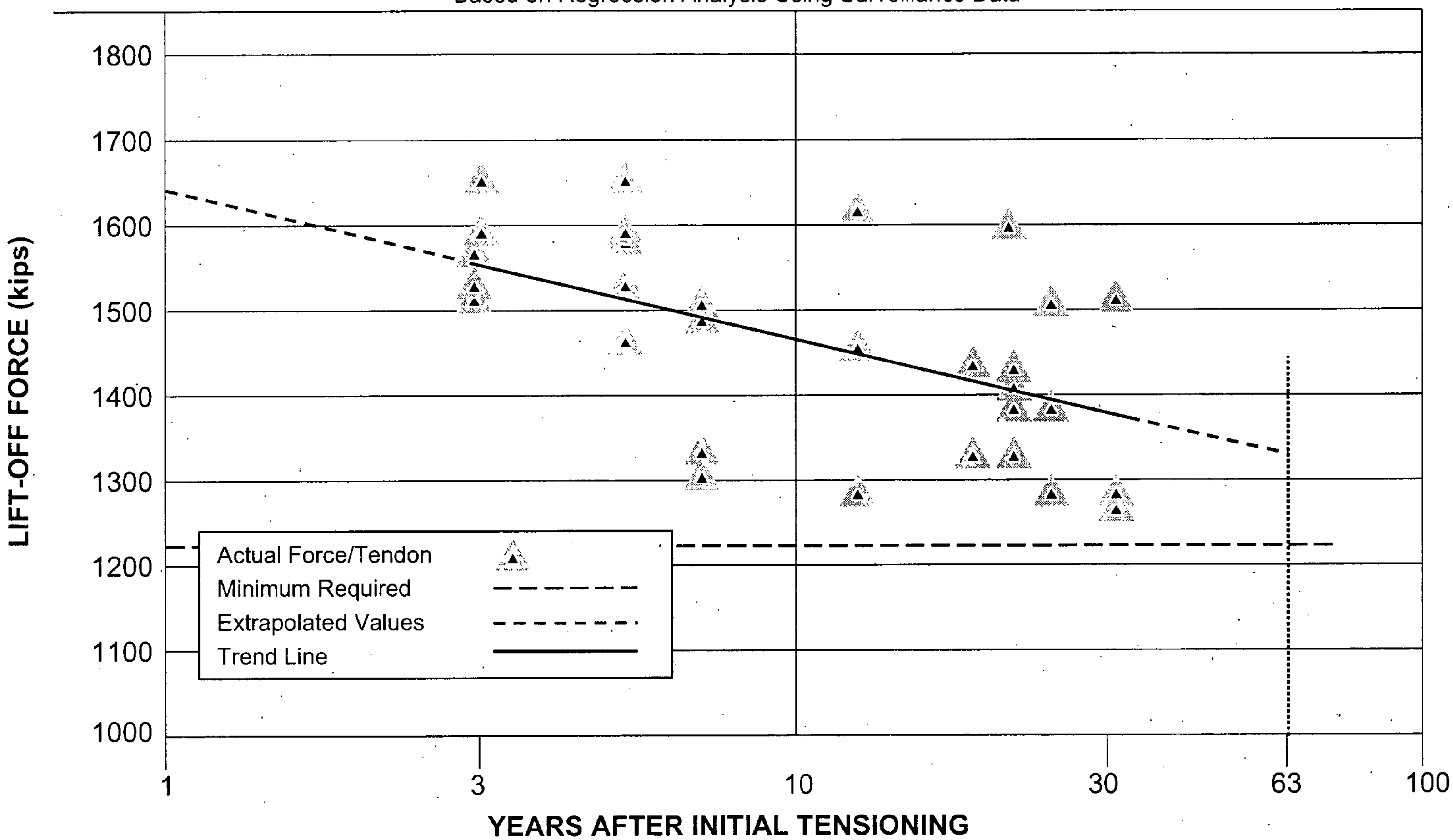


FIGURE 4.5-2 PROJECTED FORCE IN DOME CONTROL TENDON D212

Based on Regression Analysis Using Surveillance Data

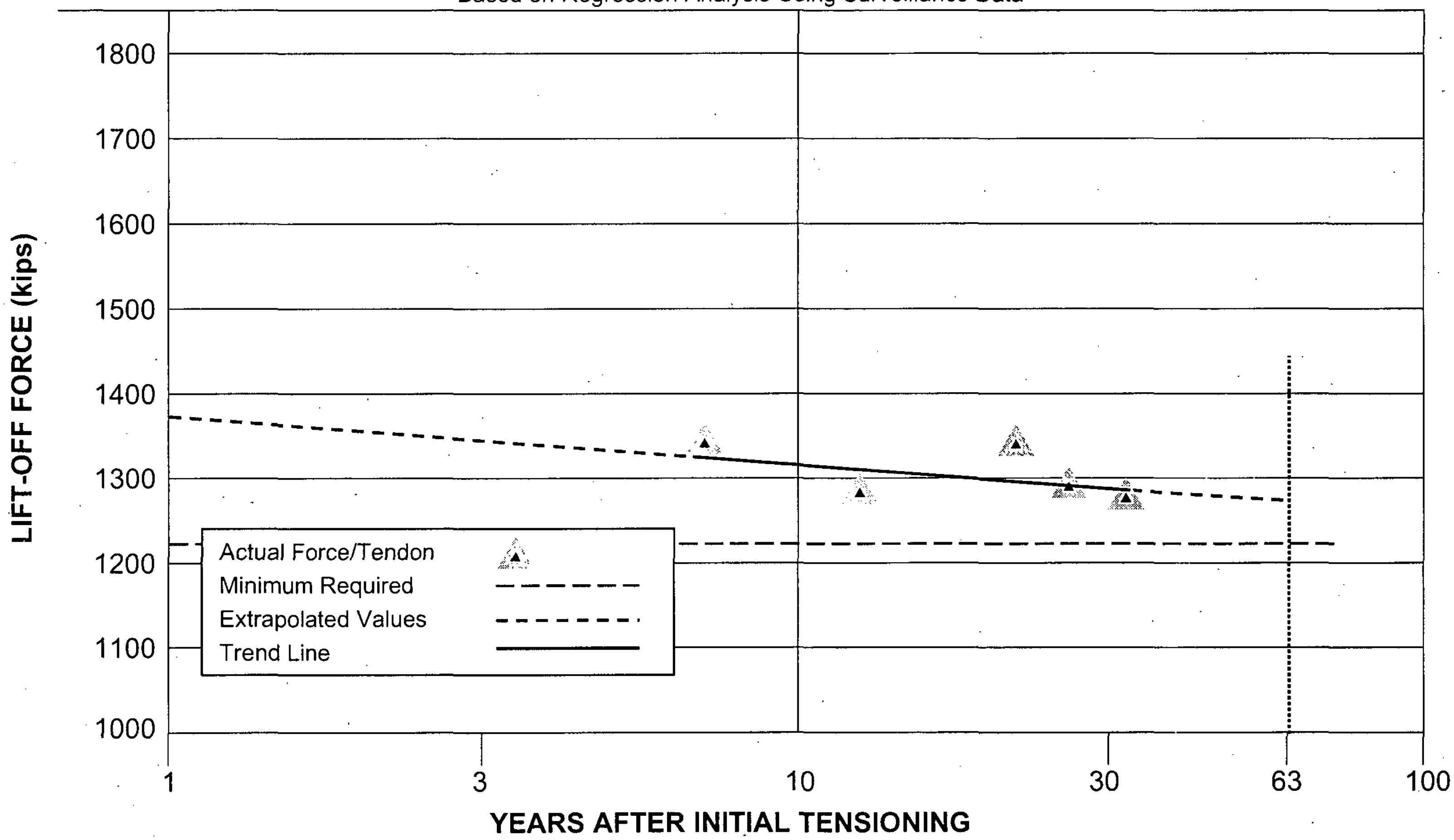


FIGURE 4.5-3 PROJECTED FORCE IN VERTICAL TENDONS

Based on Regression Analysis Using Surveillance Data

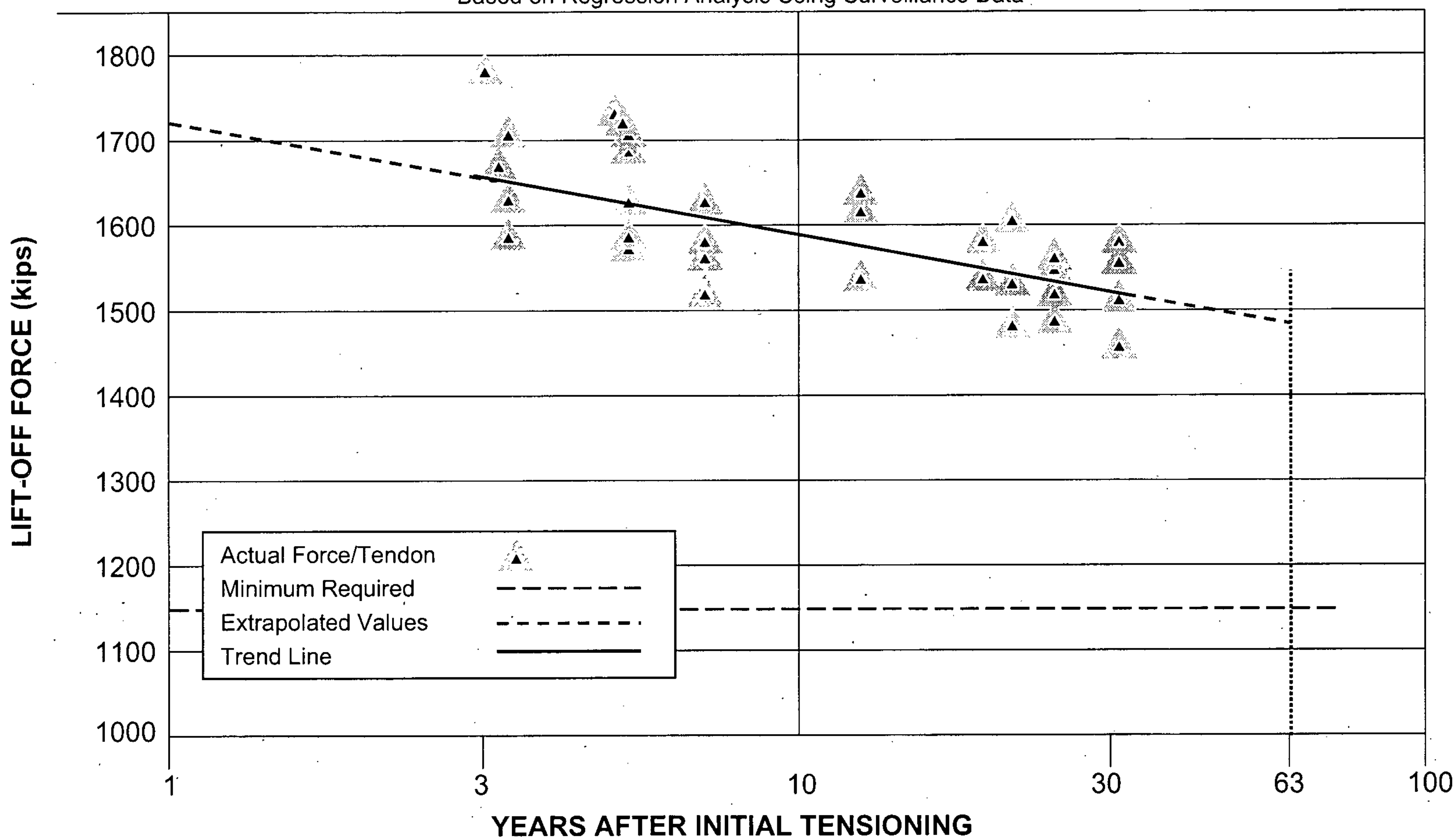


FIGURE 4.5-4 PROJECTED FORCE IN VERTICAL CONTROL TENDONS 61V08 AND 12V01

Based on Regression Analysis Using Surveillance Data

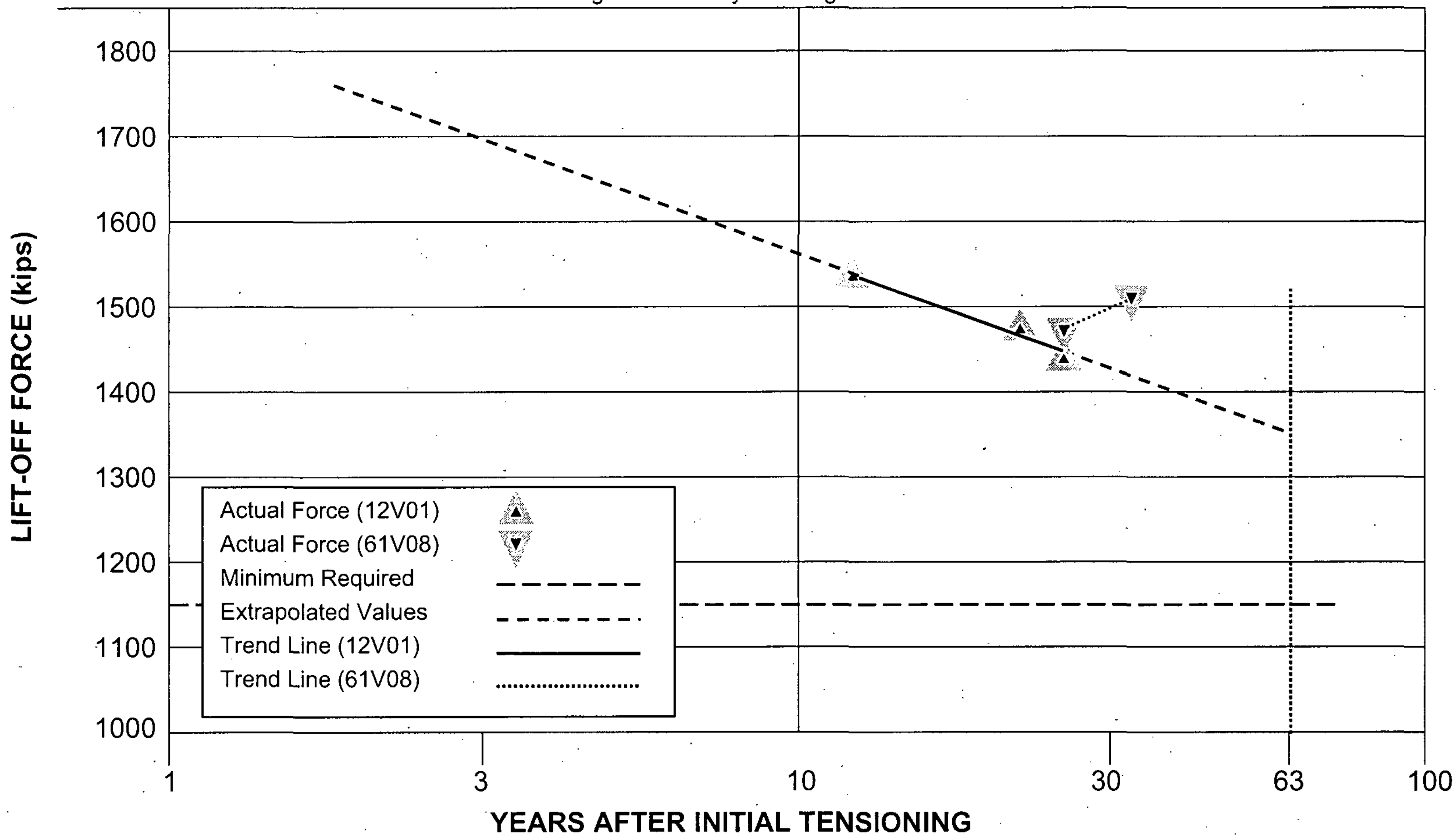


FIGURE 4.5-5 PROJECTED FORCE IN HOOP TENDONS

Based on Regression Analysis Using Surveillance Data

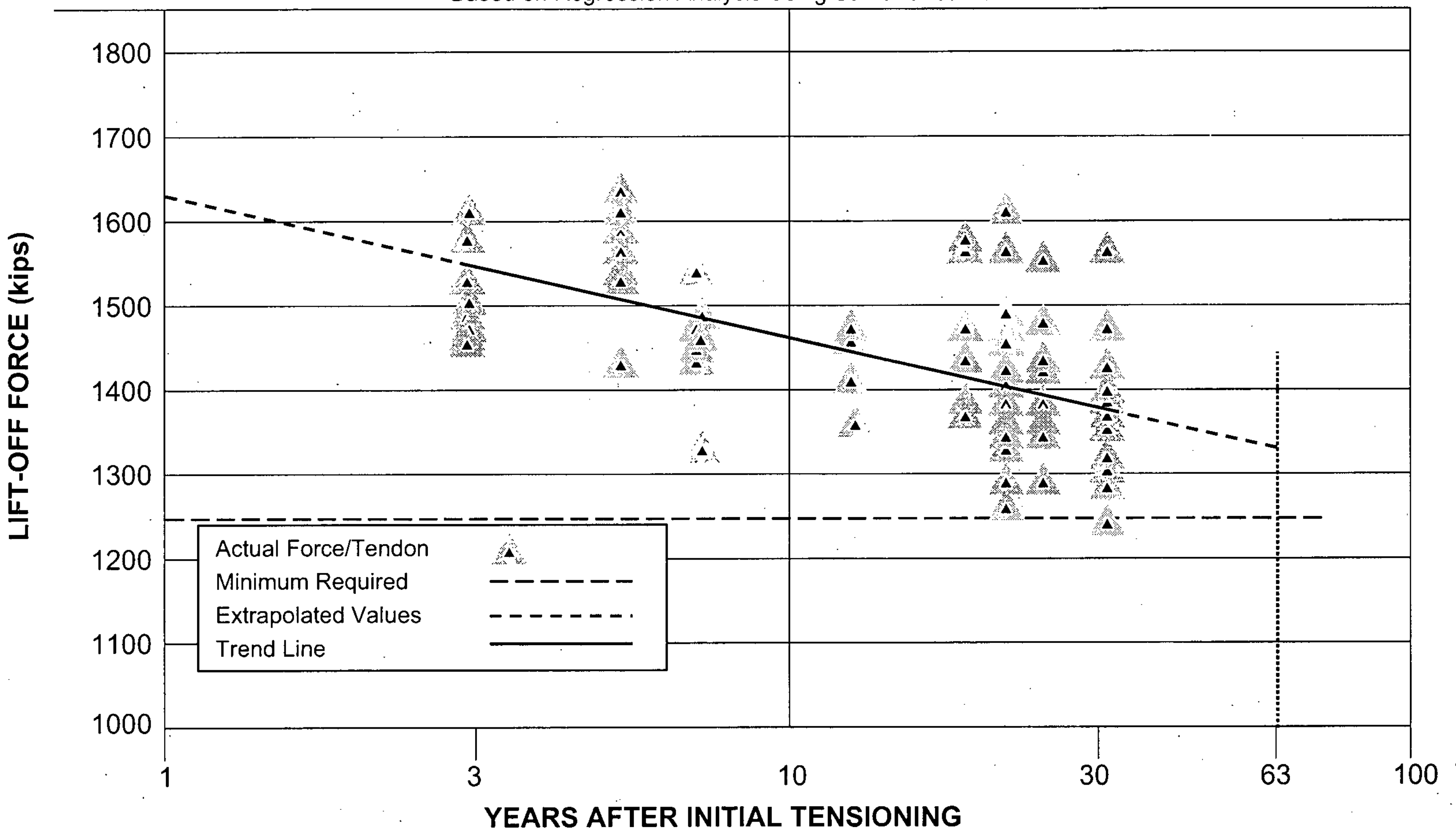
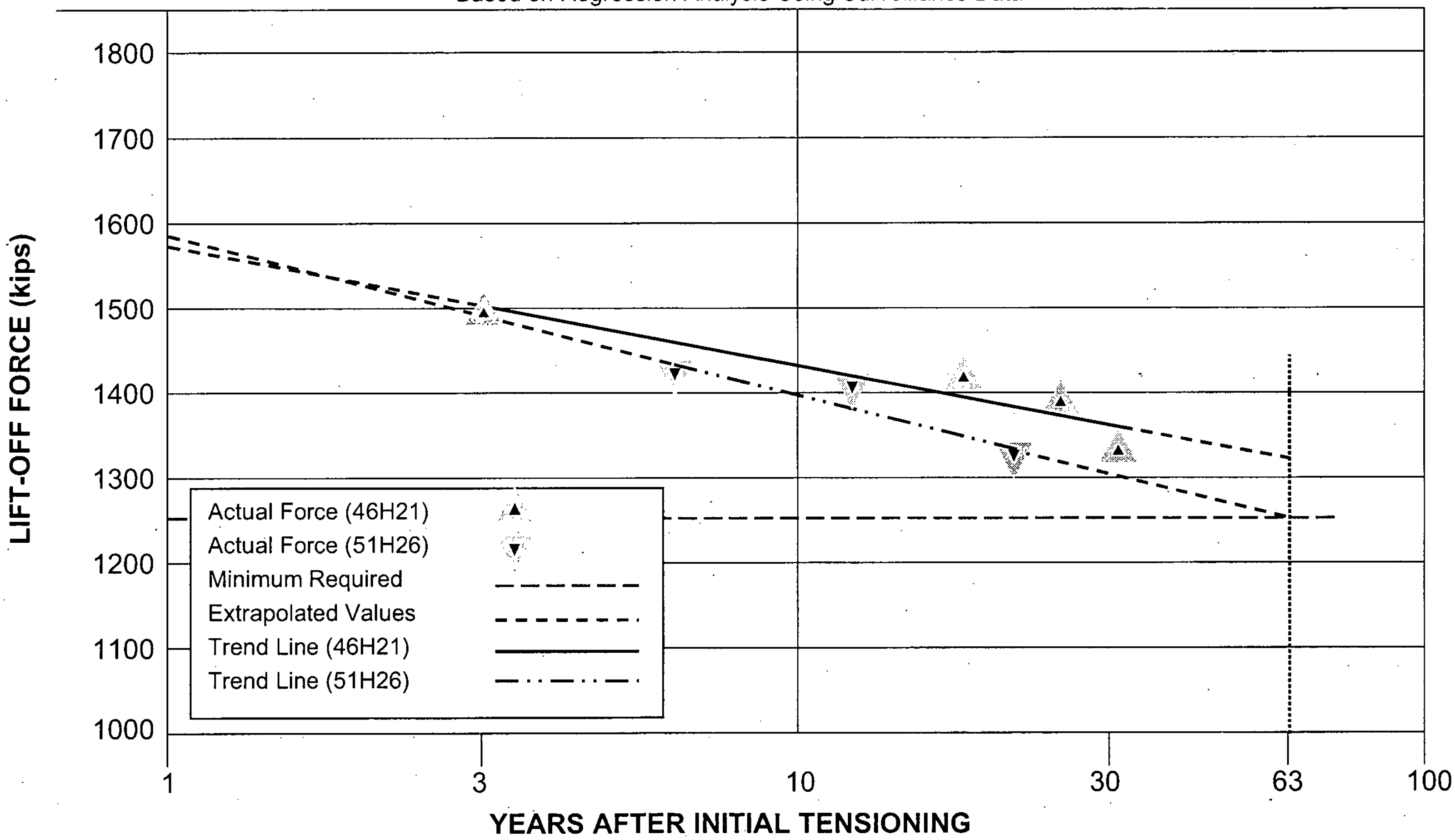


FIGURE 4.5-6 PROJECTED FORCE IN HOOP CONTROL TENDONS 51H26 AND 46H21

Based on Regression Analysis Using Surveillance Data



4.6 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

4.6.1 FUEL TRANSFER TUBE EXPANSION BELLOWS CYCLES

Summary Description

The Fuel Transfer Tubes are essentially tubular passageways connecting the transfer canal in the Reactor Building with the Spent Fuel Pool in the Auxiliary Building. The Fuel Transfer Tube Expansion Bellows connect the Fuel Transfer Tubes to the Refueling Canal in the Reactor Building and to the Spent Fuel Pool in the Auxiliary Building. Per plant specifications, the Expansion Bellows shall be fabricated, as a minimum, to the requirements of Section VIII of the ASME Code and shall be inspected in accordance with the requirements of ASME Code, Section III, Class B vessels. Each Expansion Bellows is designed to withstand a total of 5,000 cycles of expansion and compression over a lifetime of 40 years. This TLAA addresses the requirement to ensure that the lifetime as described above may be extended to 60 years without exceeding the design criterion of 5,000 cycles.

Analysis

Expansion bellows cycles occur each refueling outage due to thermal cycling when the Fuel Transfer Tubes are flooded with refueling water then drained for return of the plant to operation. Assuming a period of mid-loop operation that involves a partial drain and refilling of the canal, bellows cycling would occur twice every refueling outage; however, cycling has been assumed to occur three times every refueling outage for additional conservatism. The number of cycles applied to the expansion bellows in the Reactor Building is assumed also to apply to the expansion bellows in the Auxiliary Building. There are 19 refueling outages planned for the 40-year life of the plant. The number of refueling outages over 60 years of life is 60/40 x 19 = 28.5 or 29 refueling outages. The maximum number of operating cycles projected to be experienced over the 29 refueling outages during a 60-year period is:

29 refueling outages x 3 cycles/refueling outage = 87 cycles.

Since the total number of expansion and compression cycles for the Fuel Transfer Tube Expansion Bellows is less than 5,000 cycles, no reanalysis of the design calculations is necessary. Therefore, an evaluation was performed as required by 10 CFR 54.21(c)(1) and was successful in demonstrating that the Fuel Transfer Tube Expansion Bellows design analyses of record remain valid for the period of extended operation.

Disposition: **10 CFR 54.21(c)(1)(i) – The qualification analyses for the Fuel Transfer Tube Expansion Bellows remain valid for the period of extended operation.**

4.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

4.7.1 ANALYSIS OF BEDROCK DISSOLUTION FROM GROUNDWATER

Summary Description

FSAR Section 2.5.3.4 documented a Bedrock Solutioning Study at CR-3. The solutioning process is the result of fresh water entering the underground areas below the plant and attacking the limestone sediments causing a destructive alteration of the carbonate rock leaving a labyrinth of channels throughout the rock mass. The purpose of the study was to determine the rate at which this solutioning process takes place and to establish the effect such a deleterious process would have upon the foundation of the CR-3 power plant during its 40-year life. The percent of rock dissolved over the 40-year life of the plant was calculated using different methods; and a determination was made that the percent of rock dissolved in accordance with these analyses represents an insignificant amount and that the small percentages of bedrock solutioning remain insignificant to the stability of the rock mass.

Analysis

One method presented in the FSAR determined that 1.5×10^{-5} % of the bedrock was dissolved over the forty year life of the plant. This was based on assuming the law of uniformitarianism was applicable and that 15% of the rock mass has been dissolved in 40 million years and definitely in more than 40,000 years. The 15% was based on the results of the exploratory and grout hole drilling at the site which indicated that the volume of solution channels was probably not greater than 15%. On this basis, the solution rate of the limestone was determined to be 15% per 40,000,000 years or approximately 3.75×10^{-7} % per year. In the 40-year life of the plant, 1.5×10^{-5} % could be expected to be dissolved. To extend this value to 60 years, the total maximum volume of dissolved bedrock was multiplied by the ratio of 60years/40years for an additional 20 years of extended life. Thus,

$$1.5 \times 10^{-5}\ \% \times 60/40 = 2.25 \times 10^{-5}\ \%.$$

In addition, the FSAR considered an extreme case by assuming that all of the 40-year solutioning has occurred during the last 10,000 years after the base level of the limestone formation was established as it essentially is today. This calculation produces the maximum solution rate. Assuming that only 10,000 years have been required for 15% of the rock mass to dissolve, the solution rate is 1.5×10^{-3} % per year. In a 40-year life of the plant, 6.0×10^{-2} % of the total volume would be dissolved. The FSAR determined that such a small percentage of solutioning would still be insignificant to the stability of the rock mass. To determine the percent dissolved during a 60-year plant life, this value was multiplied by the 60/40 ratio. Thus,

$$6.0 \times 10^{-2}\ \% \times 60/40 = 9.0 \times 10^{-2}\ \%.$$

It should be noted that this extreme case of reasoning for determining the percent of the rock dissolved at 40 years of plant life was not used in the conclusion for FSAR Section 2.5.3.4; and, therefore, the projection for this case was not used in this analysis.

Another method of evaluating bedrock dissolution is provided in the FSAR. This method determined that 4×10^{-3} % of the bedrock would be dissolved over the 40-year life of the plant. This was based on information obtained from the U.S. Geologic Survey for dissolved solids over a large land area that included the CR-3 site. Using this information it was determined that 764 lbs/day/mi^2 was dissolved. Comparing this to the actual area of the power plant resulted in 6.3 lbs per day of dissolved solids daily beneath the plant. This, in turn, results in 23 ft^3 of limestone per year dissolved from 23,040,000 ft^3 of rock based on limestone density of 100 lbs/ft^3 and assuming the solutioning occurs in the first 100 feet of depth beneath the ground surface. The conclusion of this analysis was that the solution rate was 1×10^{-4} % per year or 4×10^{-3} % for 40 years. For an additional 20 years of extended life, the total maximum volume of dissolved bedrock was determined by multiplying by 60/40:

$$4 \times 10^{-3}\ \% \times 60/40 = 6 \times 10^{-3}\ \%.$$

The conclusions of the 60-year projections based on the methods presented in FSAR Section 2.5.3.4 are that the range in percent of the rock dissolved would be between 2.25×10^{-5} % and 6×10^{-3} %. Dissolved volumes calculated by any of these methods still represent insignificant amounts. Further, the grouting process used in the foundation of Crystal River Units 2 and 3 reduced the permeability of the carbonate rocks from a figure in excess of 65,500 feet per year to less than 2,000 feet per year. With the permeability decreased by more than 30 times, exposure of the limestone to potential solvent groundwater is effectively reduced by the same factor. It is concluded that the natural solution process will not affect the structural integrity of the foundation of the power plant for the period of extended operation. Therefore, the analysis of the volume of bedrock solutioning from FSAR Section 2.5.3.4 has been projected to the end of the period of extended operation.

Disposition: **10 CFR 54.21(c)(1)(ii) – The analysis has been projected to the end of the period of extended operation.**

APPENDIX A

FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

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A.0 FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

This appendix provides the information to be submitted in a Final Safety Analysis Report Supplement as required by 10 CFR 54.21(d) for the CR-3 License Renewal Application. The License Renewal Application contains the technical information required by 10 CFR 54.21(a) and (c). Chapter 3 of the CR-3 License Renewal Application identifies the programs and activities that manage the effects of aging for the proposed period of extended operation, and Appendix B describes the programs and activities. Chapter 4 contains the evaluations of time-limited aging analyses for the period of extended operation. License Renewal Application Chapters 3 and 4 and Appendix B have been used to prepare the program and activity descriptions that are contained in this Appendix. The information presented here will be incorporated into the CR-3 FSAR following issuance of the renewed operating license.

A.1 NEW FSAR SECTION

The following information will be integrated into the FSAR to document aging management programs and activities credited in the CR-3 License Renewal review and time-limited aging analyses evaluated to demonstrate acceptability during the period of extended operation.

A.1.1 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment and evaluation of time-limited aging analyses (TLAA) 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 (CLB) for the period of extended operation. This section describes the programs and their implementation activities.

Three elements common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the CR-3 Quality Assurance (QA) Program, which implements the requirements of 10 CFR 50, Appendix B.

In accordance with the guidance of NUREG-1801, "Generic Aging Lessons Learned (GALL)," Rev. 1, U.S. Nuclear Regulatory Commission, September 2005, regarding aging management of reactor vessel internals components for aging mechanisms, such as embrittlement and void swelling, CR-3 will: (1) participate in the industry programs for investigating and managing aging effects on reactor internals, (2) evaluate and implement the results of the industry programs as applicable to the reactor internals, and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

In accordance with the guidance of NUREG-1801, regarding activities for managing the aging of nickel alloy and nickel-clad components susceptible to primary water stress corrosion cracking, CR-3 will comply with applicable NRC Orders and will implement: (1) applicable Bulletins and Generic letters, and (2) staff-accepted industry guidelines.

A.1.1.1 ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program

The American Society of Mechanical Engineers (ASME) Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program is an existing program that consists of periodic volumetric, surface, and/or visual examinations of components for assessment, signs of degradation, and corrective actions. The Program for the Fourth

10-Year Inservice Inspection (ISI) interval at CR-3 will be implemented in accordance with Section XI of the ASME B&PV Code, 2001 Edition with addenda through 2003.

A.1.1.2 Water Chemistry Program

To mitigate aging effects on component surfaces that are exposed to water as a process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion and cracking. The CR-3 Water Chemistry Program is an existing program that relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. Alternatively, chemical agents, such as corrosion inhibitors, oxygen scavengers, and biocides, may be introduced to prevent certain aging mechanisms. The CR-3 Water Chemistry Program is currently based on the latest version of the Electric Power Research Institute (EPRI) pressurized water reactor guidelines, "Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1 and 2," and "Pressurized Water Reactor Secondary Water Chemistry Guidelines." The CR-3 Water Chemistry Program is updated as revisions to the guidelines are released.

A.1.1.3 Reactor Head Closure Studs Program

The CR-3 Reactor Head Closure Studs Program is an existing condition monitoring program which is implemented primarily through the CR-3 ASME Section XI Inservice Inspection Program. In addition, the Program includes certain preventive measures recommended by Regulatory Guide 1.65, "Material and Inspection for Reactor Vessel Closure Studs." This Program is credited for aging management of the Reactor Vessel Closure Head Stud Assembly (i.e., closure studs, nuts, and washers) for cracking due to stress corrosion cracking and loss of material due to wear.

Prior to the period of extended operation, the Reactor Head Closure Studs Program will be enhanced to select an alternate lubricant that is compatible with the fastener material and the contained fluid.

A.1.1.4 Boric Acid Corrosion Program

The Boric Acid Corrosion Program is an existing program that manages the aging effects for susceptible materials of structures and components that perform a License Renewal intended function and that are exposed to the effects of borated water leaks. The Program consists of: (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy of corrective actions. This Program is implemented in response to NRC Generic Letter 88-05.

The scope of the Boric Acid Corrosion Program includes components that may be susceptible to exposure to boric acid including mechanical, structural, and electrical components. The Boric Acid Corrosion Program includes plant-specific reactor coolant pressure boundary (RCPB) boric acid leakage identification and inspection procedures to ensure that leaking borated coolant does not lead to degradation of the leakage source or adjacent structures, and provides assurance that the RCPB will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

A.1.1.5 Nickel-alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure heads of Pressurized Water Reactors Program

The Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program is an existing program that provides for the periodic inspection of the Reactor Pressure Vessel head and Vessel Head Penetration nozzles. This Program effectively manages the aging effect by identifying cracking in the upper penetration nozzles or the J-groove welds prior to loss of intended function. The required inspections are performed in the CR-3 ISI Program as augmented inspections.

A.1.1.6 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program that will manage loss of fracture toughness due to thermal aging and/or neutron irradiation embrittlement of CASS reactor vessel internals. This Program will be based upon the evaluation and inspection recommendations of NUREG-1801, Section XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)." The Program will effectively manage the aging effect to prevent loss of intended function.

A.1.1.7 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion (FAC) Program is an existing program that provides for prediction, inspection, and monitoring of carbon steel piping, valves, and fittings for loss of material due to FAC so that timely and appropriate action may be taken to minimize the probability of experiencing a FAC-induced leak or rupture. The FAC Program elements are based on the recommendations identified in NSAC-202L, “Recommendations for an Effective Flow-Accelerated Corrosion Program,” which require controls to assure the structural integrity of carbon steel lines containing high-energy fluids, both two-phase, as well as single-phase.

A.1.1.8 Bolting Integrity Program

The Bolting Integrity Program is an existing program that addresses aging management requirements for bolting on mechanical components within the scope of License Renewal. The CR-3 Bolting Integrity Program utilizes industry recommendations and

EPRI guidance that considers material properties, joint and gasket design, chemical control, service requirements, and industry and site operating experience in specifying torque and closure requirements. The Program relies on recommendations for a Bolting Integrity Program, as delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," and industry recommendations, as delineated in EPRI reports NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and TR-104213, "Bolted Joint Maintenance & Applications Guide," for pressure retaining bolting within the scope of License Renewal. Bolting and closures inspections, monitoring and trending, and repair and replacement are performed under the ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program and External Surfaces Monitoring Program requirements, as applicable. The Program includes periodic inspection of high-strength structural bolting for cracking. Degraded conditions are also subject to the Corrective Action Program. The Structures Monitoring Program and the ASME Section XI Inservice Inspection, Subsection IWF Program are credited for aging management of structural bolting.

Prior to the period of extended operation, Program administrative control documents will be enhanced to include: (1) guidance for torquing and closure requirements based on the EPRI documents endorsed by NUREG-1801, (2) requirements to remove instances where molybdenum disulfide lubricant is allowed for use in bolting applications in specific procedures and to add a general prohibition against use of molybdenum disulfide lubricants for bolted connections, (3) guidance for torquing and closure requirements that include proper torquing of the bolts and checking for uniformity of gasket compression after assembly, (4) guidance for torquing and closure requirements based on the guidance of EPRI 5067, "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Personnel," Volumes I and II, (5) a centralized procedure based on EPRI-5067 containing guidance regarding bolted joint leak tightness and pre-installation inspections consistent with the recommendations of that document, (6) periodic examinations of a representative sample of bolting identified as potentially having actual yield strength >150 ksi for SCC consisting of periodic in situ ultrasonic testing or, alternatively, surface examination or bolt replacement, (7) examination of NSSS support high strength bolting for SCC concurrent with examinations of the associated supports at least once per 10-year ISI period, and (8) acceptance standards for examination of high strength structural bolting consistent with the recommendations of EPRI NP-5769 or application specific structural analyses.

A.1.1.9 Steam Generator Tube Integrity Program

The Steam Generator Tube Integrity Program is an existing program credited for aging management of the tubes, tube plugs, sleeves, tube supports, and the secondary-side components whose failure could prevent the steam generator from fulfilling its intended safety function for the period of extended operation. The Steam Generator Tube Integrity Program is based on an existing program, the Steam Generator Integrity Program that has been established to meet the intent of the Steam Generator Program guidance in Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program

Guidelines," Revision 2. The Steam Generator Integrity Program is based on Technical Specification requirements and NEI 97-06.

A.1.1.10 Open-Cycle Cooling Water System Program

The Open Cycle Cooling Water System Program is an existing program that addresses the aging effects of material loss, flow blockage, and reduction in heat transfer due to micro- or macro-organisms and various corrosion mechanisms in raw water piping systems. This Program was originally developed in response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The Program includes monitoring, inspecting, and testing to verify that Nuclear Services and Decay Heat Seawater System aging effects can be managed and that the system can perform its intended safety related functions.

The Program will be enhanced to: (1) include the Nuclear Services and Decay Heat Seawater System Pumps in a periodic inspection and/or rebuild program. This Program will be initiated during the current license period and inspect one or more pumps prior to the period of extended operation, (2) subject the Nuclear Services and Decay Heat Seawater System Discharge Conduits to inspection and evaluation subsequent to the SG replacement project, but prior to the period of extended operation, in order to determine the extent of activities required during the period of extended operation to support the intended function of these components, and (3) establish periodic maintenance activities for Nuclear Services and Decay Heat Seawater System expansion joints prior to the period of extended operation.

A.1.1.11 Closed-Cycle Cooling Water System Program

The Closed-Cycle Cooling Water System Program is an existing program that addresses aging management of components in CR-3 conventional closed-cycle cooling water systems, diesel engine jacket water systems, and chilled water systems. These systems are closed cooling loops with controlled chemistry, consistent with the NUREG-1801 description of a closed-cycle cooling water system. These systems employ an effective chemistry program augmented by component testing and inspection based on EPRI Closed Cooling Water Chemistry Guidelines to assure License Renewal intended functions are maintained.

A.1.1.12 Inspection of Overhead Heavy Load and Light Load Handling Systems Program

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program is an existing program that manages the aging effects of corrosion of structural components and wear of rails for the following cranes:

Structure	Crane(s)
Reactor Building	RB Polar Crane Reactor Vessel Tool Handling Jib Crane 5-Ton Jib Crane CRDM Jib Crane Main Fuel Handling Bridge Crane
Auxiliary Building	120-Ton Fuel Handling Area Crane Spent Fuel Pit Missile Shield Crane Spent Fuel Pool Handling Bridge Crane
EFW Pump Building	EFW Pump Building 3-Ton Crane
Circulating Water Intake Structure	Intake Gantry Crane

Administrative controls for the Program will be enhanced, prior to the period of extended operation to: (1) include in the Program all cranes within the scope of License Renewal; (2) require the responsible engineer to be notified of unsatisfactory crane inspection results involving loss of material; (3) specify the frequency of inspections for the cranes within the scope of License Renewal to be every refueling outage for cranes in the RB and every two years for cranes outside the RB; and (4) clarify that crane rails are to be inspected for abnormal wear and that members to be inspected for cracking include welds.

A.1.1.13 Fire Protection Program

The CR-3 Fire Protection Program is an existing program credited for aging management of fire protection components including penetration seals; expansion joints; fire barrier walls, ceilings, and floors; fire rated doors; Diesel Fire Service Pump fuel oil supply lines; fire barrier assemblies, such as, fire wraps on trays, pipes, and conduits; and the Halon system used for the Control Complex cable spreading room.

Prior to the period of extended operation, the Program administrative controls will be enhanced to: (1) include specific guidance for periodic inspection of fire barrier walls, ceilings, and floors including a requirement to notify Fire Protection of any deficiencies having the potential to adversely affect the fire barrier function; (2) include additional inspection criteria as described in NUREG-1801 for penetration seals; (3) include additional inspection criteria for corrosion of fire doors; and (4) specify minimum qualification requirements for personnel performing visual inspections of penetrations seals and fire doors.

A.1.1.14 Fire Water System Program

The Fire Water System Program is an existing program that includes system pressure monitoring, wall thickness evaluations, periodic flow and pressure testing in accordance with applicable National Fire Protection Association (NFPA) commitments, and periodic visual inspection of overall system condition. These activities provide an effective means to determine whether corrosion and biofouling are occurring. Inspections of the sprinkler heads assure that corrosion products that could block flow are not accumulating. These measures will allow timely corrective action in the event of system degradation to ensure the capability of the water-based Fire Suppression System to perform its intended functions.

Prior to the period of extended operation, the Program will be enhanced to: (1) incorporate a requirement to perform one or a combination of the following two activities: (a) Implement periodic flow testing consistent with the intent of NFPA 25, or (b) Perform wall thickness evaluations to verify piping is not impaired by pipe scale, corrosion products, or other foreign maternal. For sprinkler systems, this may be done by flushing, internal inspection by removing one or more sprinkler heads, or by other obstruction investigation methods, (such as technically proven ultrasonic and X-ray examination) that have been evaluated as being capable of detecting obstructions. (These inspections will be performed before the end of the current operating term. The results from the initial inspections will be used to determine inspection intervals thereafter during the period of extended operation.), (2) perform internal inspections of system piping at representative locations as required to verify that loss of material due to corrosion has not impaired system intended function. Alternately, non-intrusive inspections (e.g., ultrasonic testing) can be used to verify piping integrity. (These inspections will be performed before the end of the current operating term. The results from the initial inspections will be used to determine inspection intervals thereafter during the period of extended operation.), (3) incorporate a requirement to perform a visual inspection of yard fire hydrants annually consistent with the intent of NFPA 25 to ensure timely detection of signs of degradation, such as corrosion, and (4) consistent with the intent of NFPA 25, either replace the sprinkler heads prior to reaching their 50-year service life or revise site procedures to perform field service testing, by a recognized testing laboratory, of representative samples from one or more sample areas. (Subsequent test intervals will be based on test results.)

A.1.1.15 Aboveground Steel Tanks Program

The Aboveground Steel Tanks Program is a new program that will manage the aging effect of loss of material by performing inspections of the Fire Service Water Storage Tanks and the Condensate Storage Tank. The Program includes measures to monitor corrosion or degradation by observing the external surface of tanks, which have a protective coating, and the seal at the concrete foundation. Only carbon steel tanks are included. Monitoring of the tanks includes periodic walkdown inspections and planned preventive maintenance activities. Volumetric examinations of tank bottoms will also be performed. These actions will provide reasonable assurance that the tanks will perform their intended function consistent with the CLB throughout the period of extended operation.

A.1.1.16 Fuel Oil Chemistry Program

The Fuel Oil Chemistry Program is an existing program that maintains fuel oil quality by the purchase of quality fuel and the establishment of a diesel fuel oil testing program for both new and stored fuel oil. The Program includes sampling and testing requirements and acceptance criteria in accordance with applicable American Society for Testing Materials (ASTM) Standards specified in the CR-3 Technical Specification surveillance requirements and chemistry program procedures for fuel oil testing. Exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by verifying the quality of new oil and the addition of a biocide, a stabilizer, and corrosion inhibitors. Continued quality levels are assured by periodically checking for and removing water from tanks and by sampling to confirm that the bulk properties of water, sediment, particulate contamination, and biological growth are within administrative target values or Technical Specification limits. The effectiveness of the Program is verified using visual inspections of tanks to ensure that significant degradation is not occurring and that the component intended function will be maintained during the extended period of operation.

Prior to the period of extended operation, the Program will be enhanced to: (1) adjust the inspection frequency for the Diesel-Driven Emergency Feedwater Pump Fuel Oil Storage Tank to ensure an inspection is performed prior to the period of extended operation, (2) inspect the internal surfaces of the Diesel-Driven Fire Pump Fuel Oil Storage Tanks, and (3) develop a work activity to periodically inspect the internal surfaces of the Diesel-Driven Fire Pump Fuel Oil Storage Tanks.

A.1.1.17 Reactor Vessel Surveillance Program

The Reactor Vessel Surveillance Program is an existing program that manages the reduction of fracture toughness of the reactor vessel beltline materials due to neutron embrittlement. The Program fulfills the intent and scope of 10 CFR 50, Appendix H, by participation in the Master Integrated Reactor Vessel Surveillance Program (MIRVP) and by maintaining a fluence monitoring program. The Program evaluates neutron embrittlement by projecting upper shelf energy and pressurized thermal shock reference temperatures for reactor materials with projected neutron exposure greater than 10^{17} n/cm^2 ($E > 1.0$ MeV) after 60 years of operation and by the development of pressure-temperature limit curves. Embrittlement information is obtained in accordance with NRC Regulatory Guide 1.99, Revision 2, chemistry tables and with surveillance capsules, which have provided credible data for both the current operating period and for the period of extended operation. The surveillance program design, capsule withdrawal schedule, and evaluation of test results are in accordance with ASTM E 185-82.

Prior to the period of extended operation, the Program will be enhanced to: (1) ensure that neutron exposure conditions of the reactor vessel remain bounded by those used to project the effects of embrittlement to the end of the 60-year extended license period and (2) establish formalized controls for the storage of archived specimens to ensure availability for future use by maintaining the identity, traceability, and recovery of the archived specimens throughout the storage period.

A.1.1.18 One-Time Inspection Program

The One-Time Inspection Program is a new aging management program that employs one-time inspections to verify the effectiveness of other aging management programs or to confirm the slow progression, or the absence of, an aging effect. The Program scope includes the Water Chemistry Program, ASME Section XI Inservice Inspection Program, Fuel Oil Chemistry Program, and Lubricating Oil Analysis Program verifications specified by NUREG-1801, as well as plant-specific inspections. The Program will be completed by the addition of procedural controls for implementation and tracking.

A.1.1.19 Selective Leaching of Materials Program

The Selective Leaching of Materials Program is a new program that includes one-time inspections and qualitative determinations of the presence of selective leaching in potentially susceptible components. A sample population of susceptible components will be selected for the inspections prior to commencing the period of extended operation. The inspections will determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function(s) for the period of extended operation. This Program includes an exception to the corresponding program described in NUREG-1801 involving the use of qualitative determinations, other than Brinell hardness testing, to identify the presence of selective leaching.

A.1.1.20 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program is a new program that manages the aging effect of loss of material for the external surfaces of buried steel piping components and tanks in CR-3 systems within the scope of License Renewal. The Program includes preventive measures to mitigate corrosion by protecting the external surface of buried components through use of coating or wrapping. The Program also includes visual examination of buried piping components made accessible by excavation. Program administrative controls to be developed include ensuring an appropriate as-found pipe coating and material condition inspection is performed whenever buried piping within the scope of this Program is exposed, with a minimum frequency of at least one buried piping inspection every 10 years; verifying that there is at least one opportunistic or focused inspection performed within the ten year period prior to the period of extended operation; specifying that an inspection datasheet will be used; requiring inspection results to be documented; including precautions concerning excavation and use of backfill for License Renewal piping and tanks, including a requirement that buried pipe and tank coating inspection shall be performed, when excavated, by qualified personnel to assess its condition; and including a requirement that a coating engineer or other qualified individual should assist in evaluation of any pipe and tank coating damage or degradation found during the inspection.

A.1.1.21 One-Time Inspection of ASME Code Class 1 Small-Bore Piping

The CR-3 One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program is a new program that will manage cracking in small-bore (less than NPS 4) Class 1 piping through the use of a combination of volumetric examinations and visual inspections. The Program will manage the aging effect through the identification and evaluation of cracking in small-bore Class 1 piping. Any cracking identified in small-bore Class 1 piping resulting from stress corrosion or thermal and mechanical loading will result in periodic inspections to be managed by a plant-specific program. The Program will effectively manage the aging effect by identifying and evaluating cracking in small-bore Class 1 piping prior to loss of intended function.

A.1.1.22 External Surfaces Monitoring Program

The External Surfaces Monitoring Program is an existing program based on system inspections and walkdowns. The Program consists of periodic visual inspections of components such as piping, piping components, ducting, and other equipment within the scope of License Renewal and subject to aging management review in order to manage aging effects.

Prior to the period of extended operation, the Program will be enhanced to: (1) ensure that the Program encompasses all of the systems and components that credit it for aging management, (2) include inspection attributes adequate for detecting aging effects and mechanisms and for characterizing degradation consistent with the expected degradation of the systems and components crediting the Program for aging management, (3) incorporate measures to assure the integrity of surfaces that are inaccessible or not readily visible during both plant operations and refueling outages, and (4) incorporate inspection attributes for degradation of coatings.

A.1.1.23 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program that relies upon work order tasks that provide opportunities for the visual Inspection of internal surfaces of piping and ducting components. Work task activities will monitor parameters that may be detected by visual inspection and include change in material properties, cracking, flow blockage, hardening, loss of material, and reduction of heat transfer effectiveness. In addition to visual inspection of internal surfaces, the Program includes a limited scope of preventive maintenance activities that involve 1) physical manipulation or other investigative methods to detect aging effects and 2) inspection of outside surfaces. The extent and schedule of inspections and testing assure detection of component degradation prior to loss of intended functions.

A.1.1.24 Lubricating Oil Analysis Program

The Lubricating Oil Analysis Program is an existing program that maintains lubricating oil quality by periodic sampling for contamination in accordance with site program procedures. Exposure to contaminants, such as water and particulates, is minimized by monitoring the lube oil quality and taking corrective action when monitored parameters trend toward unacceptable or administrative limits. The program also implements periodic oil changes at fixed intervals for selected components. A particle count and check for water are performed on the old oil prior to disposal to detect evidence of abnormal wear rates, contamination by moisture, or excessive corrosion. The Program has proven effective at managing the aging effects for components exposed to lubricating oil.

A.1.1.25 ASME Section XI, Subsection IWE Program

The ASME Section XI, Subsection IWE Program is an existing program used for the aging management of accessible and inaccessible pressure retaining Containment Structure Class MC components. This Program is implemented in accordance with ASME Section XI, Subsection IWE, 2001 Edition, through the 2003 Addenda, as modified by 10CFR50.55a.

A.1.1.26 ASME Section XI, Subsection IWL Program

The ASME Section XI, Subsection IWL Program is an existing program used for the aging management of the reinforced concrete and the unbonded post-tensioning system of the CR-3 Class CC containment structure. This Program is implemented in accordance with 10 CFR 50.55(a) and ASME Section XI, Subsection IWL, 2001 Edition, through the 2003 Addenda.

A.1.1.27 ASME Section XI, Subsection IWF Program

The ASME Section XI, Subsection IWF Program consists of periodic visual examination of component supports for loss of material, change in material properties, and loss of mechanical function. The Program is an existing program implemented through plant procedures, which provide for visual examination of ISI Class 1, 2, and 3 supports. The CR-3 Program for component and pipe supports is in accordance with the requirements of ASME Section XI, Subsection IWF: 2001 Edition through the 2003 Addenda.

A.1.1.28 10 CFR 50, Appendix J Program

The 10 CFR 50, Appendix J Program is an existing program that consists of monitoring leakage rates through the containment pressure boundary, including penetrations and access openings. Containment leak rate tests assure that leakage through the primary containment, and systems and components penetrating the primary containment do not exceed the allowable leakage limits specified within the CR-3 Technical Specifications. Corrective actions are taken if leakage rates exceed established administrative limits for individual penetrations or the overall containment pressure boundary. Seals and gaskets are also monitored under the program.

The CR-3 10 CFR 50, Appendix J Program utilizes the performance-based approach of 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" Option B, and includes appropriate guidance from Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak-Test Program," as modified by NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J."

A.1.1.29 Masonry Wall Program

The CR-3 Masonry Wall Program is an existing program designed to manage the aging effects of masonry walls. For License Renewal, the Program will assure that the evaluation basis established for each masonry wall within the scope of License Renewal remains valid through the period of extended operation. The Program includes masonry walls identified as performing License Renewal intended functions within the Auxiliary Building, Control Complex, Turbine Building, Fire Service Pumphouse, and the Switchyard Relay Building. The Program is a condition monitoring program with the inspection frequencies established such that no loss of intended function would occur between inspections.

Prior to the period of extended operation, Program administrative controls will be enhanced to identify the structures that have masonry walls in the scope of License Renewal.

A.1.1.30 Structures Monitoring Program

The Structures Monitoring Program consists of periodic inspection and monitoring of the condition of structures and component supports to ensure that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. This Program is an existing program that is implemented in accordance with the Maintenance Rule, 10 CFR 50.65; NEI 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The inspection criteria are based on American Concrete Institute Standard, ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures;" and American Society of Civil Engineers, ASCE 11-90, "Guideline for Structural Condition Assessment of Existing Buildings;" as well as, Institute for Nuclear Power Operations (INPO) Good Practice document 85-033, "Use of System Engineers;" and NEI 96-03, "Guidelines for Monitoring the Condition of Structures at Nuclear Plants."

Prior to the period of extended operation, the Structures Monitoring Program will be enhanced by revising the administrative controls that implement the Program to: (1) identify all License Renewal structures and systems that credit the Program for aging management in the corporate procedure for condition monitoring of structures, (2) require notification of the responsible engineer when below grade concrete including concrete pipe is exposed so an inspection may be performed prior to backfilling, (3) require periodic groundwater chemistry monitoring including consideration for potential seasonal variations, (4) require periodic inspections of the water control structures, i.e., Circulating Water Intake Structure, Circulating Water Discharge Structure, Nuclear Service Sea Water Discharge Structure, Intake Canal, and Raw Water Pits, on a frequency not to exceed five years, (5) require periodic inspections of the Circulating Water Intake Structure submerged portions on a frequency not to exceed five years, (6) identify additional civil/structural commodities and associated inspection attributes and performance standard required for License Renewal in the corporate procedure for condition monitoring of structures, (7) identify additional inspection criteria for structural commodities in the site system walkdown checklist, (8) add inspection for corrosion to the inspection criteria for the bar racks at the Circulating Water Intake Structure as a periodic maintenance activity, (9) add an inspection of the earth for loss of form and loss of material for the Wave Embankment Protection Structure as a periodic maintenance activity, (10) include additional in-scope structures and specific civil/structural commodities in periodic maintenance activities, and (11) require periodic inspections of the Fluorogold slide bearing plates used in structural steel platform applications in the Reactor Building.

A.1.1.31 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 is a new program credited for the aging management of cables and connections not included in the CR-3 Environmental Qualification (EQ) Program. Under this Program, accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years for cable and connection jacket surface anomalies, such as embrittlement, discoloration, cracking, swelling, or surface contamination, which are precursor indications of conductor insulation aging degradation from heat, radiation or moisture. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the electrical cable or connection.

A.1.1.32 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 is a new program credited for the aging management of radiation monitoring and nuclear instrumentation cables not included in the CR-3 EQ Program. Exposure of electrical cables to adverse localized environments caused by heat, radiation, or mositure can result in reduced insulation resistance (IR). A reduction in IR is a concern for circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation circuits since it may contribute to signal inaccuracies. For radiation monitoring circuits and Gamma Metrics circuits, the review of calibration results or findings of surveillance testing will be used to identify the potential existence of cable system aging degradation. This review will be performed at least once every 10 years, with the first review to be completed before the end of the current license term. Power range cable systems used in the Excore Monitoring System will be tested at a frequency not to exceed 10 years based on engineering evaluation, with the first testing to be completed before the end of the current license term. Testing may include IR tests, time domain reflectometry tests, current versus voltage testing, or other testing judged to be effective in determining cable system insulation condition.

A.1.1.33 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 is a new program credited for the aging management of cables not included in the CR-3 EQ Program. In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested at least once every 10 years 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, or polarization index, or other testing that is state-of-the-art at the time the test is performed. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Periodic exposures that last less than a few days (e.g., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage for more than 25% of the time. Manholes associated with inaccessible non-EQ medium-voltage cables will be inspected for water accumulation and drained, as needed. The manhole inspection frequency will be based on actual field data and shall not exceed two years.

A.1.1.34 Metal Enclosed Bus Program

The Metal Enclosed Bus Program is a new program credited for the aging management of non-segregated 4.16KV and 250/125VDC Metal Enclosed Bus within the scope of

License Renewal. The Program involves various activities conducted at least once every 10 years to identify the potential existence of aging degradation. In this Program, a sample of accessible bolted connections will be checked for loose connection by using thermography or by measuring connection resistance using a low range ohmmeter. In addition, the internal portions of the bus enclosure will be visually inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion. The bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The internal bus supports will be visually inspected for structural integrity and signs of cracks. As an alternative to thermography or measuring connection resistance of bolted connections, for the accessible bolted connections that are covered with heat shrink tape, sleeving, insulating boots, etc., visual inspection of the insulation material may be used to detect surface anomalies, such as discoloration, cracking, chipping or surface contamination. If this alternative visual inspection is used to check bolted connections, the first inspection will be completed before the period of extended operation and every five years thereafter.

A.1.1.35 Fuse Holder Program

The Fuse Holder Program is a new program credited for aging management of fuse holders that are susceptible to aging effects and are located outside of active devices. Fuse holders inside active devices, such as switchgear, power supplies, power inverters, battery chargers, control panels and circuit boards are not within the scope of this Program. The Program focuses on the metallic clamp (or clip) portion of the fuse holder. The parameters monitored include corrosion and oxidation. Identified fuse holders within the scope of License Renewal will be tested at least once every 10 years. Testing may include thermography, contact resistance testing, or other appropriate testing (to be determined prior to implementation). The first test for license renewal will be completed before the period of extended operation.

A.1.1.36 Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program credited for the aging management of cable connections not included in the CR-3 EQ Program. The Program will be implemented as a one-time inspection on a representative sample of non-EQ cables connections within the scope of License Renewal prior to the period of extended operation to provide an indication of the integrity of the cable connection. The specific type of test performed will be determined prior to the test, and is to be a proven test for detecting loose connections, such as thermography, contact resistance testing, or other appropriate testing judged to be effective in determining cable connection integrity. The factors considered for sample selection are application (high, medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.) in both indoor and outdoor environments. The technical basis for the sample selections of cable connections to be tested will be provided. This Program does not include high-voltage (>35KV) switchyard connections or metal enclosed bus connections.

A.1.1.37 Carborundum (B_4C) Monitoring Program

An existing Carborundum (B_4C) Monitoring Program manages the effects of aging on the Carborundum (B_4C) panels that are located in the high density spent fuel storage racks in Spent Fuel Pool A.

Administrative controls for the Program will be enhanced, prior to the period of extended operation to: (1) include provisions to monitor and trend data for incorporation in test procedures to ensure the projection meets the acceptance criteria and (2) incorporate acceptance criteria tables for accumulated weight losses of monitored Carborundum samples.

A.1.1.38 High-Voltage Insulators in the 230KV Switchyard Program

The High-Voltage Insulators in the 230KV Switchyard Program is a new program credited for aging management of the high-voltage insulators used in the power path for the overhead transmission conductors that connect CR-3 230KV Switchyard to the Backup Engineered Safeguards Transformer (BEST). The Program inspects the insulators for salt deposits or surface contamination and mechanical wear of the steel hardware connecting the insulators to one another. The high-voltage insulators within the scope of this Program are to be inspected at least once every four years.

A.1.1.39 Reactor Coolant Pressure Boundary Fatigue Monitoring Program

The Reactor Coolant Pressure Boundary (RCPB) Fatigue Monitoring Program is an existing program that includes preventive measures to mitigate fatigue cracking caused by anticipated cyclic strains in metal components of the RCPB. This is accomplished by monitoring and tracking the significant thermal and pressure transients for limiting RCPB components in order to prevent the fatigue design limit from being exceeded. The Program addresses the effects of the reactor coolant environment on component fatigue life by including, within the Program scope, environmental fatigue evaluations of the sample locations specified in NUREG/CR-6260, "Application of NUREG/CR-5999, Interim Fatigue Curves to Selected Nuclear Power Plant Components."

A.1.1.40 Environmental Qualification (EQ) Program

The existing CR-3 EQ Program, which implements the requirements of 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. 10 CFR 50.49 requires EQ components that are not qualified for the current license term to be refurbished, replaced, or have their qualifications extended prior to reaching the aging limits established in the aging evaluation. Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ Program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met). Time-Limited Aging Analysis (TLAA) demonstration option 10 CFR §54.21(c)(1)(iii), which states that the effects of aging will be adequately managed for the period of extended operation, has been chosen. The EQ Program will manage the aging effects of the components associated with the environmental qualification TLAA.

A.1.2 EVALUATION OF TIME LIMITED AGING ANALYSES

A.1.2.1 Reactor Vessel Neutron Embrittlement

Neutron embrittlement is the term used to describe changes in mechanical properties of reactor vessel (RV) materials that result from exposure to fast neutron flux (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 defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence. Since extending the operating period from 40 years to 60 years will further increase the fluence levels, the 60-year fluence value must be determined and used to determine its impact upon the analyses used to support operation. The approach taken was that, if the existing analyses could not be demonstrated to remain valid for 60 years, new analyses were prepared for 60 years. If a revised analysis was not feasible, the aging effect identified within the time-limited aging analysis (TLAA) will be managed during the period of extended operation.

A.1.2.1.1 Neutron Fluence

End-of-life fluence is based on a projected value of effective full power years (EFPY) over the licensed life of the plant. For the current term of operation, end-of-life for CR-3 is 40 years and reactor vessel embrittlement calculations for pressurized thermal shock and upper shelf energy are based on fluence projections at 32 EFPY. The plant began operation in December 1976, and the plant lifetime capacity factor through 2005 is 68.2%. Assuming a plant capacity factor of 98.5% beyond 2005, CR-3 will accrue approximately 50.3 EFPY by December 2036. Therefore, a 54 EFPY fluence estimate used for calculating reactor vessel embrittlement for 60 years of operation is bounding for the period of extended operation.

AREVA NP (previously Framatome) developed a fluence analysis methodology that can be used to accurately predict the fast neutron fluence in the reactor vessel using surveillance capsule dosimetry and/or cavity dosimetry to verify the fluence predictions. This methodology was developed through a full-scale benchmark experiment that was performed at the Davis-Besse Unit 1 reactor. The benchmark experiment demonstrated that the AREVA NP methodology was unbiased and was accurate well within the NRC suggested standard deviation of 20%. The AREVA NP fluence analysis methodology is compliant with NRC Regulatory Guide (RG) 1.190, as described in topical report BAW-2241NP-A, Revision 1, "Fluence and Uncertainty Methodologies," December 1999. The AREVA NP methodology was used to calculate the neutron fluence exposure to the CR-3 reactor vessel. The fast neutron fluence (neutron energy (E) > 1.0 MeV) at the reactor vessel upper and lower plates, as well as specific welds, was calculated in accordance with the requirements of RG 1.190.

The 54 EFPY fluence values include ex-vessel cavity dosimetry data from Cycles 11 and 12 and plant operation through Cycle 14. To account for a measurement uncertainty recapture, the Cycle 14 fluxes were used for Cycle 15 and increased by a factor of 1.02 for Cycles 16 and 17; the Cycle 16 and 17 flux was increased by a factor of 1.25 for Cycles 18 through 60 years.

Reactor pressure vessel boundary components outside the beltline region have been evaluated to determine whether additional materials should be considered "beltline" material for the period of extended operation. The beltline, as defined by 10 CFR 50.61(a)(3), is the region of the reactor pressure vessel that directly surrounds the effective height of the active core and adjacent regions of the reactor pressure vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection for the most limiting material with regard to radiation damage. The threshold fluence for potential beltline material is 1.0E+17 n/cm^2, E > 1.0 MeV. The beltline materials for CR-3 for 60 years (54 EFPY) include the following:

Component	Material
Nozzle Belt Forging - Lower	AZJ 94
Upper Shell Plate	C4344-1
Upper Shell Plate	C4344-2
Lower Shell Plate	C4347-1
Lower Shell Plate	C4347-2
Upper Shell Circumferential (Circ.) Weld (Inside 40%)	SA-1769
Upper Shell Circ. Weld (Outside 60%)	WF-169-1
Upper Shell Axial Weld	WF-8
Upper Shell Axial Weld	WF-18
Upper Shell to Lower Shell Circ. Weld	WF-70
Lower Shell Axial Welds	SA-1580

The limiting beltline circumferential weld based on fluence for CR-3 at 54 EFPY is WF-70, heat number 72105. The fluence at 54 EFPY for weld WF-70, is 1.56E+19 n/cm^2. In the Master Integrated Reactor Vessel Material Surveillance Program (MIRVP), two capsules with weld wire heat number 72105 have been irradiated to fluence values equal to or greater than 1.56E+19 n/cm^2 and tested. Therefore, the MIRVP program covers the fluence at 54 EFPY for CR-3 weld WF-70, and no additional surveillance material or testing is required for 60 years of operation.

The limiting beltline axial weld based on fluence for CR-3 at 54 EFPY is WF-8, heat number 8T1762. This heat of material is not in the MIRVP, and there is no need to add this material since the CR-3 Linde 80 beltline weld materials, including WF-8, are adequately represented by the eight heats of material in the MIRVP program.

The limiting shell plate material for CR-3 is C4344-1, which was included in CR-3-specific capsules, and all specimens have been removed and tested. The 54 EFPY fluence at plate C4344-1 is predicted to be 1.60E+19 n/cm^2. Capsule CR3-F, which contained C4344-1 material, received a fluence of 1.08E+19 n/cm^2 and was removed and tested. The MIRVP has determined that no further testing is required for material C4344-1 since the plate material is not the limiting material for the CR-3 vessel and the MIRVP meets the requirements of 10 CFR 50, Appendix H.

Therefore, the neutron fluence has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii) using a methodology previously approved by the NRC. These fluence projections will be used for evaluating fluence-based TLAAs for CR-3 License Renewal.

A.1.2.1.2 Upper Shelf Energy Evaluation

Upper-shelf energy (USE) is a measure of the average energy absorbed by Charpy impact specimens tested at a temperature above the upper end of the transition region. 10 CFR 50, Appendix G, states that reactor vessel beltline materials must have Charpy upper-shelf energy (C_VUSE) in the transverse direction for base metal and along the weld for weld metal of no less than 75 ft-lb in the unirradiated condition, and must maintain C_VUSE of no less than 50 ft-lb throughout the licensed life of the vessel, unless it can be demonstrated that lower values of energy will provide margins of safety against fracture equivalent to those required by ASME Code, Section XI, Appendix G.

Upper shelf energies for beltline plates and forgings at 54 EFPY were determined using Regulatory Guide 1.99, Revision 2, Position 1.2, and are all above 50 ft-lb, which is acceptable.

As is the case for the current term of operation, the C_VUSE values for all beltline welds are below 50 ft-lb, requiring an equivalent margins analysis (EMA) for the period of extended operation. The methodology used to evaluate CR-3 beltline welds at 60 years is consistent with the EMA methods reported in BAW-2192PA, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A & B Service Loads," April 1994; BAW-2178PA, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads," April 1994; and BAW-2275A, "Low Upper Shelf Toughness Fracture Mechanics Analysis of B&W

Designed Reactor Vessels for 48 EFPY," August 1999. BAW-2275A comprises Appendix B of BAW-2251A, "Demonstration of Management of Aging Effects for the Reactor Vessel," January 2002.

An updated EMA was performed on CR-3 limiting beltline welds WF-70, WF-8, and WF-18 to consider the effect of increased fluence on the J-integral, which is a function of fluence. The applied J-integral, which is due to loading, is not a function of fluence and remains unchanged from earlier analyses. The results of the updated analysis show that the first acceptance criterion of $J_{0.1} / J_1 > 1.0$ from ASME Section XI, Article K-2200(a)(1) for Level A and B service loading is met. The results show that the second acceptance criterion of $J_{0.1} / J_1 > 1.0$ for Level C and D service loading is also met. Therefore, the limiting CR-3 welds provide margins of safety equivalent to those of Appendix G of the Section XI of the ASME Code and have adequate upper-shelf toughness, and satisfy the requirements of Appendix G to 10 CFR 50 for operation through 54 EFPY.

Based on the above discussion, the USE analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.1.2.1.3 Pressurized Thermal Shock Analysis

10 CFR 50.61 defines screening criteria for embrittlement of reactor pressure vessel materials in pressurized-water reactors, as well as actions that are required if these screening criteria are exceeded. The screening criteria limit the degree that a vessel material may increase in its reference temperature for pressurized thermal shock - RT_{PTS}, following neutron irradiation of the reactor pressure vessel. For circumferential welds, the pressurized thermal shock (PTS) screening criterion is 300°F maximum. For plates, forgings, and axial weld materials, the screening criterion is 270°F maximum. The projected EOL RT_{PTS} values must be shown to remain below the applicable screening temperature.

A PTS evaluation for the CR-3 RV beltline materials was performed in accordance with 10 CFR 50.61. The PTS reference temperature, RT_{PTS}, values are calculated by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT} and the margin term to cover the uncertainties in the values of initial RT_{NDT} copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective RV beltline materials copper and nickel contents and the neutron fluence applicable to the CR-3 RV for License Renewal at 54 EFPY.

Evaluations for the CR-3 RT_{PTS} values were performed for each CR-3 reactor vessel beltline material with chemistry factors determined from Tables 1 and 2 in 10 CFR 50.61. In addition, the chemistry factors for the upper shell plate, heat number C4344-1 was recalculated using the available CR-3 surveillance data in accordance with RG 1.99, Revision 2.

The CR-3 RT_{PTS} values for the reactor vessel beltline materials for the period of extended operation were calculated using 54 EFPY inside wetted surface fluence projections. The limiting longitudinal welds are WF-8 and WF-18 with an RT_{PTS} of 231.3°F, which is below the screening criterion of 270°F. The limiting circumferential weld is WF-70 with an RT_{PTS} of 253.8°F, which is below the screening criterion of 300°F. Therefore, the analyses for the shift in PTS reference temperature have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.1.2.1.4 Operating Pressure-Temperature (P-T) Limits Analysis

The adjusted reference temperature (ART) is the value of Initial RT_{NDT} + ΔRT_{NDT} + margins for uncertainties at a specific reactor vessel location. Neutron embrittlement increases the ART. Thus, the minimum temperature at which a reactor vessel is allowed to be pressurized increases over the licensed period. The ART of the limiting beltline material is used to adjust the beltline pressure-temperature (P-T) limits to account for radiation effects. 10 CFR Part 50, Appendix G requires reactor vessel thermal limit analyses to determine operating P-T limits for boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences. Operating limits for pressure and temperature are required for three categories of operation: 1) hydrostatic pressure tests and leak tests, 2) non-nuclear heat-up/cooldown and low level physics tests, and 3) core critical operation.

The ART values for the CR-3 reactor vessel beltline region materials are calculated in accordance with RG 1.99, Revision 2, by adding the initial RT_{NDT} to the predicted radiation-induced ΔRT_{NDT}, and a margin term to cover the uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 54 EFPY. The evaluations for the CR-3 ART were performed at the 1/4T and 3/4T wall location of each beltline material with chemistry factors determined from Tables 1 and 2 in RG 1.99, Revision 2. In addition, the chemistry factors for the Upper Shell Plate, heat number C4344-1, were recalculated using the available CR-3 surveillance data.

In this manner, ART results for the CR-3 reactor vessel beltline region materials applicable to 54 EFPY were determined. Based on these, the controlling beltline material for the CR-3 reactor vessel with respect to P-T limits are the Upper Shell Circumferential Weld (Inside 40%) SA-1769 (at 1/4T) and the Upper/Lower Shell Circumferential Weld WF-70 (at 3/4T).

The pressure-temperature operating limits were developed in accordance with the requirements of 10 CFR Part 50, Appendix G, utilizing the analytical methods and flaw acceptance criteria of topical report BAW-10046A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," Revision 2, June 1996, and ASME Code Section XI, Appendix G, 2001 edition through 2003 Addenda. CR-3 has implemented changes in the P-T limit curves throughout the current period of operation. ASME Code Cases N-588 and N-640 are incorporated in ASME Section XI, Appendix G, 2001 edition through 2003 Addenda. With the incorporation of the new methodology from ASME Code Section XI, Appendix G, 2001 edition through 2003 Addenda, and the improved replacement RV head, the 54 EFPY uncorrected P-T limits provide more operating room than the 32 EFPY uncorrected P-T curves.

CR-3 will continue to implement changes in the P-T limit curves in the PTLR, as required by Appendix G of 10 CFR part 50, for the remainder of the current period of operation and for the extended period of operation. The P-T limits for the remainder of the current period of operation and for the extended period of operation will be managed by using approved fluence calculations when there are changes in power or core design, and with surveillance capsule results. Updating the P-T limit curves using the described approach will assure that the operational limits remain valid for the remainder of the current period of operation and for the extended period of operation. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

A.1.2.1.5 Low-Temperature Overpressure Limits Analysis

ASME Section XI, Appendix G, establishes procedures and limits for RCS pressure and temperature primarily for low temperature conditions to provide protection against non-ductile failure of the RV. The Low Temperature Overpressure Protection System (LTOPS) assures that these limits are not exceeded when it is enabled at low temperatures.

The LTOP setpoints for CR-3 have been reanalyzed to support operation to the end of the period of extended operation. The revised LTOP setpoints will be implemented when the revised P-T limit curves are implemented, prior to exceeding 32 EFPY. Maintaining the LTOP setpoints in accordance with Appendix G of 10 CFR 50 and 10 CFR 50.60 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

A.1.2.1.6 Reactor Vessel Underclad Cracking

Underclad cracking (UCC) refers to intergranular separations in the heat affected zones of low alloy base metal under austenitic stainless steel cladding. B&W conducted an intensive investigation of UCC in the 1970 and showed the subject flaws are present only in A-508, Class 2, forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc process such as the six-wire, strip, and the two-wire series arc. The investigations also noted that no anomalies were observed in SA-533

Grade B, Class 1 plate materials clad by any of the high-heat-input processes. The results of the fracture mechanics analysis demonstrated that the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted growth due to design fatigue cycles.

The fracture mechanics analysis for underclad cracking was updated in BAW-2274A, "Fracture Mechanics Analysis of Postulated Underclad Cracks in B&W Designed Reactor Vessels for the Period of Extended Operation," August 1999, to include the period of extended operation. The revised analysis concluded that postulated underclad cracking in the RVI meets the acceptance criteria of the ASME Code, Section XI, IWB-3612. The maximum crack growth and applied stress intensity factor for normal and upset conditions occur in the nozzle belt region. The fracture toughness margin for normal and upset conditions was determined to be 3.63, which is greater than the required toughness margin of 3.16. The maximum applied stress intensity for the emergency and faulted condition occurs in the closure head to head flange regions. The fracture toughness margin for emergency and faulted condition was 2.42, which is greater than the required toughness margin of 1.41.

The revised analysis was based on fracture toughness properties associated with 60-year fluences and was intended to bound the B&W fleet. While CR-3 is not specifically listed as a participant in BAW-2274A, the generic evaluation used bounding loads from the entire fleet of B&W 177 FA lowered loop operating plants. The loads used in the analysis are bounding for CR-3, provided that the material properties of applicable CR-3 vessel are bounded by those presented in BAW-2274A. Three vessel regions were evaluated: (1) nozzle belt, (2) closure flange, and (3) beltline.

The ART at the inside surface of CR-3 Lower Nozzle Belt Forging AZJ 94 is 3.0°F higher than the ART evaluated for the previously limiting forging. Therefore, the CR-3 nozzle belt forging is not bounded and was re-analyzed for 54 EFPY. The results show that the postulated 0.353 in.-deep flaw on the inside surface of the CR-3 Lower Nozzle Belt Forging satisfies the IWB-3612 acceptance criteria for fracture toughness margin. Considering 54 EFPY of fatigue crack growth, the final flaw size is 0.487 in., and the fracture toughness margin of 3.49 for Level A and B Service Loadings is greater than the required value of 3.16. The available fracture toughness margin for Level C and D Service Loadings is 2.50 which exceeds the required value of 1.41. The results demonstrate that a postulated underclad crack in the CR-3 Lower Nozzle Belt Forging would satisfy the flaw acceptance criteria of the ASME Code for 54 EFPY of operation over a period of 60 years.

Evaluation of the closure flange in BAW-2274A identified limiting closure flange material based on an inside surface fluence of 7.78E+16 n/cm^2. For CR-3, the fluence at 54 EFPY at the closure flange is 4.38E+13 n/cm^2 and remains bounded.

CR-3 beltline upper and lower shell plates are fabricated from SA-533 Grade B, Class 1 and are not susceptible to underclad cracking. Since CR-3 does not have A-508, Class 2 forgings in the upper and lower shell region, the increase in ART due to increased fluence at 54 EFPY is not relevant for the evaluation of underclad cracking.

Based on the above, the underclad cracking analysis for CR-3 has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.1.2.1.7 Reduction in Fracture Toughness of Reactor Vessel Internals

Reduction of fracture toughness of 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 both the irradiation temperature and neutron fluence. The reactor vessel internals components most susceptible to reduction in fracture toughness are those nearest to the reactor core.

The effect of irradiation on the mechanical properties and deformation limits for the reactor vessel internals was evaluated for the current term of operation in Appendix E of topical report BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," June, 1970. The 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. This analysis is a TLAA for the current term of operation.

In accordance with the guidance of NUREG-1801, Revision 1, regarding the aging management of reactor vessel internals components, CR-3 will:

1. Participate in the industry programs for investigating and managing aging effects on reactor internals,

2. Evaluate and implement the results of the industry programs as applicable to the reactor internals, and

3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor vessel internals to the NRC for review and approval.

Based on this evaluation, the reduction in fracture toughness of reactor vessel internals will be managed, consistent with the commitment to participate in industry programs related to the reactor vessel internals, through the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). This commitment is reiterated in Subsection A.1.1.

A.1.2.2 Metal Fatigue

The CR-3 approach is to identify the latest design fatigue analyses associated with each NSSS component within the reactor coolant pressure boundary (RCPB) in order to demonstrate that the design analyses will remain bounding through the period of extended operation. Components within the scope of this review include non-pressure boundary reactor internals components.

The first step in the evaluation was to establish the current fatigue design bases for the major NSSS components. This was done by reviewing component design reports, amendments to those reports, and the assessment of the impact of the NRC approved measurement uncertainty recapture 1.6% power uprate to identify the full set of NSSS design transients used in the fatigue evaluations.

The second step in the evaluation was to gather and review plant design information, actual plant transient data from the RCS and other sources, and archived RCS operational parametric data. This information was used to develop actual operational transients experienced from plant startup through December 2007. The transient data was obtained from the CR-3 Cycle and Transient Monitoring Program, input from plant personnel, and historical data obtained from CR-3 records.

There is considerable margin after 30 years of operation to the NSSS design transient cycles originally defined for 40 years, and CR-3 has determined there is no need to increase the number of NSSS design transients for the period of extended operation. The RCS CUFs may be conservatively projected to 60 years of operation by multiplying the 40-year CUFs by a factor of 1.5; this is equivalent to multiplying the NSSS design transient cycles by a factor of 1.5. Therefore, 40-year usage factors in excess of 0.67 (1.0/1.5) may be assumed to exceed the ASME Code, Section III limit of 1.0 at 60-years. This method of usage factor projection is conservative since CR-3 has determined that it is unlikely that the NSSS design transients for 40 years will be exceeded at 60 years of operation.

The final step in the evaluation was to consider the effects of the reactor water environment on 40-year fatigue usage factors at selected NSSS locations as identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Design Curves to Selected Nuclear Power Plant Components," as required by NUREG-1801, Revision 1.

A.1.2.2.1 Reactor Vessel Fatigue Analyses

The reactor vessel (RV) was designed in accordance with Section III of the ASME Code – Class 1, for the replacement closure head, and Class A, for the remaining vessel items; therefore, metal fatigue was considered in the design of the RV components. CUF analyses for the RV are applicable TLAAs, since they are based on NSSS design transient cycles originally defined for 40 years. For the components that are part of the

RV, one pressure-retaining item has a 40-year CUF that exceeds 0.67: the Lower Service Support Structure attachment weld with a CUF of 0.72. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation using the RCPB Fatigue Monitoring Program, and the effects of aging will be adequately managed according to 10 CFR 54.21(c)(1)(iii).

A.1.2.2.2 Reactor Vessel Internals Fatigue Analyses

The CR-3 reactor vessel internals (RVI) were designed and constructed prior to the development of ASME Code requirements for core support structures. Therefore, existing industry structural practice was used in the design of the internals structural members; and the only specific fatigue analyses performed in the original design were those that addressed high cycle fatigue reported in BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibration," September 1, 1972. In modifications following original design, plant-specific fatigue analyses were performed for the reactor vessel internals replacement bolts as presented in BAW-1843PA, "The B&WOG Evaluation of Internals Bolting Concerns in 177 FA Plants," January 1986, and BAW-1789P, "The B&WOG Evaluation of Internals Bolting Concerns in 177 FA Plants," August 1984. These topical reports summarize fatigue analyses performed to the ASME Code, Section III, Subsection NG, including both high-cycle fatigue from flow induced vibrations (FIV) and low-cycle fatigue from NSSS design transients.

BAW-10051 calculated stress values for the redesigned RVI and compared them to endurance limit stress values. The methodology used in BAW-10051 was extended from 40 years to 60 years by multiplying the assumed endurance limit cycles by 1.5 and then using 1013 cycles to determine the endurance limit based on more recent ASME fatigue curves which extend now to 1011 cycles. The component item stress values in BAW-10051 were compared to the recalculated endurance limit values and were shown to be acceptable. In this manner, the FIV analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii),

The RV internals bolts that were replaced at CR-3 include 120 Upper Core Barrel bolts made from A-286, 60 Lower Core Barrel bolts made from X-750, 96 Lower Thermal Shield bolts made from X-750, and 72 Surveillance Specimen Holder Tube (SSHT) bolts made from X-750. The Lower Core Barrel bolts and Lower Thermal Shield replacement bolts have 40-year CUFs that exceed 0.67. Therefore, the effects of aging on the intended functions will be adequately managed for the period of extended operation using the RCPB Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

A.1.2.2.3 Control Rod Drive Mechanism Fatigue Analysis

The "Type C" control rod drive mechanism (CRDM) motor tube was designed in accordance with ASME Code, Section III, Class A, 1968 Edition with Addenda through

Summer 1970, and metal fatigue was considered in the design of the component. CUFs of the CRDM motor tube were not calculated as it was shown that the motor tube did not require analysis for cyclic operation in accordance with ASME Section III, paragraph N-415.1. Calculations performed for the CRDM motor tube are based on NSSS design transients which have not been increased for the period of extended operation. Therefore, the analyses are acceptable for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i), since the NSSS design transients have not been revised.

A.1.2.2.4 Reactor Coolant Pump Fatigue Analysis

The reactor coolant pumps (RCPs) were designed in accordance with the ASME Code, Section III, Class A, but were not code stamped, and metal fatigue was considered in the design of the component. CUFs of the RCPs are applicable TLAAs since the CUFs are based on NSSS design transient cycles originally defined for 40-years of operation. Considering the RCP casing, cover, and shaft, the cover has the largest 40-year design usage factor at 0.65. In addition, calculations performed in accordance with N-415.1(a) through N-415.1(f) of the ASME Code, Section III, for the RCP seal and heat exchanger are based on NSSS design transients that have not been increased for the period of extended operation. Therefore, the analyses for the RCP casing, cover, and shaft have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii), and the analyses of the RCP seal and heat exchanger performed in accordance with N-415.1(a) through N-415.1(f) of the ASME Code, Section III, are acceptable for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.1.2.2.5 Steam Generator Fatigue Analysis

The Once-Through Steam Generators (OTSGs) were designed in accordance with the ASME Code, Section III, Class A, and metal fatigue was considered in the design of the components. CUFs of the OTSG components are applicable TLAAs since the CUFs are based on NSSS design transient cycles originally defined for 40 years of operation. For the components that are part of the OTSG, five items have 40-year CUFs that exceed 0.67: the Emergency Feedwater (EFW) Nozzle Studs, Main Feedwater (MFW) Nozzle, Mechanical Sleeves, Remote Welded Plug, and the Support Skirt. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation by means of the CR-3 RCPB Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

A.1.2.2.6 Pressurizer Fatigue Analysis

The Pressurizer was designed in accordance with the ASME Code, Section III, Class A, and metal fatigue was considered in the design of the component. The Pressurizer surge nozzle was modified in 2007 to include a weld overlay over the Alloy 600 weld that connects the surge nozzle to a stainless steel safe end. The weld overlay was designed in accordance with the 1989 Edition of ASME Code, Section III, Subsection NB. For the components that are part of the Pressurizer, three items have 40-year CUFs that exceed 0.67: the Surge Nozzle with weld overlay, the Heater Bundle closure seal weld, and the Thermowell Nozzle. Therefore, the effects of aging on the intended functions will be adequately managed for the period of extended operation by means of the CR-3 RCPB Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

A.1.2.2.7 Reactor Coolant Pressure Boundary Piping (USAS B31.7) Fatigue Analysis

RCPB piping includes all piping within the ASME, Section XI, Subsection IWB inspection boundary at CR-3. The IWB inspection boundary includes B&W-supplied main coolant piping and portions of Architect/Engineer-supplied ancillary systems, e.g., Decay Heat Removal, Core Flood, and Make Up & Purification Systems, including Low Pressure Injection, High Pressure Injection, and Makeup/Letdown piping, attached to the Reactor Coolant System piping. The IWB inspection boundary within the ancillary systems typically extends to the first or second isolation valve or to a flow restricting orifice. The B&W-supplied main coolant piping was designed in accordance with USAS B31.7, and the ancillary systems connected to the main coolant piping were designed in accordance with USAS B31.1.

The scope of USAS B31.7 piping at CR-3 includes the 36 in. hot leg piping, including attached branch connections and safe ends; 28 in. cold leg piping, including attached branch connections and safe ends; Pressurizer surge line piping; and Pressurizer spray line piping. CUFs of USAS B31.7 RCPB piping are applicable TLAAs since they are based on NSSS design transient cycles originally defined for 40 years of operation.

For the components that are part of the RCPB piping, the Pressurizer spray line piping and High Pressure Injection/Makeup (HPI/MU) Nozzle safe end CUFs exceed 0.67 at 40 years.

In accordance with NRC letter (H. Silver) to FPC (P. Beard), "Crystal River Unit 3 - NRC Bulletin 88-08 'Thermal Stress in Piping Connected to Reactor Coolant Systems,' (TAC No. M69621)," dated June 18, 1992, the piping items within the scope of NRC Bulletin 88-08 at CR-3 include the HPI/MU nozzle, safe end, and thermal sleeve. Fatigue of the HPI/MU nozzle, safe end, and thermal sleeve is evaluated above for the period of extended operation.

Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii) by means of the RCPB Fatigue Monitoring Program.

A.1.2.2.8 USAS B31.1.0 Piping - RCPB Class 1 Fatigue Analysis

RCPB Class 1 piping designed in accordance with USAS B31.1.0 Piping Code includes piping in ancillary systems connected to the B&W-supplied main coolant piping. These systems include Decay Heat Removal, Core Flood, and Makeup & Purification Systems, including Low Pressure Injection, High Pressure Injection, and Makeup/Letdown piping. For piping designed in accordance with the USAS B31.1.0-1967 Code rules, the designer was required to determine the overall number of thermal cycles anticipated for the component in 40 years, and was required to apply stress range reduction factors if this number exceeded 7,000. Power piping at CR-3 complies with USAS B31.1.0-1967. Since these analyses were based upon the number of cycles expected to occur during the original license period, these analyses are also considered to be TLAAs.

The applicable transient cycles for piping systems designed in accordance with USAS B31.1.0-1967 rules were originally determined by summing the individual transients to which the component would be exposed in 40 years. In order to evaluate these TLAAs for 60 years, the numbers of cycles now expected to occur in 60 years should be compared to the numbers of design cycles that were considered in these analyses. For the RCPB systems, the number of thermal cycles correlates with plant heatups and cooldowns. Since the transient set (and associated cycles) in the RCS Functional Specification is being maintained, the analytical basis for these components remains unchanged. Therefore, the analyses for these components remain valid for the period of extended operation in accordance with10 CFR 54.21(c)(1)(i).

The HPI/MU safe end is welded to a stainless steel spool piece that was analyzed for fatigue analysis in accordance with USAS B31.7 to support NRC Bulletin 88-08. The 40-year CUF for the spool piece is 0.94. Therefore, the effects of aging on the intended function(s) for the period of extended operation will be adequately managed by means of the RCPB Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

A.1.2.2.9 USAS B31.1.0 Piping - Non-Class 1 Fatigue Analysis

Piping designed in accordance with USAS B31.1.0 Piping Code was not required to have analyses of cumulative fatigue usage, but cyclic loading was considered in a simplified manner in the design process. For piping designed in accordance with the USAS B31.1.0-1967 code rules, the designer was required to determine the overall number of thermal cycles anticipated for the component in 40 years, and was required to apply stress range reduction factors if this number exceeded 7,000. Power piping at CR-3 complies with USAS B31.1.0-1967. Since these analyses were based upon the number of cycles expected to occur during the original license period, these analyses are also considered to be TLAAs.

The applicable transient cycles for piping systems designed in accordance with USAS B31.1.0-1967 rules were originally determined by summing the individual transients to which the component would be exposed in 40 years. In order to evaluate these TLAAs for 60 years, the numbers of cycles now expected to occur in 60 years should be compared to the numbers of design cycles that were considered in these analyses. For most systems, the number of thermal cycles correlates with plant heatups and cooldowns, which are limited to 240 cycles. Since the transient set (and associated cycles) in the RCS Functional Specification is being maintained, the analytical basis for these components remains unchanged. Therefore, the analyses for these components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

For components in systems whose cycles do not track plant heatups and cooldowns, a specific evaluation of the components operating history was performed. Examples of components in this group include engine exhaust components for diesel engines in the Emergency Diesel Generator, Emergency Feedwater, and Fire Protection Systems; Sampling piping and components in the Liquid and Post-Accident Liquid Sampling Systems; and the Turbine-Driven Emergency Feedwater Pump Turbine. Evaluations were performed that projected the number of expected cycles in 60 years. The evaluations concluded that the components remain qualified through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.1.2.2.10 Environmentally-Assisted Fatigue Analysis

The effects of reactor water environment on fatigue were evaluated for a subset of representative components. The representative components selected were based upon the evaluations in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The representative components evaluated are as follows:

- Reactor Vessel Shell and Lower Head (including incore instrumentation nozzles)
- Reactor Vessel Inlet and Outlet Nozzles
- Pressurizer Surge Line (including hot leg and Pressurizer surge nozzles)
- HPI/MU Nozzle
- Core Flood Nozzle
- Decay Heat Removal System Class 1 Piping

The methods used to evaluate environmental effects on fatigue were based on NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels," NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," and NUREG/CR-6717, "Environmental Effects of Fatigue Crack Initiation in Piping and Pressure Vessel Steels." In addition, the method used to obtain environmental effects for nickel-based alloy was obtained from H. S. Metha and S. R. Goeeslin, "Environmental Factor Approach to Account for Water Effects in Pressure Vessel and Piping Fatigue Evaluations," Nuclear Engineering and Design, 1998. Environmental fatigue life correction factors (F_{en}) were used to obtain adjusted cumulative fatigue usage (U_{en}) which includes the effects of reactor water environments.

Environmentally-adjusted U_{en} factors at all locations are based on application of environmental penalty factors to the ASME 40-year CUF values. Bounding F_{en} values of 2.45 for low-alloy steel, 15.35 for stainless steel, and 1.49 for Alloy 600 were applied to the 40-year design CUFs with the exception of surge line piping and decay heat injection piping.

For surge line piping, the ASME Section III analysis of record for CR-3 was revised to include the effects of environmentally assisted fatigue. The environmental correction factor F_{en} from NUREG/CR-5704 was used to determine the number of allowable cycles for each load pair. The F_{en} correction factor was obtained by integration from peak to valley considering transformed metal temperature, transformed strain rate, and transformed dissolved oxygen. The strain rate was assumed to be at 0.0004%/sec or less, and transformed strain rate was held constant at Ln (0.001). Based on historical data, dissolved oxygen is 0.05 ppm or less, and transformed oxygen was held constant at 0.026. Transformed metal service temperature was determined by integration of metal temperature for the load pair analyzed. Therefore, the F_{en} varies from 2.55 (when metal temperature is less than 392 °F) to a maximum of 15.35 (when metal temperature equals or exceeds 392 °F). Thermal striping, which was considered separately, was assigned an F_{en} of 1.0 as the maximum calculated strain amplitude is less than the threshold strain amplitude of 0.097% listed in NUREG/CR-5704.

The Decay Heat Injection piping at CR-3 was designed in accordance with USAS B31.1 and therefore did not receive an explicit CUF evaluation. A fatigue evaluation of the Decay Heat Injection piping was performed specifically for License Renewal using USAS B31.7, 1969 Edition. The CUF was multiplied by the bounding F_{en} value of 2.55.

Based on the results of this evaluation, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation using the CR-3 RCPB Fatigue Monitoring Program in accordance with 10 CFR 54.21(c)(1)(iii).

A.1.2.2.11 RCS Loop Piping Leak-Before-Break Analysis

The application of leak-before-break (LBB) to the CR-3 RCS main coolant piping is described in Topical Report BAW-1847, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," Revision 1, September 1985, which provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted loading conditions and was approved by the NRC. The TLAA in BAW-1847, Revision 1, addresses flaw growth. In addition, the report included a qualitative assessment of thermal aging of cast austenitic stainless steel (CASS) RCP inlet and outlet nozzles; this assessment is not considered a TLAA. However, reduction of fracture toughness by thermal aging of the RCP inlet and exit nozzles was evaluated for license renewal to ensure that the conclusions of the LBB evaluation reported in BAW-1847, Revision 1, remain valid for the period of extended operation.

The LBB analysis reported in BAW-1847, Revision 1, postulated a surface flaw at selected locations of the piping system, and a fatigue crack growth analysis was performed to demonstrate that the flaws are likely to propagate in the through-wall direction and develop leakage before they will propagate circumferentially around the pipe. Flaw growth calculations were based on the original transient cycles that were defined for 40 years of operation for the RCS components. These transient cycles have not been revised for License Renewal and are being monitored by the Reactor Coolant Pressure Boundary Fatigue Monitoring Program. If a transient cycle count approaches or exceeds the allowable design limit, corrective actions are taken. Therefore, the flaw growth evaluation reported in BAW-1847, Revision 1, remains valid for the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(i) since CR-3 has not revised the transients defined in the RCS design specification for License Renewal.

The susceptibility of the RCS main coolant piping to thermal aging was qualitatively addressed in Section 3.3.4.3 of BAW-1847, Revision 1. There are no RCS main coolant piping segments fabricated from CASS. However, the heat affected zone of the welded joint that connects the wrought austenitic stainless steel pump transition piece to the CASS RCP inlet and exit nozzles may be susceptible to thermal embrittlement. At the time of the report, it was assumed that the fracture toughness values for aged CASS were bounded by the ferritic piping and ferritic weldments. Since that time, a significant amount of data has been obtained regarding thermal aging of CASS materials. Therefore, the fracture toughness curves for the ferritic base metal and ferritic weld metals used in the RCS piping LBB analysis were compared to the lower-bound fracture toughness curves of CR-3 RCP CASS materials (i.e., statically cast CF8M) from NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," by Chopra and Shack, Argonne National Laboratory Report, U.S. Nuclear Regulatory Commission, Washington DC, May 1994. The fracture toughness curve of the lower-bound CASS material is below the fracture toughness curves used in the RCS piping LBB analysis. Therefore, the assumption in BAW-1847, Revision 1, that the fracture toughness of the ferritic piping and ferritic weldments bounds the fracture toughness of CASS required further evaluation for License Renewal.

A flaw stability analysis was performed using the lower-bound CASS fracture toughness curves from the Argonne report cited above to show acceptability of LBB for the RCS main coolant piping for the period of extended operation. The most limiting material and location used in the RCS piping LBB analysis was determined to be the base metal material of the straight section of the 28 in. cold leg pipe. Both the suction and discharge nozzles of the RCP casings are attached to the 28 in. cold leg pipes and have similar geometries and applied loads as the limiting location used for the LBB analysis. The discharge and suction nozzles of the RCP casings were evaluated for LBB using lower-bound CASS fracture toughness properties.

Bounding 10 gpm leakage crack sizes for the RCP suction and discharge nozzle were determined using a method consistent with that reported in BAW-1847, Revision 1. In the revised analysis, the applied loadings were considered using the absolute sum load combination method. Therefore, in accordance with NUREG-0800, SRP 3.6.3, a margin of 1.0 on load was used. The leakage flaw size for the suction nozzle was determined to be 4.31 in. and the leakage flaw size for the discharge nozzle was determined to be 4.43 in. In addition, a crack extension value of 0.6 in. was considered in the flaw stability analysis. A flaw stability analysis was performed for the RCP suction and discharge nozzles, and the discharge nozzle was found to be limiting. The maximum applied J value at the discharge nozzle, for the 10 gpm leakage flaw size, was determined to be 0.510 kips/in. The critical crack size was determined to be 10.8 in. Therefore, the margin on flaw size was determined to be 2.4 (i.e., 10.8/4.43). This is greater than the required margin of 2.0 in accordance with SRP 3.6.3. Based on the results of this analysis, it is concluded that the required margins for LBB per SRP 3.6.3 are met, even with consideration of the lower-bound CASS fracture toughness properties for the suction and discharge nozzles.

Therefore, it has been demonstrated that the fatigue flaw growth analysis reported in BAW-1847, Revision 1, remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) since the number of NSSS design transients will not be revised for License Renewal. The remainder of the generic LBB analysis for the B&W operating plants reported in BAW-1847, Revision 1, remains valid for the period of extended operation with the exception of the original qualitative assessment of reduction of fracture toughness by thermal aging of CASS. The assessment of reduction of fracture toughness by thermal aging of CASS is not considered a TLAA. Reduction of fracture toughness of the RCP nozzles was determined to be acceptable for the period of extended operation through the flaw stability analysis described above. In addition, recent NRC concerns related to Alloy 82/182 and LBB analyses are addressed in the industry's submittal MRP-140, "Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds," EPRI, Palo Alto, CA: 2005, 1011808. The Alloy 82/182 welds within the scope of BAW-1847, Revision 1, are the welds that connect the 28 in. stainless steel carbon steel cold leg piping to the stainless safe pump transition pieces. Based on the above, the flaw growth analysis remains valid for the period of extended operation.

A.1.2.3 Environmental Qualification of Electric Equipment

The existing CR-3 EQ Program is credited for aging management of electric equipment important to safety in accordance with the requirements of 10 CFR 50.49. 10 CFR 50.49 requires EQ components that are not qualified for the current license term to be refurbished, replaced, or have their qualifications extended prior to reaching the aging limits established in the aging evaluation. Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ Program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met). TLAA demonstration option 10 CFR §54.21(c)(1)(iii), which states that the effects of aging will be adequately managed for the period of extended operation, has been chosen. The EQ Program will manage the aging effects of the components associated with the environmental qualification TLAA.

A.1.2.4 Concrete Containment Tendon Prestress

The CR-3 Reactor Building consists of a prestressed reinforced concrete cylinder and hemispherical dome. The cylinder wall is prestressed utilizing a two-way post-tensioning system. The dome roof is prestressed utilizing a three-way post-tensioning system. The prestressing tendons tend to lose their prestressing forces with time due to creep and shrinkage of concrete and relaxation of the prestressing steel. Loss of tendon prestress is a TLAA; therefore, the adequacy of the prestressing forces is reviewed for the period of extended operation.

There have been eight tendon surveillance tests since CR-3 plant startup in December 1976. Since 1997, these tests have been performed under the ASME Section XI, Subsection IWL Program. The IWL program inspects a sample of tendons from each category (dome, vertical, and hoop) and confirms that the acceptance criteria have been met and, therefore, that tendon prestresses will remain above minimum required values for the succeeding inspection interval. The program recalculates the regression analysis trend lines of these three groups, based on individual tendon forces consistent with NRC Information Notice 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," using individual-tendon data rather than averages, to confirm whether average prestresses are expected to remain above their minimum required values for the remainder of the licensed operating period.

For the purposes of extending the CR-3 plant operating period, the regression analysis was used to extrapolate the tendon prestress forces to the end of the extended period of operation. The values computed demonstrated that prestress in all three groups of tendons should remain above the applicable minimum required values for the 60-year period of extended operation and that, therefore, the tendons should maintain their design basis function.

The TLAA evaluation addressed tendon loss of preload, using 10 CFR 54.21(c)(1)(ii), to project the tendon preload to the end of the 60-year service period for each group of tendons. The projected "average" preload values at the end of the 60-year service period are then compared with the required minimum average tendon preload. For each group of tendons, the projected preload value exceeds the required minimum average tendon preload. Therefore, prestress in all three groups of tendons will remain above the applicable minimum required values for the period of extended operation; and the tendons will perform their intended function.

A.1.2.5 Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis

A.1.2.5.1 Fuel Transfer Tube Expansion Bellows Cycles

Fuel Transfer Tube Expansion Bellows connect the Fuel Transfer Tubes to the Refueling Canal in the Reactor Building and to the Spent Fuel Pool in the Auxiliary Building. Per plant specifications, the Expansion Bellows shall be fabricated, as a minimum, to the requirements of Section VIII of the ASME Code and inspected in accordance with the requirements of ASME Code, Section III, Class B vessels. Each Expansion Bellows is designed to withstand a total of 5,000 cycles of expansion and compression over a lifetime of 40 years. This analysis ensures that the lifetime may be extended to 60 years without exceeding the design criterion of 5,000 cycles.

Expansion bellows cycles occur each refueling outage due to thermal cycling when the Fuel Transfer Tubes are flooded with refueling water then drained for return of the plant to operation. Assuming a period of mid-loop operation that involves a partial drain and refilling of the canal, bellows cycling would occur twice every refueling outage; however, cycling has been assumed to occur three times every refueling outage for additional conservatism. The number of cycles applied to the expansion bellows in the Reactor Building is assumed also to apply to the expansion bellows in the Auxiliary Building. There are 19 refueling outages planned for the 40-year life of the plant. The number of refueling outages over 60 years of life is 60/40 x 19 = 28.5 or 29 refueling outages. The maximum number of operating cycles projected to be experienced over the 29 refueling outages during a 60-year period is:

29 refueling outages x 3 cycles/refueling outage = 87 cycles.

Since the total number of cycles for the Fuel Transfer Tube Expansion Bellows is less than 5,000 cycles, no reanalysis of the design calculations is necessary. Therefore, the Fuel Transfer Tube Expansion Bellows design analyses of record remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.1.2.6 Other Plant-Specific Time-Limited Aging Analyses

A.1.2.6.1 Bedrock Dissolution from Groundwater Analysis

FSAR Section 2.5.3.4 documented a Bedrock Solutioning Study at CR-3. The solutioning process is the result of fresh water entering the underground areas below the plant and attacking the limestone sediments leaving a labyrinth of channels throughout the rock mass. The percent of rock dissolved over the 40-year life of the plant was calculated using different methods; and a determination was made that the percent of rock dissolved represents an insignificant amount and would have an insignificant effect on the stability of the rock mass. To extend this value to 60 years, the total maximum volume of dissolved bedrock was multiplied by the ratio of 60years/40years for an additional 20 years of extended life.

The analysis for 40 years was computed in three ways; however, the conclusion of one of these methods, the extreme case, was not used in the conclusions presented in the FSAR. Therefore, the 40-year results reported in FSAR were computed in two ways. One method determined that 1.5×10^{-5} % of the bedrock was dissolved over the forty year life of the plant. This was based on assuming the law of uniformitarianism was applicable and that 15% of the rock mass has been dissolved in 40 million years and definitely in more than 40,000 years. The 15% was based on the results of the exploratory and grout hole drilling at the site which indicated that the volume of solution channels was probably not greater than 15%. Thus, the total maximum volume of dissolved bedrock was:

$$1.5 \times 10^{-5}\ \% \times 60/40 = 2.25 \times 10^{-5}\ \%.$$

Another method determined that 4×10^{-3} % of the bedrock would be dissolved over the 40-year life of the plant. This was based on information obtained from the U.S. Geologic Survey for dissolved solids over a large land area that included the CR-3 site. For an additional 20 years of extended life, the total maximum volume of dissolved bedrock was determined as follows:

$$4 \times 10^{-3}\ \% \times 60/40 = 6 \times 10^{-3}\ \%.$$

The conclusions of the 60-year projections are that the range in percent of the rock dissolved would be between 2.25×10^{-5} % and 6×10^{-3} %. Dissolved volumes calculated by these methods still represent insignificant amounts. Further, the grouting process used in the foundation of Crystal River Units 2 and 3 reduced the permeability of the carbonate rocks from in excess of 65,500 feet per year to less than 2,000 feet per year. With the permeability decreased by more than 30 times, exposure of the limestone to potential solvent groundwater is effectively reduced by the same factor. It is concluded that the natural solution process will not affect the structural integrity of the foundation of the CR-3 for the period of extended operation. Therefore, the analysis of the volume of bedrock solutioning has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

APPENDIX B

AGING MANAGEMENT PROGRAMS

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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 AMP discussed in this Appendix has ten (10) program elements. These elements are defined in Appendix A.1, Section A.1.2.3, of NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Rev. 1, U. S. Nuclear Regulatory Commission, September 2005, (the SRP-LR). These elements have been incorporated into the AMPs described in Sections X and XI of NUREG-1801, "Generic Aging Lessons Learned (GALL)," Rev. 1, U.S. Nuclear Regulatory Commission, September 2005, (NUREG-1801). For those plant-specific, non-NUREG-1801, AMPs, SRP-LR guidance has been used to develop a detailed discussion of the 10 elements. Therefore, the AMP descriptions in this Appendix address the ten elements either implicitly, by means of a consistency review using the programs in NUREG-1801, or explicitly, by a comparison to the program elements in NUREG-1800.

B.1.2 METHOD OF DISCUSSION

For those AMPs whose acceptability is based on consistency with the programs in Sections X and XI of NUREG-180, each program discussion is presented in the following format:

- A summary description of the program is provided.
- A statement is made regarding consistency of the program with NUREG-1801.
- If applicable, exceptions to the NUREG-1801 program are summarized, and justifications provided.
- If applicable, enhancements to ensure consistency with NUREG-1801 are proposed. A proposed schedule for completion is discussed.
- Operating Experience information specific to the program is provided.
- A conclusion section provides a statement of reasonable assurance that the program is, or will be, effective.

For those programs that are plant-specific, the following format is followed:

- A summary description of the program is provided.
- A discussion of each of the 10 elements in the program is provided. Operating experience is one of the 10 elements.
- A conclusion section provides a statement of reasonable assurance that the program is, or will be, effective.

B.1.3 QUALITY ASSURANCE PROGRAM AND ADMINISTRATIVE CONTROLS

Three elements common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the CR-3 Quality Assurance (QA) Program that implements the requirements of 10 CFR 50, Appendix B. A description of the QA Program is provided in FSAR Section 1.7, Quality Program (Operational).

Corrective actions:

Corrective actions are implemented in accordance with procedures established to implement the Corrective Action Management Policy and requirements of 10 CFR 50, Appendix B, Criterion XVI. Conditions adverse to quality, such as, failures, malfunctions, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, measures are implemented to ensure that the cause of the nonconformance is determined and that corrective action is taken to prevent recurrence. In addition, the root cause of the significant condition adverse to quality and the corrective action implemented are documented and reported to appropriate levels of management. The Corrective Action Program is consistent with the guidelines in the appendix to Volume 2 of NUREG-1801.

Confirmation Process:

The focus of the confirmation process is on the follow-up actions that must be taken to verify effective implementation of corrective actions and 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 determinations, and
- reviews of corrective action effectiveness.

The corrective action process 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 an investigation with appropriate follow-up corrective action. The CR-3 confirmation process is consistent with the appendix to Volume 2 of NUREG-1801.

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 QA for the operational phase of nuclear power plants and (2) the requirements of 10 CFR 50, Appendix B, Criterion VI. The CR-3 administrative controls process is consistent with the appendix to Volume 2 of NUREG-1801.

B.1.4 OPERATING EXPERIENCE

Industry operating experience (OE) was incorporated into the License Renewal process through a review of industry documents to identify aging effects and mechanisms that could challenge the intended function of systems and structures within the scope of License Renewal. Review of plant-specific OE was performed to identify aging effects experienced. The review of plant-specific OE involved electronic database searches of plant information. As appropriate, discussions with system engineers were conducted to identify additional aging concerns.

OE regarding existing programs/activities, including past corrective actions resulting in program enhancements, was considered. This information provides objective evidence that the effects of aging have been, and will continue to be, adequately managed.

B.1.5 AGING MANAGEMENT PROGRAMS

The AMPs addressed in this Appendix are listed on Table B-1. Information on the table notes whether programs are either existing or new. Each AMP is addressed in the individual Subsections of Section B.2.

B.1.6 TIME-LIMITED AGING ANALYSES AGING MANAGEMENT PROGRAMS

Table B-1 also includes a listing of AMPs used to resolve Time-Limited Aging Analyses (TLAAs). Evaluation of TLAA-related AMPs in accordance with 10 CFR 54.21(c), are discussed in Section B.3.

B.2 AGING MANAGEMENT PROGRAMS

The correlation between NUREG-1801 programs and CR-3 AMPs is shown on the following table.

TABLE B-1 CORRELATION OF NUREG-1801 AND CR-3 AGING MANAGEMENT PROGRAMS

NUREG-1801 Number	NUREG-1801 Program	CR-3 Program	NUREG-1801 Comparison
NUREG-1801 Chapter XI			
XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program See Subsection B.2.1.	Existing program consistent with NUREG-1801
XI.M2	Water Chemistry	Water Chemistry Program See Subsection B.2.2.	Existing program consistent with NUREG-1801
XI.M3	Reactor Head Closure Studs	Reactor Head Closure Studs Program See Subsection B.2.3..	Existing program consistent with NUREG-1801 with enhancement
XI.M4	BWR Vessel ID Attachment Welds	Not applicable to PWRs.	Not applicable
XI.M5	BWR Feedwater Nozzle	Not applicable to PWRs.	Not applicable
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not applicable to PWRs.	Not applicable
XI.M7	BWR Stress Corrosion Cracking	Not applicable to PWRs.	Not applicable
XI.M8	BWR Penetrations	Not applicable to PWRs.	Not applicable
XI.M9	BWR Vessel Internals	Not applicable to PWRs.	Not applicable
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Program See Subsection B.2.4.	Existing program consistent with NUREG-1801
XI.M11	Nickel-Alloy Nozzles and Penetrations	Not credited for aging management.	Not applicable; see Note 1
XI.M11A	Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors	Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program See Subsection B.2.5.	Existing program consistent with NUREG-1801
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Based on a thermal aging susceptibility evaluation, the applicable CASS components are not susceptible to thermal aging.	Not applicable

NUREG-1801 Number	NUREG-1801 Program	CR-3 Program	NUREG-1801 Comparison
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program See Subsection B.2.6.	New program consistent with NUREG-1801
XI.M14	Loose Part Monitoring	Not credited for aging management.	Not applicable
XI.M15	Neutron Noise Monitoring	Not credited for aging management.	Not applicable
XI.M16	PWR Vessel Internals (no longer an AMP in NUREG-1801, Rev. 1)	Not credited for aging management.	Not applicable; see Note 2
XI.M17	Flow-Accelerated Corrosion	Flow-Accelerated Corrosion Program See Subsection B.2.7.	Existing program consistent with NUREG-1801
XI.M18	Bolting Integrity	Bolting Integrity Program See Subsection B.2.8.	Existing program consistent with NUREG-1801 with enhancement
XI.M19	Steam Generator Tube Integrity	Steam Generator Tube Integrity See Subsection B.2.9.	Existing program consistent with NUREG-1801
XI.M20	Open-Cycle Cooling Water System	Open-Cycle Cooling Water System Program See Subsection B.2.10.	Existing program consistent with NUREG-1801 with enhancement
XI.M21	Closed-Cycle Cooling Water System	Closed-Cycle Cooling Water System Program See Subsection B.2.11.	Existing program consistent with NUREG-1801 with exceptions
XI.M22	Boraflex Monitoring	Not credited for aging management.	Not applicable
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Inspection of Overhead Heavy Load and Light Load Handling Systems Program See Subsection B.2.12.	Existing program consistent with NUREG-1801 with enhancement
XI.M24	Compressed Air Monitoring	Not credited for aging management.	Not applicable
XI.M25	BWR Reactor Water Cleanup System	Not applicable to PWRs.	Not applicable
XI.M26	Fire Protection	Fire Protection Program See Subsection B.2.13.	Existing program consistent with NUREG-1801 with exceptions and enhancement
XI.M27	Fire Water System	Fire Water System Program See Subsection B.2.14.	Existing program consistent with NUREG-1801 with enhancement
XI.M28	Buried Piping and Tanks Surveillance	Not credited for aging management.	Not applicable

NUREG-1801 Number	NUREG-1801 Program	CR-3 Program	NUREG-1801 Comparison
XI.M29	Aboveground Steel Tanks	Aboveground Steel Tanks Program See Subsection B.2.15.	New program consistent with NUREG-1801
XI.M30	Fuel Oil Chemistry	Fuel Oil Chemistry Program See Subsection B.2.16.	Existing program consistent with NUREG-1801 with exceptions and enhancement
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance Program See Subsection B.2.17.	Existing program consistent with NUREG-1801 with exception and enhancement
XI.M32	One-Time Inspection	One-Time Inspection Program See Subsection B.2.18.	New program consistent with NUREG-1801
XI.M33	Selective Leaching of Materials	Selective Leaching of Materials Program See Subsection B.2.19.	New program consistent with NUREG-1801 with exception
XI.M34	Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection Program See Subsection B.2.20.	New program consistent with NUREG-1801
XI.M35	One-Time Inspection of ASME Code Class 1 Small-Bore Piping	One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program See Subsection B.2.21.	New program consistent with NUREG-1801
XI.M36	External Surfaces Monitoring	External Surfaces Monitoring Program See Subsection B.2.22.	Existing program consistent with NUREG-1801 with enhancement
XI.M37	Flux Thimble Tube Inspection	Not credited for aging management.	Not applicable
XI.M38	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program See Subsection B.2.23.	New program consistent with NUREG-1801
XI.M39	Lubricating Oil Analysis	Lubricating Oil Analysis Program See Subsection B.2.24.	Existing program consistent with NUREG-1801 with exception
XI.S1	ASME Section XI, Subsection IWE	ASME Section XI, Subsection IWE Program See Subsection B.2.25.	Existing program consistent with NUREG-1801
XI.S2	ASME Section XI, Subsection IWL	ASME Section XI, Subsection IWL Program See Subsection B.2.26.	Existing program consistent with NUREG-1801

NUREG-1801 Number	NUREG-1801 Program	CR-3 Program	NUREG-1801 Comparison
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI, Subsection IWF Program See Subsection B.2.27.	Existing program consistent with NUREG-1801
XI.S4	10 CFR Part 50, Appendix J	10 CFR Part 50, Appendix J Program See Subsection B.2.28.	Existing program consistent with NUREG-1801
XI.S5	Masonry Wall	Masonry Wall Program See Subsection B.2.29.	Existing program consistent with NUREG-1801 with enhancement
XI.S6	Structures Monitoring	Structures Monitoring Program See Subsection B.2.30.	Existing program consistent with NUREG-1801 with enhancement
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Not credited for aging management.	Not applicable
XI.S8	Protective Coating Monitoring and Maintenance	Not credited for aging management.	Not applicable
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 Subsection B.2.31.	New program consistent with NUREG-1801
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 Subsection B.2.32.	New program consistent with NUREG-1801
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 Subsection B.2.33.	New program consistent with NUREG-1801
XI.E4	Metal Enclosed Bus	Metal Enclosed Bus Program See Subsection B.2.34.	New program consistent with NUREG-1801
XI.E5	Fuse Holders	Fuse Holder Program See Subsection B.2.35.	New program consistent with NUREG-1801 with exceptions
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 Program See Subsection B.2.36.	New program consistent with NUREG-1801 with exception

NUREG-1801 Number	NUREG-1801 Program	CR-3 Program	NUREG-1801 Comparison
NUREG-1801 Chapter X			
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Reactor Coolant Pressure Boundary Fatigue Monitoring Program See Subsection B.3.1.	Existing program consistent with NUREG-1801
X.E1	Environmental Qualification (EQ) of Electric Components	Environmental Qualification (EQ) Program See Subsection B.3.2.	Existing program consistent with NUREG-1801
Plant-Specific			
None	Not applicable	Carborundum (B_4C) Monitoring Program See Subsection B.2.37.	Not applicable - Plant-Specific
None	Not applicable	High-Voltage Insulators in the 230KV Switchyard Program See Subsection B.2.38.	Not applicable - Plant-Specific

Notes:

1. CR-3 has provided in the FSAR Supplement a commitment to comply with applicable NRC Orders and to implement applicable (1) Bulletins and Generic Letters and (2) staff-accepted industry guidelines.

2. CR-3 has provided in the FSAR Supplement a commitment to: (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

B.2.1 ASME SECTION XI, INSERVICE INSPECTION, SUBSECTIONS IWB, IWC AND IWD PROGRAM

Program Description

The American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program consists of periodic volumetric, surface, and/or visual examination, and leakage testing of Class 1, 2, and 3 pressure retaining components and their integral attachments to detect degradation of components and determine appropriate corrective actions. The Program Plan for the Fourth 10-Year interval at CR-3 has been developed and prepared to meet the ASME Code, Section XI, 2001 Edition with addenda through 2003.

NUREG-1801 Consistency

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program is an existing program consistent with NUREG-1801, Section XI.M1.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

CR-3 OE includes cracking of HPI Nozzles/Thermal Sleeves. Cracking was initially detected in a weld in the safe end/thermal sleeve region of the normal duty makeup line. Investigation as to the cause of cracking identified concerns with the design/installation of the existing thermal sleeves. Corrective actions included the replacement of all four of the thermal sleeves with a modified design. Follow-up actions include a commitment to perform nondestructive examination to confirm nozzle and thermal sleeve integrity during selected refueling outages. These inspections have been integrated into the CR-3 4th interval ISI Program. Notably, these small-bore Class 1 lines would ordinarily be subject to a sampling based verification of integrity under the CR-3 One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program. Since cracking has already been detected in these lines, they will not be included in sample populations under this program, but rather will default to the defined inspection schedule in the CR-3 Section XI Program.

A search of ISI program results from the third inspection interval was conducted and provides evidence that the CR-3 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program is effective in identifying aging effects and is critically monitored, effective, and continually improving. Inspection findings from this interval include:

- A CRDM Motor Housing had a rejectable indication and was replaced.
- An unacceptable UT indication was found on CRDM Nozzle #32. A nozzle repair was required.
- A rejectable exam result for OTSG B lower inspection cover flange bolting was reported. The bolting was removed and visually examined with satisfactory results.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the Program is effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews; qualified personnel are assigned as program managers and are given authority and responsibility to implement the Program; and adequate resources are committed to Program activities.

Conclusion

Implementation of the CR-3 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program provides reasonable assurance that applicable aging effects will be managed such that the ISI Class 1, 2, and 3 components within the scope of License Renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation.

B.2.2 WATER CHEMISTRY PROGRAM

Program Description

To mitigate aging effects on component surfaces that are exposed to water as a process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chlorides, fluorides, and sulfates) that accelerate corrosion and cracking. This Program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. Alternatively, chemical agents, such as corrosion inhibitors, oxygen scavengers, and biocides, may be introduced to prevent certain aging mechanisms. The CR-3 Water Chemistry Program is currently based on the latest version of the Electric Power Research Institute (EPRI) guidelines. The CR-3 Water Chemistry Program will be updated as revisions to the guidelines are released.

NUREG-1801 Consistency

The Water Chemistry Program is an existing program that is consistent with NUREG-1801, Section XI.M2.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The EPRI guideline documents have been developed based on plant experience and have been shown to be effective over time with their widespread use in the industry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system from unacceptable levels of contaminants in the boric acid, introduction through the free surface of the spent fuel pool (which can be a natural collector of airborne contaminants), or introduction of oxygen during cooldown. Ingress of demineralizer resins into the primary system has caused IGSCC of Alloy 600 vessel head penetrations. Inadvertent introduction of sodium thiosulfate into the primary system has caused IGSCC of steam generator tubes. The SCC has occurred in safety injection lines, charging pump casing cladding, instrument nozzles in safety injection tanks, and stainless steel piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant. Steam generator tubes and plugs and Alloy 600 penetrations have experienced primary water stress corrosion cracking. Steam generator tubes have experienced SCC, intergranular attack, wastage, and pitting.

Carbon steel support plates in steam generators have experienced general corrosion. The steam generator shell has experienced pitting and stress corrosion cracking.

CR-3 has reviewed the industry OE related to maintenance of a benign environment described in NUREG-1801 to ensure that applicable recommendations have been captured.

Progress Energy has performed five assessments of the CR-3 Water Chemistry Program from 2001 through 2007. These assessments have identified issues and weaknesses to be addressed through the Corrective Action Program.

Nuclear Condition Reports (NCRs) for the Environmental and Chemistry Unit at CR-3 were reviewed for the period from June, 2001 to March, 2008. These NCRs were searched for items related to chlorides, fluorides, sulfates, oxygen, etc. or programmatic deficiencies. There were instances related to increases in contaminants due to equipment issues. However, trending data for these contaminants indicate that for the vast majority of the time the levels were well below the threshold for age related degradation. Specific examples of OE include:

- During startup from a recent refueling outage, CR-3 experienced high dissolved oxygen levels in the Pressurizer water space. Pressurizer oxygen levels per procedure are required to be below 100 ppb when the Pressurizer temperature exceeds 250°F. During plant heatup, as expected the Pressurizer temperature reached 250°F. Pressurizer water space oxygen levels were measured and eventually increased to greater than 2000 ppb before returning to less than 100 ppb. Pressurizer water space oxygen levels again increased to greater than 2000 ppb but were returned to less than 100 ppb.

  During an outage, when the RCS/Pressurizer are drained to lower levels and/or opened for maintenance, oxygen ingress into the system is inevitable. Oxygen is removed from the RCS and Pressurizer through controlled system venting and hydrazine additions during the startup process. During the outage, substantial welding occurred on Pressurizer penetrations associated with the Alloy 600 mitigation project. This evolution introduced significantly higher levels of oxygen into the Pressurizer than during a normal outage. Much more of the Pressurizer was open to atmosphere allowing more oxygen to eventually become trapped in the Pressurizer.

  The apparent cause was not performing sufficient venting of the Pressurizer during startup from the outage. Although normal venting occurred, it proved insufficient. No additional venting of the Pressurizer occurred to compensate for the additional oxygen that had been introduced. A contributing cause was that the hydrazine addition to the Pressurizer was performed in accordance with chemistry procedure guidance, but it proved insufficient as oxygen levels initially were reduced within the Pressurizer but returned to elevated levels once further plant heatup commenced.

Corrective actions included a revision to an operating procedure to include reinforcing information regarding the importance of venting the Pressurizer during startup and a revision to a chemistry procedure to include additional guidance for hydrazine additions to Pressurizer when the Pressurizer has been opened for extensive maintenance.

- The deaerator outlet is the Mode 1, greater than 15% power, feedwater dissolved oxygen monitoring point. This point is selected due to the lower temperature of the water and the shorter sample transport time which results in less hydrazine/dissolved oxygen reaction than at the outlet of feedwater heaters where the feedwater is hotter and the sample transport time is longer. When the turbine was placed on line, the deaerator outlet dissolved oxygen concentration was greater than 3 ppb and remained greater than 3 ppb until just after feedwater pump 1B was placed into service. During this time, the feedwater dissolved oxygen concentration was in the chemistry procedure Action Level 1 range (greater than 3 ppb but less than 10 ppb) for a total of 21 minutes. The feedwater dissolved oxygen concentration was in the chemistry procedure Action Level 2 range (greater than or equal to 10 ppb) for a total of 8 hours 27 minutes. The feedwater dissolved oxygen concentration was returned to specification well within the Action Level 1 timeframe of one week and also well within the Action Level 2 timeframe of 100 hours.

The CR-3 Water Chemistry Program is currently based on the latest version of the EPRI guidelines. EPRI periodically updates the water chemistry guidelines, as new information becomes available. The CR-3 Water Chemistry Program will be updated as revisions to the guidelines are released, to develop a more proactive program that minimizes age-related degradation.

The OE review of the CR-3 Water Chemistry Program concluded that this Program is upgraded based on industry experience and research. These continuous improvements assure the capability of the CR-3 Water Chemistry Program to support the safe operation of CR-3 throughout the extended period of operation.

Conclusion

The continued implementation of the CR-3 Water Chemistry Program provides reasonable assurance that the applicable aging effects will be managed so that the systems and components within the scope of this Program will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.3 REACTOR HEAD CLOSURE STUDS PROGRAM

Program Description

The CR-3 Reactor Head Closure Studs Program is an inspection program which manages cracking and loss of material for the Reactor Vessel Closure Head Stud Assembly. In addition to the condition monitoring elements of the Program, the CR-3 Reactor Head Closure Studs Program includes certain preventive measures recommended by Regulatory Guide (RG) 1.65, "Material and Inspection for Reactor Vessel Closure Studs." This aging management Program is implemented primarily through the plant Inservice Inspection (ISI) Program. The Closure Head Stud Assembly comprises the studs, nuts, and washers that are inspected under the CR-3 ISI Program.

The inspection schedule is in accordance with ASME B&PV Code, Section XI, IWB-2400, and the extent and frequency is in accordance with Table IWB-2500-1, Examination Category B-G-1. This will ensure that aging effects will be discovered and repaired before loss of intended function. Examination results are evaluated according to IWB-3100. Acceptance standards are identified in IWB-3400 and IWB-3500. In addition to the examinations performed under the CR-3 ISI Program, the CR-3 Reactor Head Closure Studs Program also credits Code required visual VT-2 examinations which are conducted to detect leaks during system pressure or functional tests. Repair and replacement are performed in conformance with the requirements of IWA-4000.

The CR-3 Reactor Head Closure Studs Program includes inspections that provide reasonable assurance that the effects of cracking and loss of material would be identified prior to loss of intended function. The preventive measures include using a manganese base phosphate coating, avoiding the use of metal-plated stud bolting, and avoiding the use of lubricants that contain molybdenum disulfide.

NUREG-1801 Consistency

The CR-3 Reactor Head Closure Studs Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M3.

Exceptions to NUREG-1801

None.

Enhancements

<u>Program Elements Affected</u>

- **Preventive Actions**
  The CR-3 Reactor Head Closure Studs Program will be enhanced to select an

alternate lubricant that is compatible with the fastener material and the contained fluid.

Operating Experience

A review of plant-specific OE did not identify cracking or loss of material for the CR-3 Closure Head Stud Assembly. The Inservice Inspection Summary Reports for Interval 3 were reviewed, and there have been no aging effects identified that have been attributed to wear or stress-corrosion cracking. Assessments of the Inservice Inspection Program from February 2000 through September 2007 were reviewed. The assessments indicate that the Inservice Inspection Program is effective in executing its requirements and programmatic deficiencies are identified and corrected in a timely fashion.

A review of plant-specific OE did not identify cracking or loss of material for the CR-3 Closure Head Stud Assembly. The 3rd Interval Inservice Inspection Summary Reports were reviewed and the inspection results were acceptable. Recent assessments of the Inservice Inspection/Inservice Testing (ISI/IST) Programs yielded the following results:

- The CR-3 Nuclear Assessment Section performed a focused review of the Ten-Year ISI Project. This review was accomplished using performance-based techniques including technical reviews/analysis in accordance with plant procedures.

  One issue identified was that some key project attributes required for successful completion of the Ten Year ISI project had not been implemented. The corrective action to prevent recurrence was to establish a recurring outage milestone for the project sponsor and to verify that the expectations are understood. This corrective action has been completed.

- Another assessment of the ISI/IST Programs was performed in 2007. Based on observations, document reviews, and personnel interviews conducted during the assessment, the CR-3 ISI/IST Programs was effectively executing the requirements in the areas assessed. However, two weaknesses were identified:
  - Expectations were not clear for the online Non-Destructive Examination (NDE) inspections to ensure consistency in processing design versus inservice deficiencies for plant equipment.
  - Benchmarking activities on some of the Program Health Reports were not documented as directed by procedure.

The following corrective actions to prevent occurrence were taken:
  - Create and implement Procedural guidance for NDE.
  - Provide training to increase the proficiency of personnel in performing and documenting benchmark activities.

As identified in NUREG-1801, industry OE includes cracking in BWR pressure vessel head studs. NUREG-1801 is based on industry OE through January 2005. Recent industry OE has been reviewed for applicability and no other industry OE has been identified as a result of this review. Any relevant new industry OE will be captured through the normal OE review process where it is screened for applicability. This process will continue through the period of extended operation.

Conclusion

The CR-3 Reactor Head Closure Studs Program is a condition monitoring program implemented primarily with the CR-3 Inservice Inspection Program per the requirements of the ASME Code, Section XI, Subsection IWB and includes certain preventive measures recommended by RG 1.65. Based on the evaluation of this Program, there is reasonable assurance that the CR-3 Reactor Head Closure Studs Program will adequately manage cracking and loss of material for the Reactor Vessel Closure Head Stud Assembly so that applicable intended functions will be maintained consistent with the CLB for the period of extended operation.

B.2.4 BORIC ACID CORROSION PROGRAM

Program Description

The Boric Acid Corrosion Program implements systematic measures to ensure that leaking borated coolant does not lead to the degradation of the leakage source or adjacent mechanical, electrical and structural components susceptible to boric acid corrosion. The Program consists of: (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy of corrective actions. The Boric Acid Corrosion Program includes plant-specific reactor coolant pressure boundary (RCPB) boric acid leakage identification and inspection procedures to ensure that leaking borated coolant does not lead to degradation of the leakage source or adjacent structures, and provides assurance that the RCPB will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The Program was developed in response to the recommendations of NRC Generic Letter 88-05.

NUREG-1801 Consistency

The Boric Acid Corrosion Program is an existing program consistent with NUREG-1801, Section XI.M10.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Boric Acid Corrosion Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the Boric Acid Corrosion Program is effectively implemented to meet regulatory, process, and procedure requirements, including periodic assessments and review of OE; qualified personnel are assigned as program managers and are given authority and responsibility to implement the Program; and adequate resources are committed to Program activities.

A review of responses to NRC generic correspondence, plant condition reports, and inspections was conducted and showed the CR-3 Boric Acid Corrosion Program to be critically monitored and continually improving. Specific examples of OE include:

- While performing a Boric Acid Corrosion Program inspection which involved cleaning the boric acid residue from the packing of the valve being inspected, semi-wet boric acid leakage was observed leaking outside the packing gland and dripping down on the process piping and the floor. This constituted an active leak and an NCR was initiated. There was no observed component corrosion or degradation since the affected components were constructed of stainless steel. A drip bag was installed to prevent further contamination of the floor. A work request was implemented to stop the leak and clean the area. There was no further leakage observed.

- A Swagelok fitting was found to be leaking at the rate of 2 drops per minute. This was considered to be an active leak and an NCR was initiated with a work request to tighten the fitting. An engineering assignment was initiated to perform the Boric Acid Corrosion Program evaluation. The tasks were performed as required and the leak stopped.

Based on these results, the OE review provides evidence that the Boric Acid Corrosion Program practices will continue to assure the integrity of the subject components.

Conclusion

Implementation of the Boric Acid Corrosion Program will provide reasonable assurance that applicable aging effects will be managed such that the components susceptible to boric acid corrosion within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.5 NICKEL-ALLOY PENETRATION NOZZLES WELDED TO THE UPPER REACTOR VESSEL CLOSURE HEADS OF PRESSURIZED WATER REACTORS PROGRAM

Program Description

Since the issuance of NRC Generic Letter 97-01, CR-3 has been an active participant in industry initiatives relating to Alloy 600 and the specific issue of degradation of Vessel Head Penetration (VHP) nozzles. Since GL 97-01, additional OE identified occurrences of circumferential cracking in VHP nozzles. This resulted in the issuance of NRC Bulletin 2001-01 which required CR-3 to evaluate the VHP nozzles for susceptibility.

Subsequently, NRC Bulletins 2002-01 and 2002-02 were issued as a result of several cracked and leaking Alloy 600 VHP nozzles within the industry including the degradation of the reactor pressure vessel head at Davis-Besse. In response to the referenced NRC Bulletins, CR-3 provided additional assurance that the plant programs are adequate to prevent degradation as observed in the industry.

On February 11, 2003, NRC Order EA-03-009 was issued to establish interim inspection requirements for Reactor Pressure Vessel (RPV) Heads at Pressurized Water Reactors. Subsequently, First Revised NRC Order EA-03-009 was issued on February 20, 2004 to revise certain inspection aspects of the original Order. The Order (as amended) resulted in changes to the CR-3 program for managing cracking in the VHP nozzles. The Order (as amended) provided requirements for determining a susceptibility ranking, and mandated inspection requirements commensurate with the plant's susceptibility ranking. The RPV head was replaced in the Fall of 2003. The replacement RPV head has been designed to minimize the concerns for Control Rod Drive Mechanism (CRDM) nozzle cracking and leakage associated with Primary Water Stress Corrosion Cracking (PWSCC) of the Alloy 600 nozzle material. Alloy 690 base and weld material was used for the CR-3 CRDM nozzles on the replaced RPV head. As described in NRC Order EA-03-009, CR-3 is in susceptibility category "Replaced."

In accordance with Section IV-C-(4) of the Order, for those plants in the "Replaced" susceptibility category, no RPV head and head penetration nozzle inspections were required during the outage in which the RPV head was replaced.

Beginning with initial service, until the replacement RPV head reaches 8 Effective Degradation Years, RPV head and head penetration nozzle inspections are performed as follows:

- An inspection meeting the requirements of paragraph IV.C.(5)(a)[bare metal visual inspection] must be completed at least every third refueling outage or every five (5) years, whichever occurs first.

- The requirements of paragraph IV.C.(5)(b)[non-destructive (NDE) examination] must be completed at least every four (4) refueling outages or every seven (7) years, whichever occurs first.

Since CR-3 is on 24-month fuel cycles, the inspection frequency expressed in years is the most limiting.

The CR-3 Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program is implemented through the plant Inservice Inspection (ISI) Program by the use of augmented inspections.

NUREG-1801 Consistency

The CR-3 Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program is an existing plant program that is consistent with NUREG-1801, Section XI.M11A.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

During Refueling Outage 10 (February 1996), CR-3 performed a 100% bare metal detailed inspection through the access ports. Boron was noted on the reactor vessel head (RVH) attributed to leaking CRDM flanges. Also various RVH cleaning activities were performed through the access ports. Note that CR-3 had an extended shutdown from September 1996 through February 1998.

During Refueling Outage 11 (October 1999), CR-3 performed a visual inspection for evidence of leakage at the "mouse hole" access openings with no leakage observed. However, experience from Oconee Nuclear Stations and Arkansas Nuclear One Unit 1 had indicated that the expected leakage resulting from a through-wall leak of a CRDM nozzle would yield very small quantities of boric acid residue which would not typically be visible from the mouse holes without additional inspection equipment.

During Refueling Outage 12, CR-3 proceeded to implement the planned visual inspection and subsequent repair activities as committed to in the CR-3 response to NRC Bulletin 2001-01. A bare metal visual inspection of the 69 RVH to CRDM nozzle interfaces was performed. The inspectors were VT-2 qualified with special qualifications and training related to CRDM nozzle leakage observation. The special qualifications and training used industry OE and images of leaking nozzles to sensitize inspectors to the type and quantity of boric acid crystal deposits indicative of CRDM through-wall leaks experienced at Oconee Nuclear Stations and Arkansas Nuclear One.

The results of the visual inspection indicated that there was one nozzle with a potential through-wall crack based on boric acid crystal accumulation. As a result of the visual inspection, the CRDM mounted on the affected nozzle was removed to support an ultrasonic test examination (UT) of the CRDM nozzle base material. The UT data indicated the presence of five recordable indications including two axially oriented cracks (flaw 3 and 4) that were through-wall, and extended from the bottom of the nozzle through and above the J-groove weld. These cracks originated at the weld-to-nozzle interface, propagated downward to the end of the nozzle, and upward through the weld into the annular space between the nozzle and the head. These two axial cracks were the source of leakage. These two cracks were then joined circumferentially (flaw 5) on the OD of the nozzle above the weld. The circumferential crack (flaw 5) above the weld extended about 90° and was approximately 50% through-wall. The UT identified one circumferential crack (flaw 1) below the weld. Flaw 1 extended for about 30° and was within 0.15 inch of the inside diameter (ID) (i.e., approximately 75% through-wall). Flaw 2 extended for about 195° and was through-wall. Note that flaw 2 had both axial and circumferential characteristics, extending from below the weld, through the weld and above the weld. The largest portion of the flaw was below the weld (approx. 130°). All five cracks were outside diameter (OD) initiated. No dye penetrant test (PT) of the J-groove weld was required since through-wall cracking of the nozzle base material was confirmed.

As provided in the CR-3 response to NRC Bulletin 2001-01, since through-wall cracking of CRDM nozzle #32 was confirmed by UT, an extent of condition of the cracking was performed using UT on eight nozzles where CRDMs were removed to facilitate nozzle repair or removed for CRDM replacement. The results of the additional UTs indicated that there was no cracking of the eight CRDM nozzles inspected. The eight locations selected provided reasonable assurance of bounding the extent of condition.

The UT results from the extent of condition examinations support the effectiveness of the visual inspection and the fact that accumulation of boric acid crystals did not impact the ability to discriminate between active CRDM nozzle leakage and other sources of leakage. The initial visual inspection and UT performed for extent of condition were in keeping with the commitment provided in the CR-3 response to NRC Bulletin 2001-01.

The CR-3 RVH was replaced in the Fall of 2003. In accordance with NRC Order EA-03-009, Paragraph IV.C(5)(a), the RVH was inspected during Refueling Outage 15. A bare metal visual examination was performed on 100 percent of the RPV head surface including 360° around each RPV head penetration nozzle. No evidence of boron or corrosive product was identified. A white flakey substance was observed on the surface of the head. The material was soft and non-adhering. A chemical analysis was performed on the substance, and it was determined not to be boron. This substance has been previously identified as originating from the head insulation package from above. Photographs were taken to document the discovery.

Conclusion

The CR-3 Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program meets the mandatory requirements of NRC Order EA-03-009 (as amended). The current requirements are based upon a "replaced" susceptibility ranking; however, the ranking is periodically re-calculated per the requirements of the Order to incorporate actual plant operating data. Based on the evaluation of the Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program, there is reasonable assurance that the Program will continue to adequately manage cracking in the VHP nozzles due to PWSCC so that system intended functions will be maintained consistent with the CLB for the period of extended operation.

B.2.6 THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

Program Description

The CR-3 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will be implemented as an augmented Inservice Inspection (ISI) Program to detect the effects of loss of fracture toughness due to thermal aging and/or neutron irradiation embrittlement of CASS reactor vessel internals. These inspections will be performed as augmented inspections to visual inspections already required by American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWB, Category B-N-3. Components within the scope of this augmented inspection Program include CASS reactor vessel internals components that have been determined to be potentially susceptible to thermal aging and/or are subjected to neutron fluence of greater than 10^{17} n/cm^2 (E > 1 MeV). Susceptibility to loss of fracture toughness due to thermal embrittlement is determined based on the criteria set forth in the May 19, 2000 letter from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Mr. Douglas Walters, Nuclear Energy Institute (NEI). For components deemed susceptible to loss of fracture toughness due to thermal embrittlement and/or neutron irradiation embrittlement, the Program allows for a component-specific evaluation, including a mechanical loading assessment to determine if the loading is compressive or low enough to preclude fracture. The Program evaluations and inspections will consider the recommendations of NUREG-1801, Section XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)."

The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will manage loss of fracture toughness due to thermal aging and/or neutron irradiation embrittlement in CASS reactor vessel internals components within the scope of License Renewal such that the system intended function is maintained through the extended period of operation. This Program will be implemented and required inspections completed and evaluated during the last 10-year ISI Interval prior to the period of extended operation. Inspections on potentially susceptible components will continue during the period of extended operation.

NUREG-1801 Consistency

The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new program that is consistent with NUREG-1801, Section XI.M13.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

This is a new program for Thermal Aging and Neutron Irradiation Embrittlement of CASS. There is no existing site-specific OE to validate the effectiveness of this Program at CR-3.

NUREG-1801 is based on industry OE through January 2005. Recent industry OE has been reviewed for applicability. More recent OE is captured through the normal OE review process where it is screened for applicability. This process will continue through the period of extended operation.

Conclusion

The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program includes augmented inspections which will be implemented as part of the CR-3 ISI Program. Based on the evaluation of this Program, there is reasonable assurance that, when implemented, the Program will adequately manage loss of fracture toughness so that system intended functions will be maintained consistent with the CLB for the period of extended operation.

B.2.7 FLOW-ACCELERATED CORROSION PROGRAM

Program Description

The Flow-Accelerated Corrosion (FAC) Program provides for prediction, detection, and monitoring of FAC in plant piping and other piping components so that timely and appropriate action may be taken to minimize the probability of experiencing a FAC-induced consequential leak or rupture. The FAC Program is based on the guidance provided in NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program," and includes conducting an analysis to determine critical locations, performing limited baseline inspections to determine the extent of thinning at these locations, performing follow-up inspections to confirm the predictions, and repairing or replacing the components as necessary.

NUREG-1801 Consistency

The FAC Program is an existing program consistent with NUREG-1801, Section XI.M17.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

Nuclear power plants have experienced pipe wall thinning in single-phase and two-phase high-energy piping systems which has been largely attributable to FAC. Specific CR-3 examples of OE include:

- Several components in the secondary plant systems have low margin to the limiting acceptance criteria and continue to experience FAC degradation. These components will require more frequent inspection, and will eventually require replacement.

- CR-3 FAC personnel attended the January 2008 CHECWORKS User Group meeting in which a presentation was made on the FAC entrance effect. EPRI Report TR1015072, "Flow-Accelerated Corrosion – The Entrance Effect," issued in November 2007, as well as the report recommendations, were discussed. In addition, utility representatives shared OE and new techniques for measuring wall thickness. This benchmarking OE demonstrates that CR-3 is staying abreast of FAC best practices.

The CR-3 FAC Program is based on NSAC-202L, and has evolved through monitoring of industry experience. The Program has been effective in its response to both industry and site-specific OE and provides an effective means of ensuring the structural integrity of high-energy carbon steel systems.

The NRC has audited industry programs based on the EPRI methodology at several plants and has determined that these activities can provide a good prediction of the onset of FAC so that timely corrective actions can be undertaken.

Conclusion

The Flow-Accelerated Corrosion Program provides reasonable assurance that wall thinning aging effects in piping components are adequately managed so that system intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

B.2.8 BOLTING INTEGRITY PROGRAM

Program Description

The Bolting Integrity Program addresses aging management requirements for bolting on mechanical components within the scope of License Renewal. The CR-3 Bolting Integrity Program utilizes industry recommendations and EPRI guidance that considers material properties, joint/gasket design, chemical control, service requirements, and industry and site OE in specifying torque and closure requirements. The Program relies on recommendations for a Bolting Integrity Program, as delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," and industry recommendations, as delineated in EPRI reports NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” and TR-104213, “Bolted Joint Maintenance & Applications Guide,” for pressure retaining bolting within the scope of License Renewal. Safety related bolting and closures inspections, monitoring/trending, and repair/replacement is performed under the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. In addition, both safety related and non-safety related pressure retaining bolting and closures inspection is performed under the External Surfaces Monitoring Program. The Program includes periodic inspections of high-strength structural bolting for cracking due to SCC. Degraded conditions are also subject to the Corrective Action Program.

NUREG-1801 Consistency

The Bolting Integrity Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section X1.M18.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the below-listed enhancement will be implemented:

<u>Program Elements Affected</u>

- **Scope of Program**
  The Bolting Integrity Program procedures will include guidance for torquing and closure requirements based on the EPRI documents endorsed by NUREG-1801.

- **Preventive Actions**
  1) The Bolting Integrity Program will identify and remove instances where

molybdenum disulfide lubricant is allowed for use in bolting applications in specific procedures and will add a specific prohibition against use of molybdenum disulfide lubricants in the CR-3 procedure for bolted connections.

2) The Bolting Integrity Program procedures will include guidance for torquing and closure requirements that include proper torquing of bolts and checking for uniformity of gasket compression after assembly.

3) The Bolting Integrity Program procedures will include guidance for torquing and closure requirements based on the guidance of EPRI 5067, "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Personnel," Volumes I and II.

- **Parameters Monitored/Inspected**
  The Bolting Integrity Program will include periodic ultrasonic testing (UT) examination of a representative sample of bolting identified as potentially having actual yield strength >150 ksi.

- **Detection of Aging Effects**
  1) The CR-3 Bolting Integrity Program will include a centralized procedure based on EPRI-5067 and will incorporate guidance regarding bolted joint leak tightness and pre-installation inspections consistent with the recommendations of this document.

  2) The Bolting Integrity Program will include periodic examination of a representative sample of bolting identified as potentially having actual yield strength >150 ksi. The Bolting Integrity Program includes periodic in situ UT examinations of these bolts for SCC. Alternately, bolting may be removed for surface examinations or replaced.

- **Monitoring and Trending**
  Examination of NSSS support high strength bolting for SCC will be performed concurrent with examinations of the associated supports with a minimum frequency of once per 10-year inservice inspection period.

- **Corrective Actions**
  1) The Bolting Integrity Program procedures will include guidance for torquing and closure requirements based on the recommendations of EPRI NP-5769, Volumes 1 and 2.

  2) Acceptance standards for examination of high strength structural bolting will utilize acceptance standards consistent with the recommendations of EPRI NP-5769.

Operating Experience

A review of plant specific OE associated with bolting has identified instances of leakage of bolted connections. Deficiencies noted include use of incorrect gasket material in flanged connections, and loss of preload resulting from relaxation of heat exchanger joints. Corrective actions were prescribed to address the application, including generic guidance in plant program documents as appropriate. Notably, the CR-3 Bolting Integrity Program includes an enhancement to provide centralized torquing and bolting procedures based on EPRI-5067 that will incorporate uniform guidance regarding gasket selection, bolted joint leak tightness and preinstallation inspections.

A review of current bolting practices found identified that molybdenum disulfide continues to be used in limited applications at CR-3, including applications in the primary system. While this thread compound has good lubrication properties on initial installation, industry OE has associated its use with the potential for stress corrosion cracking. Stress corrosion cracking of bolting using this compound has not been identified in CR-3 OE; nonetheless, the Bolting Integrity Program is being enhanced to discontinue its use.

The OE review shows that the CR-3 Bolting Integrity Program is continually upgraded based on industry experience, research, and routine program performance. The Program, through its continual improvement, assures the capability of mechanical bolting to support the safe operation of CR-3 throughout the extended period of operation.

Conclusion

Implementation of the CR-3 Bolting Integrity Program, with the enhancements identified above, will provide reasonable assurance that aging effects will be managed so that the systems and components within the scope of this Program will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.9 STEAM GENERATOR TUBE INTEGRITY PROGRAM

Program Description

The Steam Generator Tube Integrity Program is performed as part of the overall Steam Generator Integrity Program. The Steam Generator Tube Integrity Program is credited for aging management of the tubes, tube plugs, sleeves, tube supports, and the secondary-side components whose failure could prevent the steam generator from fulfilling its intended safety function. The Steam Generator Integrity Program is based on Technical Specification requirements, and meets the intent of NEI 97-06, "Steam Generator Program Guidelines."

The Steam Generator Tube Integrity Program manages aging effects by providing a balance of prevention, inspection, evaluation, repair, and leakage monitoring. Preventative measures are intended to mitigate degradation related to corrosion phenomena via primary-side and secondary-side water chemistry monitoring and control. Foreign material exclusion requirements are intended to inhibit wear degradation. The Steam Generator Tube Integrity Program provides the actions to be taken in response to finding foreign objects.

The Steam Generator Tube Integrity Program provides the requirements for inspection activities for the detection of flaws in tubing, plugs, sleeves, tube supports, and secondary-side internal components needed to maintain tube integrity. 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 not be exceeded during the next operating cycle. Primary-to-secondary leakage is continually monitored during operation.

The steam generators at CR-3 are scheduled to be replaced in 2009.

NUREG-1801 Consistency

The Steam Generator Integrity Program is an existing program consistent with NUREG-1801, Section XI.M19.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Steam Generator Integrity Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the Program meets regulatory, process, and procedure requirements; that qualified personnel are assigned as program managers and are given authority and responsibility to implement the Program; and that adequate resources are committed to Program activities.

The Steam Generator Integrity Program utilizes OE to promote the identification and transfer of lessons learned from both internal and industry events so that the knowledge gained can be used to improve nuclear plant safety and operations. Operating experience provides the methodology for receiving, processing, status reporting, screening, reviewing, evaluating, and taking preventive and corrective actions in response to OE information.

A review of NRC Generic Letters found that CR-3 steam generator tube inspection activities are consistent with NRC positions. Additionally, CR-3 has submitted an application for improved Technical Specifications consistent with NRC and industry adoption of improved steam generator Technical Specifications. Adoption of the improved Technical Specification requirements has been approved by the NRC. In addition, NRC Information Notices and Licensee Event Reports were reviewed for applicability to CR-3. Although all the OE was not directly applicable to the CR-3 steam generators, the underlying aging mechanisms were also reviewed and were found to be addressed by the CR-3 Steam Generator Tube Integrity Program. Also, INPO OE reports were reviewed for applicability to the CR-3 Steam Generator Tube Integrity Program. For those events that were directly related to the CR-3 steam generators, it was found that the CR-3 Steam Generator Tube Integrity Program addressed the concerns identified. For those events that were not directly related to the CR-3 steam generators, the underlying aging mechanisms were also reviewed. The aging mechanisms associated with the INPO OE were found to be addressed by the CR-3 Steam Generator Tube Integrity Program.

Examples of plant-specific OE include the following items:

- <u>Crack Indications in Steam Generator Alloy 600 Rolled Plugs (INPO OE)</u>: CR-3 uses a rotating coil probe to inspect 100% of the Alloy 600 rolled plugs in the hot and cold legs of OTSGs. OTSGs have previously experienced Alloy 600 rolled plug cracking, but the problems appeared to be confined to certain susceptible material heats. CR-3 did not have any of the susceptible plug material heats and had not found any crack indications in previous inspections. CR-3 normally inspects 100% of the existing Alloy 600 rolled plugs. Two of the four plugs with crack indications were found on the cold leg end of the tube. While it was expected that the plugs could develop cracks over time due to the material, they were not expected to occur on the cold leg end in this time frame based on vendor calculations The crack indications were all axially orientated and on the "heel" (non-pressure boundary) side of the rolled joint. The cause for the crack indications was determined to be the use of Alloy 600 material for the rolled plugs. Corrective actions included the repair of the four plugs by removing the old plug and installing either a new Alloy 690 rolled or welded plug. Future outages will continue to eddy-current inspect all remaining Alloy 600 rolled plugs. The safety significance was that the crack indications were found on the non-pressure end of the plug roll joint. The indications were all axial and would not have resulted in plug failure. There was no impact on plant, personnel, or public safety.

- Foreign Material in OTSG-B (CR-3 NCR):
  Eddy current testing conducted during the Fall 2007 outage discovered an unidentified object in one of the OTSG-B tubes that prevented complete inspection of the tube. Efforts to dislodge the object failed. It was recommended to plug the tube instead of expending significant dose to identify and retrieve the object. Additional investigation determined that the object was a piece of fuel assembly grid strap. Corrective actions include the prevention of future occurrences of loose fuel assembly grid strap fragments. Since the remainder of the tube did not have any significant degradation and the object was captured within the tube, it was concluded that the tube could be removed from service by plugging, and that no additional actions were necessary. The tube was plugged by installing AREVA roll plugs made from Alloy 690 material.

  Causal factors were:
  - Fuel assembly damage/failure such that a fragment of the fuel grid strap had separated and was lost into the RCS. The fuel damage was caused either by fuel handling failure or baffle plate wear.

  Corrective actions were:
  - Refueling planning and fuel handling techniques have been refined and improved over the last several cycles to limit the potential interactions as the core is loaded. Also, the use of the improved cladding will result in less warpage and twist, leading to better loading characteristics. Baffle plate wear is a recently identified phenomenon that is the subject of a combined AREVA/Progress Energy root cause review. Corrective actions to prevent baffle plate wear damage will be derived through existing efforts. Eddy current testing of the unobstructed sections of the tube were conducted prior to plugging to ensure that a tube stabilizer was not required. No degradation other than some minor wear at the 6th tube support plate was found. Therefore, stabilization of the tube was found to not be necessary.

Inspections performed during CR-3 refueling outages indicate the following active degradation mechanisms in the current steam generators:

- Upper bundle axial outside diameter stress corrosion cracking/intergranular attack (ODSCC/IGA),
- Axial ODSCC/IGA in the upper tube sheet crevice,
- Axial and circumferential primary water stress corrosion cracking in roll expansion regions,
- General volumetric degradation,
- Wear at tube support locations,
- Volumetric degradation in the first span Alternate Repair Criteria region of OTSG-B, and
- Tube end cracks confined exclusively to the depth of the tubesheet clad.

The steam generators at CR-3 are scheduled to be replaced in 2009.

This OE review shows that the Steam Generator Tube Integrity Program is continually upgraded based on industry experience, external and internal assessments, and routine program performance, and has provided an effective means of ensuring steam generator tube integrity. The overall effectiveness of the Steam Generator Integrity Program is supported by the OE for systems, structures, and components; no tube integrity-related degradation has resulted in loss of component intended function.

Conclusion

Continued use of the Steam Generator Tube Integrity Program, as implemented by the Steam Generator Integrity Program will provide reasonable assurance that applicable aging effects are managed such that the steam generator components/commodities within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.10 OPEN-CYCLE COOLING WATER SYSTEM PROGRAM

Program Description

The CR-3 Open Cycle Cooling Water (OCCW) System Program relies on implementation of the recommendations in NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," and the guidance in its supplement, "Service Water Problems Affecting Safety-Related Equipment (Generic Letter 89-13, Supplement 1)," to ensure that the effects of aging associated with the Nuclear Services and Decay Heat Seawater System will be managed for the period of extended operation. The Program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, and silting in the Nuclear Services and Decay Heat Seawater System or structures and components serviced by the System.

The OCCW System Program addresses the Nuclear Services and Decay Heat Seawater System, as well as the raw water side of the Decay Heat Closed Cycled Cooling and the Nuclear Service Closed Cycle Cooling System heat exchangers.

NUREG-1801 Consistency

The OCCW System Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M20.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

<u>Program Elements Affected</u>

- **Preventive Actions**
  1) The Nuclear Services and Decay Heat Seawater System Pumps will be included in a periodic inspection/rebuild program. This program will be initiated during the current license period and will inspect one or more pumps prior to the period of extended operation.

  2) The Nuclear Services and Decay Heat Seawater System Discharge Conduits will be subject to inspection / evaluation subsequent to the steam generator replacement project, but prior to the period of extended operation. The results

from this activity will determine the extent of activities required during the period of extended operation to support the intended function of these components.

3) Periodic maintenance activities will be established for Nuclear Services and Decay Heat Seawater expansion joints RWEJ-3, -4, -5, -6, -7, -8, -9, and -10.

Operating Experience

The seawater environment associated with the Nuclear Services and Decay Heat Seawater System is an aggressive environment. A review of plant OE identifies instances of degradation including:

- Macro-fouling in the Nuclear Services and Decay Heat Seawater and Decay Heat Closed Cycle Cooling heat exchangers by loose marine shells,
- Tube plugging activities in the Nuclear Services and Decay Heat Seawater heat exchangers,
- Degradation of protective lining in piping spools,
- Minor system leakage, and
- Cyclone separator and strainer fouling.

The CR-3 AMR methodology predicts aging effects consistent with plant OE. The OCCW System Program incorporates an extensive range of inspection and maintenance activities to ensure system intended functions are maintained.

Past inspections performed at CR-3 have revealed multiple instances of selective leaching in equipment in seawater applications. The discharge heads on the seawater pumps in the Nuclear Services and Decay Heat Seawater System have been replaced multiple times due to damage from decarbonization of the cast iron. Action Requests document a failure of an aluminum bronze check valve hinge pin stop due to "dealloying and environmentally assisted stress" in valve RWV-34. "Dealloying" of aluminum bronze cladding on seawater heat exchangers has also been identified. Typically the Selective Leaching of Materials Program would be specified to manage the aging effect of selective leaching, using a one-time inspection of a representative sampling of components. However, since selective leaching has already been identified as an existing aging mechanism in these applications, the Selective Leaching of Materials Program will not be specified, rather the aging effect will be managed by periodic inspections under the OCCW System Program.

Conclusion

The OCCW System Program has been effective at managing aging effects for safety related components wetted by raw water. The Program has been improved through evaluation of site and industry OE. Following enhancement, the continued use of the OCCW System Program will provide reasonable assurance that the aging effects will be managed such that the applicable components will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.11 CLOSED-CYCLE COOLING WATER SYSTEM PROGRAM

Program Description

The Closed-Cycle Cooling Water (CCCW) System Aging Management Program addresses aging management of components in or cooled by CR-3 CCCW Systems, including the Decay Heat Closed Cycle Cooling Water System, the Nuclear Services Closed Cycle Cooling Water System, the Secondary Services Closed Cycle Cooling Water System, and the Industrial Cooling System. The AMP scope also includes components in Control Complex Chilled Water and Appendix R Chilled Water Systems, the diesel engine Jacket Coolant System, and the Instrument Air System. These cooling systems are closed cooling loops with controlled chemistry, consistent with the NUREG-1801 description of a closed-cycle cooling water system. This Program relies on maintenance of system corrosion inhibitor concentrations within specified limits of the EPRI Closed Cooling Water Chemistry Guidelines to minimize corrosion. Surveillance testing and inspection in accordance with standards in the EPRI report for CCCW systems is performed to evaluate system and component performance. These measures will ensure that the CCCW systems and components serviced by the CCCW systems will continued to perform their intended functions acceptably.

NUREG-1801 Consistency

The CCCW System Program is an existing program consistent with NUREG-1801, Section XI.M21, with exceptions.

Exceptions to NUREG-1801

<u>Program Elements Affected</u>

- **Parameters Monitored/Inspected**
  1) The Secondary Services Closed Cycle Cooling Water System and Instrument Air System closed cooling pumps are not subject to a formal test program. However, the ability of these systems to maintain adequate flow rates and heat transfer is verified on an ongoing basis by routine operation of this system in support of the operating unit. Industrial Cooling System pumps are likewise not subject to a formal test program; however, this system is only in scope for spatial interaction, and pump flow rate is not relevant to the spatial interaction intended function.

  2) The Secondary Services Closed Cycle Cooling Water System and Instrument Air System heat exchangers are not subject to a formal performance monitoring program. However, acceptable thermal/hydraulic performance of these systems is verified on an ongoing basis by operation of the systems in support of unit operations. Likewise, Industrial Cooling System heat exchangers are not subject

to a formal test program, however, this system is in scope only for spatial interaction and heat transfer is not relevant to the intended function.

Enhancements

None.

Operating Experience

Plant-specific OE review associated with components cooled by CCCW noted a number of events associated with fouling and corrosion of the Nuclear Services Closed Cycle Cooling Water Heat Exchangers. These include fouling of tubes, tube leakage, and dealloying of the aluminum bronze cladding on the tubesheets. These deficiencies were associated with the tube side of the heat exchangers exposed to seawater. The associated aging effects credit the Open Cycle Cooling Water System Program for aging management including management of selective leaching of the Nuclear Services Closed Cycle Cooling Water Heat Exchanger tubesheet cladding.

Other items were noted associated with isolated events, including instances of leakage and low flow. A conductivity excursion was also noted in the closed cycle cooling water portion of the Industrial Cooling System. These events do not involve corrosion related to CCCW environments, or provide indication of generic weaknesses in the CCCW System Program. The Program is continually upgraded based on industry experience, external and internal assessments, and routine program performance, and has provided an effective means of mitigating loss of material, cracking, and reduction of heat transfer effectiveness.

Conclusion

Continued implementation of the CR-3 CCCW System Program provides reasonable assurance that the aging effects will be managed so that the systems and components within the scope of this Program will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.12 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD HANDLING SYSTEMS PROGRAM

Program Description

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program provides for inspection of the following cranes:

Structure	Crane(s)
Reactor Building	RB Polar Crane Reactor Vessel Tool Handling Jib Crane 5-Ton Jib Crane CRDM Jib Crane Main Fuel Handling Bridge Crane
Auxiliary Building	120-Ton Fuel Handling Area Crane Spent Fuel Pit Missile Shield Crane Spent Fuel Pool Handling Bridge Crane
EFW Pump Building	EFW Pump Building 3-Ton Crane
Circulating Water Intake Structure	Intake Gantry Crane

The inspections monitor structural members for the absence of signs of corrosion, other than minor surface corrosion, and crane rails for abnormal wear. The inspections are performed every refuel cycle for cranes inside the Reactor Building. Cranes outside the Reactor Building are inspected every two years.

NUREG-1801 Consistency

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M23.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

Program Elements Affected

- **Scope of Program**
  Revise administrative controls to include all cranes that are within the scope of License Renewal.

- **Parameters Monitored/Inspected**
  1) Revise administrative controls to require notifying the responsible engineer of unsatisfactory inspection results involving loss of material, including loss of material owing to wear of rails, for cranes in scope of License Renewal.

  2) Revise administrative controls to include all cranes that are within the scope of License Renewal.

- **Detection of Aging Effects**
  1) Revise administrative controls to clarify that crane rails are to be inspected for abnormal wear, and that members to be inspected for cracking include welds, and

  2) Revise administrative controls to specify frequency of inspections for in-scope cranes to be every refueling outage for cranes inside the RB and every two years for cranes outside the RB.

Operating Experience

Based on a review of plant history, CR-3 has performed periodic inspections of cranes, has utilized assessments to identify programmatic deficiencies and improvements, and has tracked the resolutions by means of the Corrective Action Program. A review of crane inspections and assessments identified no evidence of corrosion of structural members or wear of rails. Nevertheless, the aging effect of corrosion has been found for other carbon steel components for similar environments; and, therefore, monitoring for these aging effects is appropriate. Crane monitoring programs are continually being upgraded based upon industry and Progress Energy plant experience. In addition, self-assessments are periodically scheduled at CR-3 and on other Progress Energy nuclear plants. A self-assessment was performed in 2006 to review Control of Heavy Loads for all Progress Energy plants. For CR-3, it was determined that there was no dedicated rigging engineer. Another assessment was performed in 2008, and assessment findings are being addressed. In July 2008, a system engineer for cranes was assigned at CR-3. In addition, a qualified structural engineer will perform any specific structural inspections. The results of this proactive approach to the operation and management of cranes validate the effectiveness of the procedures that implement the Inspection of Overhead Heavy Load and Light Load Handling Systems Program. Based on these results, OE provides evidence that the Program activities will continue to ensure the integrity of the cranes within the scope of License Renewal.

Conclusion

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program, with the enhancements identified above, will provide reasonable assurance that the aging effects of corrosion of structural components and crane rail wear are adequately managed so that the intended functions of cranes within the scope of License Renewal are maintained during the period of extended operation.

B.2.13 FIRE PROTECTION PROGRAM

Program Description

The CR-3 Fire Protection Program provides aging management of the fire protection components including penetration seals; expansion joints; fire barrier walls, ceilings, and floors; fire rated doors; Diesel Fire Service Pump fuel oil supply lines; fire barrier assemblies such as fire wraps on trays, pipes, and conduits; and the Halon system used for the Control Complex cable spreading room. The Program is implemented through various plant procedures and will effectively manage the aging effects associated with the subject components such that the intended functions of applicable components will be maintained through the period of extended operation.

NUREG-1801 Consistency

The CR-3 Fire Protection Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M26, with exceptions.

Exceptions to NUREG-1801

Program Elements Affected

- **Parameters Monitored/Inspected**
  NUREG-1801 recommends periodic visual inspection and function testing of Halon suppression systems be performed at least once every six months. The CR-3 Fire Protection Program performs only functional testing of the Halon system once per 18 months. Although the functional testing frequency exceeds the recommended frequency, it is sufficient to ensure the Halon system will perform its intended function. The exception is acceptable based on the Halon system being located within the cable spreading room, a conditioned air environment within the CR-3 control complex. As noted in NUREG-1801, corrosion of external surfaces is not expected in controlled air environments.

- **Detection of Aging Effects**
  1) The exception discussed under Parameters Monitored/Inspected above also affects the Detection of Aging Effects program element.

  2) NUREG-1801 recommends visual inspection of walls, ceilings, and floors be performed at least once every refueling outage. The CR-3 Fire Protection Program performs visual inspection of walls, ceilings, and floors on a frequency commensurate with the safety significance of the structure and its condition but not to exceed 10 years. The exception is acceptable based on using an existing procedure for structural inspections and that CR-3 OE has not detected degradation of fire barrier walls, ceilings, and floors which has resulted in a loss

of fire barrier function. The structural inspections are sufficient to detect gradual degradation of the fire barrier walls, ceilings, and floors. The frequency of inspections would be increased depending on the as-found condition.

- **Monitoring and Trending**
  The exception for the visual inspection of walls, ceilings, and floors discussed under Detection of Aging Effects above also affects the Monitoring and Trending program element.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

Program Elements Affected

- **Scope of Program**
  The CR-3 Fire Protection Program will be enhanced to include a procedure for periodic inspections of fire barrier walls, ceilings, and floors.

- **Parameters Monitored/Inspected**
  1) The Fire Protection Program procedure for periodic inspections of penetrations seals will be enhanced to include inspections for seal separation from walls and components, separation of layers of material, rupture and puncture of seals which are directly caused by increased hardness, and shrinkage of seal material due to weathering.

  2) The CR-3 Fire Protection Program procedure for the annual inspection of fire doors will be enhanced to include visual inspection for loss of material (corrosion) with an acceptance criterion of absence of signs of corrosion other than minor surface corrosion.

- **Detection of Aging Effects**
  The Fire Protection Program administrative controls for periodic inspections of penetrations seals and fire doors will be enhanced to specify a minimum qualification requirement for personnel performing visual inspections.

- **Monitoring and Trending**
  The program enhancements described above under the Parameters Monitored/Inspected program element are necessary for consistency with this NUREG-1801 program element.

- **Acceptance Criteria**
  The CR-3 Fire Protection Program procedures for periodic inspections of concrete fire barrier walls, ceilings, and floors will be enhanced to add a step to

notify Fire Protection of any deficiencies having the potential to adversely affect the fire barrier function of concrete walls, ceilings, and floors.

Operating Experience

The Fire Protection Program is maintained in accordance with CR-3 engineering program requirements and managed in accordance with plant administrative controls. The operating history and assessment results for the Program show it is an effective means of ensuring the preservation from fire of the safe shutdown capability of CR-3. The CR-3 Fire Protection Program is continually improving based on both industry and plant-specific OE. Industry OE is incorporated into the Fire Protection Program via the Operating Experience Program and as a result of NRC generic communications. The CR-3 Program benefits from benchmarking other Progress Energy plants as well as other industry plants. Plant-specific OE is also used to improve the Fire Protection Program through use of the Corrective Action Program and program assessments. The Corrective Action Program is being used to identify adverse conditions, track corrective actions, and improve the Fire Protection Program.

The Fire Protection Program requires a Triennial Inspection performed by NRC personnel and biennial self-assessments. The Triennial Inspection includes reviews of the CR-3 Corrective Action Program and the Operating Experience Program.

Conclusion

Following program enhancement, implementation of the CR-3 Fire Protection Program will ensure the effects of aging associated with the fire protection related components will be adequately managed such that there is reasonable assurance that their intended functions will be performed consistent with the CLB through the period of extended operation.

B.2.14 FIRE WATER SYSTEM PROGRAM

Program Description

The Fire Water System Program includes system pressure monitoring, wall thickness evaluations, periodic flow and pressure testing in accordance with applicable National Fire Protection Association (NFPA) commitments and periodic visual inspection of overall system condition. These activities provide an effective means to determine whether corrosion and biofouling are occurring. Inspections of sprinkler heads assure that corrosion products that could block flow of the sprinkler heads are not accumulating. These measures will allow timely corrective action in the event of system degradation to ensure the capability of the water-based Fire Suppression System to perform its intended function.

NUREG-1801 Consistency

The Fire Water System Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M27.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

Program Elements Affected

- **Parameters Monitored/Inspected**
  Revise the Program documents to incorporate a requirement to perform one or a combination of the following two activities:

    a) Implement periodic flow testing consistent with the intent of NFPA 25.

    b) Perform wall thickness evaluations to verify piping is not impaired by pipe scale, corrosion products, or other foreign maternal. For sprinkler systems, this may be done by flushing, internal inspection by removing one or more sprinkler heads, or by other obstruction investigation methods, such as, technically proven ultrasonic and X-ray examination, that have been evaluated as being capable of detecting obstructions.

These inspections will be performed before the end of the current operating term. The results from the initial inspections will be used to determine inspection intervals thereafter during the period of extended operation.

- **Detection of Aging Effects**
  1) Revise the Program documents to incorporate a requirement to perform internal inspections of system piping at representative locations as required to verify that loss of material due to corrosion has not impaired system intended function. Alternately, non-intrusive inspections (eg., UT exams) can be used to verify piping integrity. These inspections will be performed before the end of the current operating term. The results from the initial inspections will be used to determine inspection intervals thereafter during the period of extended operation.

  2) Enhance the Program to perform a visual inspection of yard fire hydrants annually consistent with the intent of NFPA 25 to ensure timely detection of signs of degradation, such as corrosion.

  3) Enhance the Program, consistent with the intent of NFPA 25, to either replace the sprinkler heads prior to reaching their 50-year service life or perform field service testing of representative samples from one or more sample areas by a recognized testing laboratory. Subsequent test intervals will be based on test results.

Operating Experience

CR-3 OE includes considerable maintenance associated with the Fire Water Storage Tanks. Problems noted with these tanks include corrosion of tank vents and platforms, corrosion of weld heat affected zones, undercutting and arc strikes from original construction, and coating deficiencies inside and out. To address these issues, both tanks have been reconditioned, including replacing degraded vents and appurtenances, draining, cleaning and repairing internal surfaces, and applying protective coatings inside and out. Preventive maintenance activities have been implemented for annual inspections of the tanks exterior, and inspections of the internal surfaces on a five year frequency.

The CR-3 Fire Protection Program, which includes the Fire Water System Program activities described herein, is maintained in accordance with the Corporate QA Program, and subject to regular reviews and assessment. A review of the last three triennial self-assessment inspection reports confirms that assessments are being done, and that the program is subject to continual review and improvement. Specific results from the last triennial review report identify weaknesses in specifying compensatory actions, allocating fire protection resources for outage demands and site initiatives, errors in Fire Protection administrative documents, and documenting program improvement opportunities identified through benchmarking. Corrective actions for these items are being addressed and tracked to resolution in the Corrective Action

Program. The report concluded that the CR-3 Fire Protection Functional Area is effectively executing and fulfilling its requirements.

Conclusion

Continued implementation of the CR-3 Fire Water System Program, including the enhancements identified above, will assure that the components/commodities associated with the water-based Fire Suppression System will perform their intended functions for the period of extended operation.

B.2.15 ABOVEGROUND STEEL TANKS PROGRAM

Program Description

The Aboveground Steel Tanks Program manages aging effects of loss of material for external surfaces and inaccessible locations of the Fire Service Water Storage Tanks and the Condensate Storage Tank. These tanks are constructed of carbon steel. This Program relies on periodic system walkdowns and planned preventive maintenance inspections to monitor the condition of these tanks. This Program includes an assessment of the condition of 1) tank surfaces protected by a coating, although the paint is not credited to perform a preventive function, and 2) the sealing of the concrete foundation. For inaccessible surfaces, such as the tank bottom, thickness measurements will be performed from inside the tank to assess the tank bottom condition. Performing these inspections of tank bottoms ensures that degradation or significant loss of material will not occur in inaccessible locations. The frequency of tank bottom volumetric inspections will be based on the findings of inspections performed prior to the period of extended operation.

NUREG-1801 Consistency

The Aboveground Steel Tanks Program is a new program consistent with NUREG-1801, Section XI.M29.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Aboveground Steel Tanks Program is a new program, and as such no OE exists for a demonstration of program effectiveness. Currently, periodic inspections are performed to determine the material condition of the applicable tanks. Holes were discovered in the Condensate Storage Tank bottom. After the holes were repaired, an acceptable nondestructive examination and visual inspection were performed. The Fire Service Water Storage Tanks have had concerns about their external condition. These included broken grout around the tank perimeter, chalking of paint, and corrosion of the roof vent.

Conclusion

Implementation of the Aboveground Steel Tanks Program will assure the effects of aging associated with the tanks will be adequately managed so that there is reasonable assurance that their intended functions will be maintained consistent with the CLB during the period of extended operation.

B.2.16 FUEL OIL CHEMISTRY PROGRAM

Program Description

Fuel oil quality is maintained by the purchase of quality fuel and establishment of a diesel fuel oil testing program to implement required testing of both new and stored fuel oil. The existing Fuel Oil Chemistry Program includes sampling and testing requirements and acceptance criteria in accordance with applicable American Society for Testing Materials (ASTM) Standards identified in CR-3 Technical Specification surveillance requirements and chemistry program procedures for fuel oil testing. Exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by verifying the quality of new fuel oil and the addition of a biocide, a stabilizer, and corrosion inhibitors. Subsequently, periodic sampling is performed to verify that the tanks are free of water, particulates, and biological growth. The effectiveness of the Program is verified by periodic tank inspections to ensure that significant degradation is not occurring so that the component intended function will be maintained during the extended period of operation.

The tanks within the scope of License Renewal and addressed by the Program are the Emergency Diesel Fuel Oil Storage Tanks, the Emergency Diesel Fuel Oil Day Tanks, the Diesel-Driven Fire Pump Fuel Oil Storage Tanks, and the Diesel-Driven Emergency Feedwater Pump Fuel Oil Storage Tank.

NUREG-1801 Consistency

The Fuel Oil Chemistry Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M30, with exceptions.

Exceptions to NUREG-1801

Program Elements Affected

- **Scope of Program**
  1) CR-3 uses ASTM Standard D2709, not ASTM D1796. Both of these standards are tests for water and sediment. The Progress Energy Corporate specification for fuel oil indicates that the acceptance criteria for both of these standards are the same. Based upon the similarities of tested property and acceptance criteria, this exception to NUREG-1801 is justified.

  2) CR-3 uses ASTM D2276, not ASTM D6217. Both ASTM D2276 and ASTM D6217 are tests for particulate contamination. Based upon the similarities of tested property (particulates), this exception to NUREG-1801 is justified. In addition, ASTM Standard 2276 is set forth in the CR-3 Technical Specifications as the required standard for determining particulate contamination.

- **Preventive Actions**
  1) Water is not periodically drained from the bottom of Diesel-Driven Emergency Feedwater Pump Fuel Oil Storage Tank. However, this tank is recirculated during quarterly sampling through a filter-separator water coalescer designed to remove entrained fluids such as water. Based on entrained fluids such as water being removed quarterly by the filter-separator, this exception to NUREG-1801 is justified.

  2) The Diesel-Driven Fire Pump Fuel Oil Storage Tanks are not periodically drained of water. Sampling of the tanks to determine water buildup in the tank bottom is performed quarterly. If water exceeding the limit is found, actions are taken either to remove the water or replace the fuel. Based on quarterly sampling for water buildup, this exception to NUREG-1801 is justified.

- **Parameters Monitored/Inspected**
  1) CR-3 uses the guidance in ASTM D 2276-91 for determination of particulates. The filter used is a smaller pore size than that in the ASTM Standard recommended in NUREG-1801. Since a filter with a smaller pore size traps more particulates than one with a larger pore size, this test provides more conservative results than the one recommended by NUREG-1801. Therefore, this exception to NUREG-1801 is justified.

  2) CR-3 uses ASTM Standard D2709, not ASTM D1796. Both of these standards are tests for water and sediment. The Progress Energy Corporate specification for fuel oil indicates that the acceptance criteria for both of these standards are the same. Based upon the similarities of tested property and acceptance criteria, this exception to NUREG-1801 is justified.

- **Detection of Aging Effects**
  1) Multi-level sampling is not performed on the Diesel-Driven Emergency Feedwater Pump Fuel Oil Storage Tank and the Diesel-Driven Fire Pump Fuel Oil Storage Tanks. Multi-level sampling is performed for larger fuel oil tanks. Prior to sampling the Diesel-Driven Emergency Feedwater Pump Fuel Oil Storage Tank, a two-volume recirculation of the tank is verified. This provides for sampling of mixed contents. In addition, multi-level sampling is performed on the Emergency Diesel Fuel Oil Storage Tanks, which can be the source of fuel oil for Diesel-Driven Fire Pump Fuel Oil Storage Tanks. Prior to sampling the Diesel-Driven Fire Pump Fuel Oil Storage Tanks, discharge piping flow is established to provide a mixed sample. Based on the above factors, the exception to NUREG-1801 multi-level sampling is justified.

  2) Routine sampling is not performed on the Emergency Diesel Fuel Oil Day Tanks. Day tank fuel volumes are cycled and refreshed each month during the Emergency Diesel surveillance runs. Prior to these surveillance runs, fuel oil

from the bottom of the day tanks is removed and returned to the Emergency Diesel Fuel Oil Storage Tanks. Based upon the above, this exception to NUREG-1801 is justified.

3) Ultrasonic testing (UT) measurements of tank wall thickness would only be performed on in-scope tanks if visual inspection reveals significant internal damage due to loss of material. This exception is justified because if visible damage on the internal surface is not identified, then there is no compelling reason to perform UT measurements. Prior to the period of extended operation, all of the subject tanks will have had a periodic inspection of their internal surfaces. With the exception of the Emergency Diesel Fuel Oil Storage Tanks, these are above ground tanks located inside protected structures; and their external surfaces will be monitored during the period of extended operation in accordance with the License Renewal External Surfaces Monitoring Program. For the in-scope tanks, if there is no significant corrosion identified in internal and external inspections, then additional UT inspections are not warranted. Based on the above, this exception is justified.

- **Acceptance Criteria**
  1) CR-3 uses the guidance in ASTM D 2276-91 for determination of particulates. The filter used is a smaller pore size than that in the ASTM Standard recommended in NUREG-1801. Since a filter with a smaller pore size traps more particulates than one with a larger pore size, this test provides more conservative results than the one recommended by NUREG-1801. Therefore, this exception to NUREG-1801 is justified.

  2) CR-3 uses ASTM Standard D2709, not ASTM D1796. Both of these standards are tests for water and sediment. The Progress Energy Corporate specification for fuel oil indicates that the acceptance criteria for both of these standards are the same. Based upon the similarities of tested property and acceptance criteria, this exception to NUREG-1801 is justified.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

<u>Program Elements Affected</u>

- **Preventive Actions**
  1) Adjust the inspection frequency for the Diesel-Driven Emergency Feedwater Pump Fuel Oil Storage Tank to ensure an inspection is performed prior to the period of extended operation.

  2) Inspect the internal surfaces of the Diesel-Driven Fire Pump Fuel Oil Storage

Tanks and develop a work activity to periodically inspect the internal surfaces of these tanks. Prior to the inspection, remove fuel, water, and sediment as much as practical due to limited access. UT or other non-destructive examination (NDE) will be performed if visual inspection proves inadequate or indeterminate.

- **Detection of Aging Effects**
  1) Adjust the inspection frequency for the Diesel-Driven Emergency Feedwater Pump Fuel Oil Storage Tank to ensure an inspection is performed prior to the period of extended operation.

  2) Inspect the internal surfaces of the Diesel-Driven Fire Pump Fuel Oil Storage Tanks and develop a work activity to periodically inspect the internal surfaces of these tanks. Prior to the inspection, remove fuel, water, and sediment as much as practical due to limited access. UT or other NDE will be performed if visual inspection proves inadequate or indeterminate.

Operating Experience

The Fuel Oil Chemistry Program is implemented and maintained in accordance with the general requirements for chemistry programs. This provides assurance that the Program is effectively implemented to meet regulatory, process, and procedure requirements. Qualified personnel are assigned as program managers and are given authority and responsibility to implement the Program. In addition, adequate resources are committed to Program activities. Specific examples of OE include:

- Diesel fuel oil particulates are increasing. The problem was related to the mixing of diesel fuels and the lack of a fuel stabilizer. In November 2007, while in a refueling outage, the Emergency Diesel Generator Fuel Oil Storage Tanks were off-loaded and the fuel was filtered through a very fine clay media filtration process. The particulates for both tanks were reduced significantly to about 1mg/L or less. While this cleaned the fuel, it was noted this would not prevent the recurrence of particulate formation without the use of a fuel stabilizer.

  Southwest Research Institute (SWRI) was contracted to help resolve the diesel fuel particulate issue; this same organization provided testing and recommendations in 2007 to help resolve the fuel particulate issues that were occurring at that time. SWRI previously recommended CR-3 no longer accept high sulfur diesel fuel for use onsite, clay filter the fuel during the refueling outage, and use a fuel stabilizer. The investigation is ongoing, with CR-3 currently using a fuel stabilizer

- The Diesel Driven Fire Pump Fuel Oil Storage Tanks have an increasing trend on particulates. The particulate levels are at 6.93 mg/l, just below the administrative limit of 7.0. The action was to replace the fuel oil in the tanks.

A review of plant condition reports and OE demonstrates that the Fuel Oil Chemistry Program at CR-3 is critically monitored, and continually improving. Based on these results, the OE review provides evidence that the Fuel Oil Chemistry Program practices have thus far ensured the integrity of the subject components wetted by fuel oil.

Conclusion

With the addition of the proposed enhancements, the Fuel Oil Chemistry Program will provide reasonable assurance that aging effects will be managed such that the applicable components will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.17 REACTOR VESSEL SURVEILLANCE PROGRAM

Program Description

The Reactor Vessel Surveillance Program manages the reduction of fracture toughness of the reactor vessel beltline materials due to neutron embrittlement. As part of the Reactor Vessel Surveillance Program, CR-3 participates in the Master Integrated Reactor Vessel Surveillance Program (MIRVP) and monitors fluence using periodic fluence projections and alternative dosimetry, consistent with the intent and scope of 10 CFR 50, Appendix H. The Reactor Vessel Surveillance Program evaluates the effect of neutron embrittlement by projecting upper-shelf energy (USE) and pressurized thermal shock (PTS) reference temperatures for all reactor materials with projected neutron exposure greater than 10^{17} n/cm^2 (E > 1.0 MeV) after 60 years of operation and with the development of pressure-temperature limit curves. Embrittlement information is obtained in accordance with NRC Regulatory Guide 1.99, Revision 2, chemistry tables and with surveillance capsules, which have provided credible data for the current operating period and for the period of extended operation. The surveillance program design, capsule withdrawal schedule, and evaluation of test results are in accordance with ASTM E 185-82. Select tested specimens are stored for future use, if needed. The Reactor Vessel Surveillance Program controls the remaining capsules so that withdrawal of the remaining capsules is managed through the MIRVP and has been approved by the NRC. The Reactor Vessel Surveillance Program manages the steps taken if reactor vessel exposure conditions are altered, such as, the review and updating of 60-year fluence projections to support the preparation of new pressure-temperature limit curves and pressurized thermal shock reference temperature calculations.

NUREG-1801 Consistency

The CR-3 Reactor Vessel Surveillance Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M31, with exception.

Exceptions to NUREG-1801

<u>Program Elements Affected</u>

- **Program Element 4**
  NUREG-1801 Program Element 4 states that pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage and that these specimens are saved for future reconstitution use in case the surveillance program is reestablished. Some MIRVP tested specimens have been disposed instead of being retained for future reconstitution use. However, sets of specimens from beltline weld heats at CR-3 are permanently archived at Point Beach Nuclear Plant. Program Element 4 states that the specimens from beltline

weld heats in CR-3 are being permanently saved for future use; the CR-3 Reactor Vessel Surveillance Program meets the intent of the program element.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

Program Elements Affected

- **Program Element 1**
  Enhance the Program to ensure that neutron exposure conditions of the reactor vessel remain bounded by those used to project the effects of embrittlement to the end of the 60-year extended license period.

- **Program Element 4**
  Establish formalized controls for the storage of archived specimens to ensure availability for future use by maintaining the identity, traceability, and recovery of the archived specimens throughout the period of storage.

- **Program Element 6**
  Refer to the enhancement for projecting the effects of embrittlement discussed under Program Element 1 above.

- **Program Element 7**
  Refer to the enhancement for projecting the effects of embrittlement discussed under Program Element 1 above.

- **Program Element 8**
  Refer to the enhancement for projecting the effects of embrittlement discussed under Program Element 1 above.

Operating Experience

The Reactor Vessel Surveillance Program is described in FSAR Section 4.4.5 and has provided materials data and dosimetry for the monitoring of irradiation embrittlement since plant startup. CR-3 participates in the MIRVP and monitors reactor vessel fluence using periodic fluence updates and alternative dosimetry.

Surveillance capsules have been withdrawn during the period of current operation, and the credible data from these surveillance capsules have been used to verify and predict the performance of CR-3 reactor vessel beltline materials with respect to neutron embrittlement. Calculations have been performed as required to project the degree of reduction in USE and PTS reference temperature that is expected to result from projected neutron exposure in the future, including 60-year projections.

Pressure/temperature limits have been imposed on operational parameters at CR-3 to assure that the vessel is operated within required safety margins. Even though the capsules remaining inside the CR-3 reactor vessel are not expected to provide meaningful data for CR-3, meaningful data from five capsules containing the CR-3 limiting weld materials have already been pulled and tested thereby completing the requirement for capsule withdrawals in accordance with ASME E 185-82 for 60 years for CR-3. Some of these limiting weld materials were exposed to fluences approximately equal to the 54 EFPY CR-3 reactor vessel projected peak fluence.

A review of NRC Information Notices, Bulletins, and Generic Letters, NRC Agencywide Document Access Management System (ADAMS), INPO OE database, and other relevant sources were performed for applicable OE. The following items were identified:

- The MIRVP was developed as a response to the failure of surveillance capsule holder tubes in several B&W-supplied reactor vessels and the necessity to obtain fracture toughness data for irradiated weld metals to ensure continued licenseability. For these plants, which include CR-3, the original Reactor Vessel Surveillance Programs could not provide sufficient material data and dosimetry to monitor embrittlement. The MIRVP approach is effective because it satisfies the requirements of 10 CFR 50, Appendix H, which states that an integrated surveillance program the participating plants must have similar design and operating features, an adequate dosimetry program, and an adequate arrangement for data sharing between plants. The MIRVP provides sufficient material data to meet the ASTM E-185-82 capsule program requirements for monitoring embrittlement. The NRC staff evaluated the basis for the integrated program concept, determined the MIRVP to be acceptable, and approved BAW-1543, Revision 3, by letter dated June 11, 1991. This letter concluded that the program met the applicable criteria from 10 CFR 50, Appendix H. BAW-1543, Revision 4 and its supplements have been frequently issued to incorporate program improvements, such as updated capsule withdrawal schedules, revised capsule status, and updated fluence projections.

- BAW-1543, Revision 4, Supplement 5 was issued because BAW-1543, Revision 4, Supplement 4 included a commitment regarding the removal of capsules OC1-D and OC3-F; however, that commitment could not be met because these capsules could not be removed from the CR-3 reactor vessel. By letter dated May 16, 2005, the NRC staff reviewed and approved the revised withdrawal schedule, stating that there was no impact in fulfilling the requirements of 10 CFR 50 Appendix H or ASTM E 185-82, because there were additional capsules within the MIRVP that contained the same limiting material.

The OE review showed that the CR-3 Reactor Vessel Surveillance Program is continually improving, provides for the continued safe operation of the plant by managing the reduction of fracture toughness of the reactor vessel beltline materials due to neutron embrittlement, and fulfills the intent and scope of 10 CFR 50, Appendix H.

Conclusion

The continued implementation of the Reactor Vessel Surveillance Program, with the enhancements identified above, will provide reasonable assurance that neutron embrittlement aging effects will be managed so that the systems 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 ONE-TIME INSPECTION PROGRAM

Program Description

The One-Time Inspection Program uses one-time inspections to verify the effectiveness of an aging management program and confirm the absence of an aging effect. The Program includes verification inspections specified by NUREG-1801 for the Water Chemistry Program, Fuel Oil Chemistry Program, and Lubricating Oil Analysis Program, and plant-specific inspections to confirm the condition of certain civil/structural components. Prior to the period of extended operation, procedural controls for the Program will be implemented to track, initiate, complete, and report activities associated with one-time Inspections.

The One-Time Inspection Program is credited for aging management of various structures/components at CR-3 as shown below:

Structure/Component	Building Structure/ System	Aging Effect of Concern
Heat exchanger components and tubes, tanks, pump casings, closure heads, strainers, deaerators, heaters, orifices, flow elements, piping, venturis, flanges, covers, nozzles, turbine casings, steam generator components, piping components and elements that credit the Water Chemistry Program for aging management.	Auxiliary Steam Liquid Sampling Condensate OTSG Chemical Cleaning Control Complex Chilled Water Condensate Demineralizer Decay Heat Removal Demineralized Water Emergency Feedwater Main Feedwater Gland Steam Gland Seal Water Miscellaneous Drains Main Steam Make Up & Purification Reactor Coolant Station Drains Secondary Plant Cycle Startup Nuclear Services Closed Cycle Cooling Waste Disposal Radioactive Gas Waste Disposal Waste Gas Sampling	Cracking, Flow Blockage, Loss of Material, and Reduction of Heat Transfer

Structure/Component	Building Structure/ System	Aging Effect of Concern
Flow restrictors, Class 1 piping, fittings and branch connections, valve bodies, the RV Flange Leak Detection Line tap weld, and piping that credit the Water Chemistry and ASME Section XI In-service Inspection Programs for aging management.	Reactor Coolant Incore Monitoring	Cracking
Heat exchanger components and tubes, pump casings, heater housings, tanks, pans, strainers, expansion joints, and piping, piping components, and piping elements that credit the Lubricating Oil Analysis Program for aging management.	Control Complex Chilled Water Decay Heat Closed Cycle Cooling Jacket Coolant Diesel Generator Lube Oil Emergency Feedwater Main Feedwater Main Feedwater Turbine Lube Oil Reactor Coolant Pump Lube Oil Collection Make Up & Purification Reactor Coolant Secondary Services Closed Cycle Cooling Water Nuclear Services Closed Cycle Cooling	Cracking, Flow Blockage, Loss of Material, and Reduction of Heat Transfer
Tanks, filter housings, pump casings, strainers, standpipes, hydrants, piping, piping elements, and piping components that credit the Fuel Oil Chemistry Program for aging management.	Fuel Oil Fire Protection	Flow Blockage, Loss of Material, and Cracking
Emergency Diesel Fuel Oil Storage Tank hold-down Straps	Miscellaneous Structures	Loss of Material

NUREG-1801 Consistency

The One-Time Inspection Program is a new Program that is consistent with NUREG-1801, Section XI.M32.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The One-Time Inspection Program is a new program. The CR-3 aging management review process ensures that one-time inspections will be prescribed and developed with consideration of plant and industry OE, that results of the inspections performed under the Program are disseminated and evaluated, and that industry OE is reviewed for applicability.

NUREG-1801 is based on industry OE through January 2005. This Program applies to potential aging effects for which there are currently no OE indicating the need for an aging management program. Nevertheless, the elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice. More recent OE is captured through the normal OE review process where it is screened for applicability. This process will continue through the period of extended operation.

Conclusion

Implementation of the One-Time Inspection Program provides reasonable assurance that aging effects will be managed so that the systems and components within the scope of this Program will continue to perform their intended functions consistent through the period of extended operation.

B.2.19 SELECTIVE LEACHING OF MATERIALS PROGRAM

Program Description

The Selective Leaching of Materials Program ensures the integrity of components and/or commodities (such as piping, pump casings, valve bodies and heat exchanger components) made of uninhibited copper alloys with zinc content greater than 15% or aluminum content greater than 8%, and gray cast iron exposed to a raw water, treated water, closed cycle cooling water, open cycle cooling water, fire water, steam, fuel oil, uncontrolled indoor air, or soil environment at CR-3. A new inspection procedure will define a one-time examination methodology and acceptance criteria. The Program will be implemented by the Work Management Process using a qualitative determination of selected components that may be susceptible to selective leaching. Confirmation of selective leaching may be performed with a metallurgical evaluation or other testing methods.

The examinations will determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function(s) for the period of extended operation. A sample population will be selected for the inspections which will be completed prior to commencing the period of extended operation. Evidence suggesting the presence of selective leaching will result in expanded sampling, as appropriate, and engineering evaluation.

NUREG-1801 Consistency

The Selective Leaching of Materials Program is a new program consistent with NUREG-1801, Section XI.M33, with an exception.

Exceptions to NUREG-1801

<u>Program Elements Affected</u>

- **Scope of Program**
  The exception involves the use of examinations, other than Brinell hardness testing identified in NUREG-1801, to identify the presence of selective leaching. A qualitative determination of selective leaching will be used in lieu of Brinell hardness testing for components within the scope of this Program. The exception is justified, because (1) Brinell hardness testing may not be feasible for most components due to form and configuration (e.g., heat exchanger tubes) and (2) other mechanical means, i.e., scraping, or chipping, provide an equally valid method of identification.

- **Parameters Monitored/Inspected**
  A qualitative determination of selective leaching will be used in lieu of Brinell

hardness testing for components within the scope of this Program. Refer to the discussion of this exception under the Scope of Program element above.

- **Detection of Aging Effects**
  A qualitative determination of selective leaching will be used in lieu of Brinell hardness testing for components within the scope of this Program. Refer to the discussion of this exception under the Scope of Program element above.

Enhancements

None.

Operating Experience

The Selective Leaching of Materials Program is a new program; therefore, OE to verify the effectiveness of the Program is not available. Past inspections performed at CR-3 have revealed instances of selective leaching of materials. The actions specified by the Corrective Action Program will ensure that appropriate measures will be taken to preclude or monitor for recurrence in systems or material/environment combinations in which selective leaching is detected. The Operating Experience Program ensures that other systems with similar material/environment combinations will also be inspected for selective leaching. Examples of plant-specific OE include the following items:

- Maintenance activities performed on cast iron Nuclear Service and Decay Heat Sea Water Pump 3B indicated that the discharge pump flange was degraded due to selective leaching. Corrosion products were removed mechanically and the area coated to inhibit further dealloying. No other actions were determined to be warranted since selective leaching is a slow process that is being inhibited by the coating. During a subsequent outage the degraded components were replaced.

- Investigation of a loose part in the Decay Heat Closed-Cycle Heat Exchanger B identified a bronze loose part. The loose part was found to originate from a hinge arm from valve RWV-34. The valve was found to be operational. Laboratory analysis showed that selective leaching was a principal contributor to the degradation of the valve component. Corrective actions included changing the hinge material to an aluminum bronze that is resistant to selective leaching and periodic inspections.

Conclusion

Implementation of the Selective Leaching of Materials Program will provide reasonable assurance that the aging effects will be managed such that the components within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.20 BURIED PIPING AND TANKS INSPECTION PROGRAM

Program Description

The Buried Piping and Tanks Inspection Program manages the aging effect of loss of material for the external surfaces of buried steel components in CR-3 systems within the scope of License Renewal. Components within the scope of the Program consist of steel piping components and two buried tanks. Not included are the underground concrete pipes connecting the Auxiliary Building to the Nuclear Service and Decay Heat Sea Water System Discharge Structure which are managed by the Structures Monitoring Program. The aging effects/mechanisms of concern are loss of material due to general, galvanic, pitting, and crevice corrosion and MIC. To manage the aging effects, this new program includes: (a) preventive measures to mitigate degradation (e.g. coatings and wrappings required by design), and (b) visual inspections of external surfaces of buried piping and tanks, when excavated, for evidence of coating damage and degradation.

Detailed procedural requirements for the Program will be developed and incorporated into implementing procedures. These procedures will provide the administrative controls for the Program and will: (1) ensure an appropriate as-found pipe coating and material condition inspection is performed whenever buried piping within the scope of this Program is exposed, with a minimum frequency of at least one buried piping inspection each 10 years, (2) verify that there is at least one opportunistic or focused inspection performed within the ten year period prior to the period of extended operation, (3) specify that an inspection datasheet is used, (4) require inspection results to be documented, (5) include precautions concerning excavation and use of backfill for License Renewal piping, (6) include a requirement that buried piping coating inspection shall be performed, when excavated, by qualified personnel to assess its condition, and (7) include a requirement that a coating engineer or other qualified individual (such as the Coatings Program Manager) should assist in evaluation of any buried piping coating damage and/or degradation found during the inspection. Any evidence of damage to the coating or wrapping, such as perforations, holidays or other damage will cause the protected components to be inspected for evidence of loss of material. The Program assures that the effects of aging on buried piping components are being effectively managed for the period of extended operation.

NUREG-1801 Consistency

The Buried Piping and Tanks Inspection Program is a new program that is consistent with NUREG-1801, Section XI.M34.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Buried Piping and Tanks Inspection Program is a new program applicable to buried piping. There is no existing OE to validate the effectiveness of this Program. NUREG-1801 is based on industry OE through January 2005. Recent industry OE has been reviewed for applicability. More recent OE is captured through the normal OE review process where it is screened for applicability. This process will continue through the period of extended operation.

At CR-3, buried piping leaks have occurred in the Fire Protection System. These were evaluated and determined not to be caused by age-related degradation. Based on this site experience, it can be concluded that leaks in CR-3 buried piping have been detected and that appropriate corrective actions have been taken to ensure no loss of component intended function. This experience is not atypical and justifies the use of the 10-year inspection frequency for buried components endorsed by NUREG-1801.

Conclusion

Implementation of the Buried Piping and Tanks Inspection Program provides reasonable assurance that the aging effect of loss of material due to corrosion mechanisms will be managed such that systems and components within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.21 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL-BORE PIPING PROGRAM

Program Description

The industry has experienced cracking of small-bore piping as the result of thermal and mechanical loading and intergranular stress corrosion. Specific industry-identified events include cracking caused by fatigue due to thermal stratification which resulted in the issuance of NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant System," (as supplemented). The ASME Code does not currently require volumetric examination of Class 1 small-bore piping. However, as stated in NUREG-1801, Section XI.M35, the NRC believes that the inspection of small-bore Class 1 piping (less than NPS 4) should include volumetric examinations to identify cracking. The CR-3 One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will manage this aging effect through the use of volumetric examinations. The current state of technology provides no effective, reliable method of performing volumetric examinations of small-bore socket welds. In lieu of performing volumetric inspections of socket welds, the Program will include one-time volumetric examinations of a sample of Class 1 butt welds for pipe less than NPS 4. The volumetric inspections will be completed prior to the end of, and within the last five years of, the current operating period. In addition, the Program will include controls to ensure that ASME Class 1 socket welds are inspected in accordance with the approved ASME Section XI ISI program. Any cracking identified in small-bore Class 1 piping determined to be attributable to stress corrosion or thermal and mechanical loading will result in periodic inspections to be managed by a plant-specific program.

The CR-3 One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will manage cracking in small-bore piping (less than NPS 4) such that the system intended function is maintained and loss of RCS pressure boundary does not occur through the period of extended operation. This Program will be implemented and inspections completed and evaluated prior to the period of extended operation.

NUREG-1801 Consistency

The CR-3 One-Time Inspection of ASME Code Class 1 Small-Bore Piping is a new program that is consistent with NUREG-1801, Section XI.M35.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

In 1982, CR-3 experienced a failure of a weld associated with the normal duty makeup line. A report was written as a means of documenting the activities of Florida Power Corporation and its contractors during the investigation and repair of the makeup system nozzle at Crystal River Unit 3. The report concluded in part that:

1. The leak path in the valve to safe end pipe weld was formed by the joining of a circumferential crack from the inside diameter (ID) with a circumferential crack from the outside diameter (OD).
2. The ID crack initiated at a machine tool mark in the valve body probably by thermal fatigue. Propagation probably occurred by combined mechanical and thermal loading.
3. The OD crack initiated at the discontinuity formed by the weld between the valve and safe end on the valve side. Crack initiation and propagation probably occurred by mechanical loading of the system.

The report posited two explanations that could account for hot RCS fluid backing up into the double duty line and resulting in a thermal cycle. One theory was that a loss of roll in the thermal sleeve could open up the annulus area in between the thermal sleeve and the nozzle. This would lead to a chimney affect drawing the hot RCS fluid through the annulus and lead to turbulent thermal mixing in the safe-end/thermal sleeve region. The second theory dealt with low flow velocity ratios between the RCS and normal and/or minimum makeup flow. There had been studies performed in the area of thermal shock mixing which indicates unusual and unexpected flow patterns as a function of flow velocity ratios. Further data at other B&W facilities on the HPI nozzle problem seemed to indicate that the loss of roll is a common denominator in this problem. That would tend to support the loss of roll theory.

During Refuel 13, in 2003, inspections were performed in accordance with B&W Topical Report, "HPI/MU Nozzle Component Cracking," on three High-Pressure Injection (HPI) nozzles and associated piping up to the first isolation valves using ultrasonic techniques. Two HPI Nozzles Thermal Sleeves were examined by internal remote visual techniques. One HPI thermal sleeve was found to be cracked, and it was replaced.

Components with identified cracking are managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program.

NUREG-1801 is based on industry OE through January 2005. Recent industry OE has been reviewed for applicability. More recent OE is captured through the normal OE review process where it is screened for applicability. This process will continue through the period of extended operation.

Conclusion

Based on this evaluation, there is reasonable assurance that, when implemented, the CR-3 One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will adequately manage cracking in small-bore Class 1 piping so that system intended functions will be maintained consistent with the CLB for the period of extended operation.

B.2.22 EXTERNAL SURFACES MONITORING PROGRAM

Program Description

The External Surfaces Monitoring Program is based on system inspections and walkdowns. This Program consists of periodic visual inspections of components such as piping, piping components, ducting, and other equipment within the scope of License Renewal and subject to aging management review in order to manage aging effects. The Program manages aging effects through visual inspection of external surfaces. Loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion Program. Surfaces that are inaccessible during plant operations are inspected during refueling outages. The Program includes measures to provide assurance that aging effects are managed on surfaces that are inaccessible during both plant operations and refueling outages.

NUREG-1801 Consistency

The External Surfaces Monitoring Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.M36.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

<u>Program Elements Affected</u>

- **Scope of Program**
  1) Implementing procedures will be enhanced to ensure that the Program encompasses all of the systems and components that credit the Program for aging management.

  2) Program procedures will be enhanced to include inspection attributes adequate for identifying aging effects for the ranges of materials and aging effects within the scope of the Program.

  3) Implementing procedures will be enhanced to include measures to assure that aging effects are managed on surfaces that are inaccessible or not readily visible during both plant operations and refueling outages, such that reasonable

assurance is provided that applicable components will perform their intended function during the period of extended operation.

- **Parameters Monitored/Inspected**
  Program procedures will be enhanced to detect aging effects/mechanisms and qualify degradations consistent with the demand of components crediting the External Surfaces Monitoring Program for aging management. Identified aging effects include loss of material, hardening and loss of strength of elastomers, and reduction of heat transfer caused by fouling.

- **Detection of Aging Effects**
  1) Program procedures will be revised to include inspection attributes regarding the degradation of coatings.

  2) Enhancement 3) under Scope of Program above regarding inspection of inaccessible surfaces of components is applicable to this element also.

Operating Experience

System monitoring activities at CR-3 have proven to be effective in maintaining the material condition of plant systems. System folders are maintained documenting information regarding system health, including performance monitoring and results of system walkdowns. Action Requests are initiated as needed to identify and resolve deficiencies, including material condition. A review of system health reports from the most recent reporting period (January to June 2008) confirms that system walkdowns are being performed in a timely manner, that results are being trended, that longstanding items are identified and resolved, that plans are developed and implemented to optimize system health, and that systems are being effectively monitored.

The effectiveness of systems monitoring, including system walkdowns, has been the subject of both focused and site-wide self assessments. These assessments have evaluated timeliness and frequency requirements, documentation requirements, system engineer training, and the overall effectiveness of system walkdowns. Improvements have been made as a result of these assessments, including use of PassPort for implementing walkdown schedules, and formalization of frequency and documentation requirements. These assessments have concluded that CR-3 is effectively executing system walkdowns.

Conclusion

Implementation of the External Surfaces Monitoring Program, with the enhancements identified above, will provide reasonable assurance that the aging effects will be adequately managed such that the components within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.23 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS PROGRAM

Program Description

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program that will be implemented via existing preventive maintenance, surveillance testing, and periodic testing work order tasks that provide opportunities for the visual Inspection of internal surfaces of piping and ducting components. Periodic internal inspections of components allow timely detection of degradation and determination of appropriate corrective actions. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program work activities will monitor parameters that may be detected by visual inspection and include change in material properties, cracking, flow blockage, hardening, loss of material, and reduction of heat transfer effectiveness. In addition to visual inspection of internal surfaces, the program includes a limited scope of preventive maintenance activities that involve 1) physical manipulation or other investigative methods to detect aging effects, and 2) inspection of outside surfaces. The extent and schedule of inspections and testing assure detection of component degradation prior to loss of intended functions.

NUREG-1801 Consistency

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program consistent with NUREG-1801, Section XI.M38.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a new program; and, as such, no OE exists for a demonstration of program effectiveness. The Program will be implemented via existing preventive maintenance, surveillance testing, and periodic testing work order tasks. Such tasks have been in place at CR-3 since the plant began operation. These activities have proven effective at maintaining the material condition of systems, structures, and components and detecting unsatisfactory conditions. System Engineers review OE for possible impact to the equipment in their systems. The basis for parameters monitored and inspection intervals will be based on vendor recommendations, historical performance, and industry wide OE. Operating experience is disseminated and evaluated as described in the Operating Experience Program. This ongoing review of OE will continue through the period of extended operation.

Conclusion

Implementation of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program provides reasonable assurance that applicable aging effects will be managed such that the components within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.24 LUBRICATING OIL ANALYSIS PROGRAM

Program Description

The purpose of the Lubricating Oil Analysis Program is to ensure the oil environment in mechanical systems is maintained to the required quality. The Lubricating Oil Analysis Program maintains oil system contaminants (primarily water and particulates) within acceptable limits, thereby preserving an environment that is not conducive to loss of material, cracking, flow blockage, or reduction of heat transfer. Lubricating oil testing activities include sampling and analysis of lubricating oil for detrimental contaminants. The program also implements periodic oil changes at fixed intervals for selected components; a particle count and check for water are performed on the old oil prior to disposal to detect evidence of abnormal wear rates, contamination by moisture, or excessive corrosion.

NUREG-1801 Consistency

The Lubricating Oil Analysis Program is an existing program that is consistent with NUREG-1801, Section XI.M39, with exception.

Exceptions to NUREG-1801

Program Elements Affected

- **Parameters Monitored/Inspected**
  The Lubricating Oil Analysis Program does not measure flash point on a periodic basis. Flash point is only measured upon receipt inspection or on systems where a combustible gas may accumulate. CR-3 has no lube oil reservoirs where a combustible gas may accumulate above the lube oil. Therefore, this test is not warranted.

Enhancements

None.

Operating Experience

The Lubricating Oil Analysis Program utilizes OE to promote the identification and transfer of lessons learned from both internal and industry events so that the knowledge gained can be used to improve nuclear plant safety and operations. The Program provides the methodology for receiving, processing, status reporting, screening, reviewing, evaluating, and taking preventative and corrective actions in response to applicable OE information. Examples of plant-specific OE are provided below to demonstrate program effectiveness:

- A CR-3 NCR reported that a routine lube oil sample for the outboard motor bearing of a Circulating Water Pump discovered visible ferrous wear debris. This was an indication the motor bearing had degraded to a point where replacement was likely necessary. Because of this finding, pump operation was restricted until the motor could be refurbished or replaced.

- Another NCR indicated that a routine lube oil sample collected following the replacement of a Decay Heat Pump rotating assembly was discolored. It was suspected that this discoloration was due to break-in wear of the pump bearings. A sample was shipped to oil analysis vendor for further analysis. A Work Order was initiated to drain, flush, and refill the pump bearing reservoir. The affected pump will be carried on the Predictive Maintenance Observation and Action list as an increased frequency monitoring item until wear particle analysis results have returned to normal.

The review of OE shows that the Lubricating Oil Analysis Program is continually upgraded based on industry experience, external and internal assessments, and routine program performance, and has proven effective in maintaining lube oil quality for site equipment. The overall effectiveness of the Lubricating Oil Analysis Program is supported by the OE for systems and components in that no instances of failures attributed to lubricating oil contamination have been identified.

Conclusion

Implementation of the Lubricating Oil Analysis Program will provide reasonable assurance that applicable aging effects will be managed such that the in-scope components subject to a lubricating oil environment will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.25 ASME SECTION XI, SUBSECTION IWE PROGRAM

Program Description

The ASME Section XI, Subsection IWE Program consists of periodic inspections of Class MC components of the containment structure. The Program is in accordance with the ASME Code, Section XI, Subsection IWE, 2001 Edition through the 2003 Addenda as modified by 10CFR50.55a. The ASME Section XI, Subsection IWE Program is credited for the aging management of the:

1. Metallic liner and integral attachments for the concrete containment,
2. Penetration sleeves,
3. Personnel airlock and equipment hatch,
4. Pressure retaining bolting, and
5. Moisture barriers.

The primary inspection method for the ASME Section XI, Subsection IWE Program is periodic visual examination along with limited volumetric examinations utilizing ultrasonic thickness measurements as needed.

NUREG-1801 Consistency

The ASME Section XI Subsection IWE Program is an existing program consistent with NUREG-1801, Section XI.S1.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The ASME Section XI, Subsection IWE Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the Program is effectively implemented to meet regulatory, process, and procedure requirements. Periodic program reviews are performed. The Program is upgraded based on industry and plant-specific experience. Additionally, plant OE is shared among Program personnel at all four of Progress Energy nuclear plant sites.

Plant-specific operating history includes several general visual examinations that were performed on the Reactor Building (RB) liner plate, penetrations, bolting and associated attachments. These examinations have identified instances of age-related degradation of the liner plate caused by general and pitting corrosion, general corrosion of penetrations, deterioration of the moisture barrier at the liner/floor interface, deteriorated cork material under the moisture barrier, and liner plate coating degradation. Corrective actions were taken to assure the intended function of the liner and to repair or replace the degraded components. The moisture barrier was completely removed and the deteriorated cork material below the moisture barrier was replaced. The liner plate was recoated, the cork was installed, and moisture barrier was replaced. The structural integrity of the RB liner plate was not degraded beyond its design margin. The corrosion on the penetrations was evaluated as minor surface corrosion that did not impact the structural integrity of the penetrations. An NCR report was initiated to monitor corrosion of the liner during future outages to determine if further compensatory actions need to be taken. In addition, a detailed visual examination of the condition of the moisture barrier at the liner/floor interface has been planned for a future outage.

Industry and site OE demonstrates the Program is effective at detecting and managing aging affects so that the intended functions of the applicable components will be maintained during the period of extended operation.

Conclusion

Continued implementation of the ASME Section XI, Subsection IWE Program will provide reasonable assurance that the aging effects of pressure retaining Containment Structure Class MC components are adequately managed so that the intended functions of the applicable components will be maintained during the period of extended operation.

B.2.26 ASME SECTION XI, SUBSECTION IWL PROGRAM

Program Description

The ASME Section XI, Subsection IWL Program is implemented in accordance with 10 CFR 50.55(a) and ASME Section XI, Subsection IWL, 2001 Edition, through the 2003 Addenda. The Program manages the reinforced concrete and unbonded post-tensioning system of the CR-3 Class CC containment structure. The Program requires periodic inspection of the reinforced concrete Reactor Building (RB) and inspection and testing of a sample of the unbonded post-tensioning system as specified by ASME Section XI, Subsection IWL. The Program includes ASME Section XI, Subsection IWL, examination categories L-A, for concrete surfaces, and L-B, for the unbonded post-tensioning system. The second ASME Section XI, Subsection IWL Program interval began on August 14, 2008.

NUREG-1801 Consistency

The ASME Section XI, Subsection IWL Program is an existing program consistent with NUREG-1801, Section XI.S2.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The ASME Section XI, Subsection IWL Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the Program is effectively implemented to meet regulatory, process, and procedure requirements. Periodic program reviews are performed. The Program is upgraded based on industry and plant-specific experience. Additionally, plant OE is shared among Program personnel at all four of Progress Energy nuclear plant sites.

Plant-specific OE includes the results of periodic examinations of the RB reinforced pressure boundary concrete and tendon surveillances. The 30th year examination of the RB reinforced pressure boundary concrete and the 8th Tendon Surveillance were completed in the Fall of 2007 during Refueling Outage 15.

Past examinations performed on concrete surfaces have identified several indications including staining from grease and rust, discoloration, voids, honeycomb, popouts, minor cracks of less than 0.04 in. across, spalling, efflorescence, deflection, items embedded in concrete protruding from concrete, and displacement and deterioration of grout. These items were documented and dispositioned as minor in nature and not adversely affecting the overall structural integrity of the RB. The evaluation attributed the suspect concrete areas to normal aging of the structure following exposure to the environment for approximately 30 years.

Previous Tendon Surveillance results have identified several instances where tendons have been found with lift-off forces below the Predicted Base Value. Historically, CR-3 has found numerous tendons below 95% of predicted base value, but has demonstrated the acceptability of the RB with the as-found conditions. Small grease and oil leaks were also identified on multiple tendon caps, located inside existing structures that adjoin the RB. This is not considered to be system degradation. Inspection results also identified several instances of missing or broken wires. These instances were compared against the acceptance criteria and found to be acceptable. One tendon exceeded 10% of net volume absolute difference between the amount of grease removed and the amount replaced. The condition was reviewed and found acceptable. The evaluations of OE to date have concluded that the CR-3 containment structure is functioning as designed and that the RB structure meets code requirements and has experienced no abnormal degradation of the post-tensioning system.

NRC Information Notice (IN) 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," was reviewed for applicability to CR-3. It was determined that the procedure used to control the tendon surveillance addressed the issues contained in the IN. The data for the CR-3 tendon history was informally reviewed using regression analysis, and the results did not vary appreciably from trending the group averages. The implementing procedure requires CR-3 to trend the group averages. Based on this review it was determined that there were no actions required as a result of IN 99-10.

Industry and site OE demonstrates the ASME Section XI, Subsection IWL Program is effective at detecting and managing aging affects so that the intended functions of the applicable components will be maintained during the period of extended operation.

Conclusion

Continued implementation of the ASME Section XI, Subsection IWL Program will provide reasonable assurance that the aging effects of the RB pressure-retaining reinforced concrete and unbonded post-tensioning system are adequately managed so that the intended functions of the applicable components will be maintained during the period of extended operation.

B.2.27 ASME SECTION XI, SUBSECTION IWF PROGRAM

Program Description

The CR-3 ASME Section XI, Subsection IWF Program provides for visual examination of component and piping supports within the scope of License Renewal for loss of material, change in material properties, and loss of mechanical function. The Program is implemented through plant procedures, which provide for visual examination of ISI Class 1, 2, and 3 supports. Visual examination is provided in accordance with the requirements of ASME Section XI, Subsection IWF, 2001 Edition, through the 2003 Addenda, as modified by 10 CFR 50.55a. The ASME Section XI, Subsection IWF program is credited for the aging management of the supports for ASME Class 1, 2, 3 piping and components and supports for Reactor Coolant System primary equipment.

NUREG-1801 Consistency

The CR-3 ASME Section XI, Subsection IWF Program is an existing program consistent with NUREG-1801, Section XI.S3.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The ASME Section XI, Subsection IWF Program is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the Program is effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews; qualified personnel are assigned as program managers and are given authority and responsibility to implement the Program; and adequate resources are committed to Program activities.

Plant-specific OE has identified numerous assessments, performed on both a plant-specific and corporate basis, dealing with program development, effectiveness, and implementation. The CR-3 IWF Program is continually being upgraded based upon industry and plant specific experience. Additionally, plant OE is shared between Progress Energy sites through regular peer group meetings, a common corporate sponsor, and outage participation of program managers from other Progress Energy sites.

An example of use of the Corrective Action Program occurred following the identification of a corroded support located in the Sea Water Room of the Auxiliary Building during a scheduled inservice inspection. The remaining supports in the same trench were inspected and also found to be corroded. Thus, the Corrective Action Program was used to identify and replace the affected supports with an improved support design. Another example of using the Corrective Action Program in this manner was the identification and replacement of degraded supports in the RB.

Conclusion

Continued implementation of the CR-3 ASME Section XI, Subsection IWF Program will provide reasonable assurance that the aging effects are managed such that the components/commodities within the scope of this Program will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.28 10 CFR PART 50, APPENDIX J PROGRAM

Program Description

The 10 CFR 50, Appendix J Program is an existing performance-based testing program. The program monitors leakage rates through the containment pressure boundary, including penetrations and access openings. Containment leak rate tests assure that leakage through the primary containment, and systems and components penetrating primary containment, do not exceed the allowable leakage limits specified within the CR-3 Technical Specifications. Corrective actions are taken if leakage rates exceed established administrative limits for individual penetrations or the overall containment pressure boundary. Seals and gaskets are also monitored under the program. The CR-3 10 CFR 50, Appendix J Program utilizes the performance-based approach of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, and includes appropriate guidance from Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak-Test Program," as modified by NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J."

Type A tests are conducted to measure the containment overall integrated leakage rate. Plant procedures require a general visual inspection of the accessible interior and exterior surfaces of the primary containment and components prior to each integrated leak rate test (ILRT). Type B and Type C local leak rate tests (LLRTs) are performed on containment pressure boundary access penetrations and containment isolation valves at frequencies that comply with the requirements of 10 CFR 50 Appendix J, Option B.

NUREG-1801 Consistency

The 10 CFR Part 50, Appendix J Program is an existing program consistent with NUREG-1801, Section XI.S4.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The 10 CFR Part 50, Appendix J Program is maintained in accordance with CR-3 engineering program requirements. This provides assurance that the Program is effectively implemented to meet regulatory, process, and procedure requirements, including periodic reviews; that qualified personnel are assigned as program managers and are given authority and responsibility to implement the Program; and that adequate resources are committed to Program activities.

The Containment ILRT was last performed in December, 2005 during Refueling Outage 14. The ILRT test results were satisfactory with no corrective or follow-up actions initiated. In addition, site OE confirms that the LLRTs are effective in identifying and initiating corrective actions for leakage at containment penetrations, including the equipment hatch and air locks, and in confirming the effectiveness of the corrective actions taken.

Examples of plant-specific OE are the identification of leakage and implementation of corrective actions for containment isolation valves that failed LLRTs. The following examples are typical of the effective methods used at CR-3 for identifying and correcting valve leakage problems.

- In one case, a Containment Isolation Valve failed the LLRT and was disassembled and inspected. The valve disc was replaced and the as-left LLRT was satisfactory.

- In a second case, failure of a LLRT in 2003 resulted in the valve stroke being adjusted because the internal inspection of the valve revealed no problems. The same valve failed a second test in 2007. Long-term corrective actions were planned to repair the valve during the next refueling outage since overall leakage was acceptable. During the next refueling the valve again failed leak rate testing and was subsequently disassembled and rebuilt. However, plans for replacing the valve were implemented when the valve again failed the LLRT.

Based on review of operating history, corrective actions, and self-assessments, the 10 CFR Part 50, Appendix J Program is continually monitored and enhanced to incorporate the results of OE; as such it provides an effective means of ensuring the structural integrity and leak tightness of the CR-3 containment.

Conclusion

Continued implementation of the 10 CFR Part 50, Appendix J Program will provide reasonable assurance that applicable aging effects are managed such that the components/commodities within the scope of this Program will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.29 MASONRY WALL PROGRAM

Program Description

The objective of the Masonry Wall Program is to manage aging effects so that the evaluation basis established for each masonry wall within the scope of License Renewal remains valid through the period of extended operation. The Program includes masonry walls identified as performing intended functions in accordance with 10 CFR 54.4. Included walls are the masonry walls within the Auxiliary Building, Control Complex, Turbine Building, Fire Service Pumphouse and the Switchyard Relay Building. The Program is a condition monitoring program with the inspection frequencies established such that no loss of intended function would occur between inspections.

NUREG-1801 Consistency

The Masonry Wall Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.S5.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

<u>Program Elements Affected</u>

- **Scope of Program**
  Revise Program administrative controls to identify the structures that have masonry walls in the scope of License Renewal.

Operating Experience

The CR-3 Masonry Wall Program was implemented on the schedule mandated by 10 CFR 50.65, the Maintenance Rule. A baseline inspection of masonry walls in the scope of Maintenance Rule was completed in 1997. A subsequent inspection of structures was completed in 2007 consistent with the program frequency of at least one inspection every ten years.

The 2007 Maintenance Rule inspection of Masonry Walls identified no degradation that impacted the intended functions of the walls. The baseline inspection performed in 1997 identified no unacceptable conditions for masonry walls.

The Masonry Wall Program is conducted through a corporate procedure that is implemented and maintained in accordance with the general requirements for engineering programs. This provides assurance that the Masonry Wall Program is effectively implemented to meet regulatory, process, and procedural requirements, including periodic reviews; that qualified personnel are assigned as program managers and are given the authority and responsibility to implement the Masonry Wall Program; and that adequate resources are committed to Program activities.

Inspections and assessments have been conducted and show the Masonry Wall Program through the Maintenance Rule Program to be critically monitored, and continually improving. The OE review has concluded that administrative controls are in effect and effective in identifying age related degradation and initiating corrective action.

Conclusion

Following enhancement, implementation of the Masonry Wall Program will ensure the effects of aging associated with masonry walls in the scope of License Renewal will be adequately managed so that there is reasonable assurance that their intended functions will be performed consistent with the CLB during the period of extended operation.

B.2.30 STRUCTURES MONITORING PROGRAM

Program Description

The Structures Monitoring Program manages the aging effects of civil/structural commodities within the scope of License Renewal. The Structures Monitoring Program is implemented, through procedures, in accordance with the regulatory requirements and guidance associated with the Maintenance Rule, 10 CFR 50.65; NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Rev. 2, and Nuclear Energy Institute (NEI) 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Rev. 2. The Program incorporates criteria recommended by the Institute for Nuclear Power Operations (INPO) Good Practice document 85-033, "Use of System Engineers;" NEI 96-03, "Guidelines for Monitoring the Condition of Structures at Nuclear Plants," and inspection guidance based on industry experience and recommendations from American Concrete Institute Standard ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures;" and American Society of Civil Engineers, ASCE 11-90, "Guideline for Structural Condition Assessment of Existing Buildings." The Program consists of periodic inspection and monitoring of the condition of structures and structure component supports to ensure that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined.

NUREG-1801 Consistency

The Structures Monitoring Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.S6.

Exceptions to NUREG-1801

None.

Enhancements

Prior to the period of extended operation, the below-listed enhancements will be implemented:

<u>Program Elements Affected</u>

- **Scope of Program**
  Administrative controls that implement the Program will be revised to:
  1) Specifically identify the License Renewal structures and systems that credit the Program for aging management in the corporate procedure for condition monitoring of structures,

2) Require notification of the responsible engineer when below-grade concrete, including concrete pipe, is exposed so an inspection may be performed prior to backfilling,

3) Require periodic inspections of water control structures, i.e., Circulating Water Intake Structure, Circulating Water Discharge Structure, Nuclear Service Sea Water Discharge Structure, Intake Canal, and Raw Water Pits, on a frequency not to exceed five years,

4) Require periodic inspections of the Circulating Water Intake Structure submerged portions on a frequency not to exceed five years,

5) Require periodic groundwater chemistry monitoring including consideration for potential seasonal variations,

6) Require inspection of inaccessible surfaces of reinforced concrete pipe when exposed due to removal of backfill for any reason in the corporate procedure for condition monitoring of structures, and

7) Include additional in-scope structures and specific civil/structural commodities in periodic maintenance activities.

- **Parameters Monitored/Inspected**
  Administrative controls that implement the Program will be revised to:
  1) Identify additional civil/structural commodities along with the associated inspection attributes and performance standard required for License Renewal in the corporate procedure for condition monitoring of structures,

  2) Require notification of the Responsible Engineer when below-grade concrete including concrete pipe is exposed so an inspection may be performed prior to backfilling,

  3) Require inspection of inaccessible surfaces of reinforced concrete pipe when exposed due to removal of backfill for any reason in the corporate procedure for condition monitoring of structures,

  4) Identify additional inspection criteria for structural commodities in the site system walkdown checklist,

  5) Add inspection for corrosion to the inspection criteria for the bar racks at the Circulating Water Intake Structure as a periodic maintenance activity,

  6) Add an inspection of the earth for loss of form and loss of material for the Wave Embankment Protection Structure as a periodic maintenance activity,

7) Require inspection of the Fluorogold slide bearing plates used in structural steel platform application located in the Reactor Building on an established frequency.

Operating Experience

The CR-3 Structures Monitoring Program was implemented on the schedule mandated by 10 CFR 50.65(a) for the Maintenance Rule (MR). A baseline inspection of structures in the scope of MR was completed in 1997. A subsequent inspection of structures was completed in 2007 consistent with the program frequency of not exceeding ten years. Periodic walkdowns of MR systems, including inspection of structural features, have also been performed. Intake Canal surveys for proper dimensions have been performed on a minimum frequency of every four years. The 2007 MR inspection of structures identified no significant degradation that impacted the intended functions of the structures and structural components. Corrosion of steel components was identified in several structures, e.g., equipment supports in the Auxiliary Building Seawater Room, support steel at the Circulating Water Intake Structure, the Borated Water Storage Tank enclosure, and a cable vault. Corrosion of steel members of the East Cable Bridge was identified, and a re-inspection was scheduled to be completed in 2008. Evidence of water intrusion was observed in the Tendon Access Gallery, the Decay Heat Vaults, and a cable vault. A hairline crack on the concrete wall of the Spent Fuel Pool was re-inspected and has not increased in size since the last inspection. The interval for this inspection has been established as yearly.

System Engineering walkdowns and use of the Corrective Action Program (CAP) by plant personnel have identified deficiencies on structural features. A review of the CAP back to 1999 identified several instances of corrosion on steel commodities such as supports, anchors, bolts, door hinges, and siding. The primary deficiencies for concrete were cracking, spalling, and leaching. Loss of material due to washout on the Berm was identified. Deterioration of elastomeric materials in the flood barrier, access plugs, door seals, and roof seals was identified. Water intrusion was identified in several below grade areas and was corrected through the CAP.

During 2005, a significant adverse condition associated with water intrusion into the Emergency Diesel Generator 1A Fuel Oil Storage Tank was identified, and an investigation was performed. As a result, the flood protection system surveillance was enhanced to include an inspection for structural condition of the concrete flood retaining wall and watertight seals. The Structures Monitoring Program credits this flood protection system surveillance.

As a result of the Intake Canal surveys, the CR-3 Intake Canal has been dredged twice since operation began in 1976: in 1989 and in 2004.

A review of inspection reports, self-assessments, and adverse condition reports has concluded the administrative controls are in effect, effective in identifying age related degradation, implementing appropriate corrective actions, and continually upgrading the administrative controls used for structures monitoring.

Conclusion

Following enhancement, implementation of the Structures Monitoring Program will ensure the effects of aging, associated with License Renewal civil/structural commodities, will be adequately managed so that there is reasonable assurance that their intended functions will be performed consistent with the CLB during the period of extended operation.

B.2.31 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 is credited for the aging management of cables and connections not included in the CR-3 EQ Program. Accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years for cable and connection jacket surface anomalies, such as embrittlement, discoloration, cracking, swelling, or surface contamination, which are precursor indications of conductor insulation aging degradation from heat, radiation or moisture. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the electrical cable or connection. The first inspections for license renewal will be completed before the period of extended operation. The aging effects/mechanisms of concern are as follows:

- Reduced Insulation Resistance
- Electrical Failure

The technical basis for selecting the sample of cables and connections to be inspected is defined in the implementing CR-3 program document. Sample locations will consider the location of cables and connections inside and outside Containment, as well as, any known adverse localized environments.

NUREG-1801 Consistency

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program consistent with NUREG-1801, Section XI.E1.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program with no site specific OE history. However, plant-specific and industry-wide OE was considered in the development of all electrical programs in Appendix B of the CR-3 License Renewal Application. The review of plant-specific and industry-wide OE ensures that the corresponding NUREG-1801, Chapter XI, Program will be an effective aging management program for the period of extended operation. Plant-specific OE for cables and connections has been captured by a review of one or more of the following: (1) the Action Tracking database, (2) System Engineering Notebooks and System Health Reports, and (3) discussions with site engineering personnel. This effort also included a review of applicable site correspondence (Licensee Event Reports, etc).

Industry OE that forms the basis for this program is included in the OE element of the corresponding NUREG-1801, Chapter XI, Program. Industry OE noted in NUREG-1801, has shown that adverse localized environments caused by heat, radiation or moisture for electrical cables and connections have been shown to exist and have been found to produce degradation of insulating materials that is visually observable. These visual indications, such as color changes or surface cracking, can be used as indicators of degradation.

This review of plant-specific and industry-wide OE confirms that the OE discussed in the corresponding NUREG-1801, Chapter XI, Program is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the CR-3 Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the CR-3 QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that all electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of 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 components within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.2.32 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS PROGRAM

Program Description

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is credited for the aging management of radiation monitoring and nuclear instrumentation cables not included in the CR-3 EQ Program. Exposure of electrical cables to adverse localized environments caused by heat, radiation, or moisture can result in reduced insulation resistance (IR). A reduction in IR is a concern for circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation circuits since it may contribute to signal inaccuracies. For radiation monitoring circuits and the Gamma Metrics circuits, the review of calibration results or findings of surveillance testing will be used to identify the potential existence of cable system aging degradation. This review will be performed at least once every 10 years, with the first review to be completed before the end of the current license term. Power range cable systems used in the Excore Monitoring System will be tested at a frequency not to exceed once every 10 years based on engineering evaluation, with the first testing to be completed before the end of the current license term. Testing may include IR tests, time domain reflectometry (TDR) tests, current versus voltage (I/V) testing, or other testing judged to be effective in determining cable system insulation condition. The aging effects of concern are as follows:

- Reduced Insulation Resistance
- Electrical Failure

The scope of this Program applies to non-EQ cable systems used in radiation monitoring instrumentation circuits and neutron flux monitoring instrumentation circuits that are sensitive to a reduction in IR. NUREG-1801 Section XI.E1 is not applicable to the cables utilized in these instrumentation circuits.

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 program consistent with NUREG-1801, Section XI.E2.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program is a new program with no site specific OE history. However, plant-specific and industrywide OE was considered in the development of all electrical programs in Appendix B of the CR-3 License Renewal Application. The review of plant-specific and industry-wide OE ensures that the corresponding NUREG-1801, Chapter XI, Program will be an effective aging management program for the period of extended operation. Plant-specific OE for non-EQ electrical cables and connections used in instrumentation circuits has been captured by a review of one or more of the following: (1) the Action Tracking database, (2) System Engineering Notebooks and System Health Reports, and (3) discussions with site engineering personnel. This effort also included a review of work management records and applicable site correspondence (Licensee Event Reports, etc.).

Industry OE that forms the basis for this program is included in the OE element of the corresponding NUREG-1801, Chapter XI, Program. Industry OE noted in NUREG-1801 has shown that exposure of electrical cables to adverse localized environments caused by heat, radiation, or moisture can result in reduced IR. Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive high voltage, low-level signals such as radiation monitoring and nuclear instrumentation circuits since it may contribute to signal inaccuracies.

This review of plant-specific and industry-wide OE confirms that the OE discussed in the corresponding NUREG-1801, Chapter XI, Program is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the CR-3 Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the CR-3 QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that all electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of 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 intended functions of non-EQ electrical cables and connections used in instrumentation circuits with sensitive high voltage, low-level signals exposed to adverse localized environments caused by heat, radiation, or moisture will be maintained consistent with the current licensing basis through the period of extended operation.

B.2.33 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 is credited for aging management of cables not included in the CR-3 EQ Program. In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested at least once every 10 years 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, polarization index, or other testing that is state-of-the-art at the time the test is performed. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Periodic exposures that last less than a few days (e.g., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage for more than 25% of the time. Manholes associated with inaccessible non-EQ medium-voltage cables will be inspected for water accumulation and drained, as needed. The manhole inspection intervals will be based on actual field data and shall not exceed two years. The first test and inspections for License Renewal will be completed before 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 program consistent with NUREG-1801, Section XI.E3.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program with no OE history. However, plant-specific and industry-wide OE was considered in the development of all electrical programs in Appendix B of the CR-3 License Renewal Application. The review of plant-specific and industry-wide OE ensures that the corresponding NUREG-1801, Chapter XI, Program will be an effective aging management program for the period of extended operation. Plant-specific OE for medium-voltage cables has been captured by a review of one or more of the following: (1) the Action Tracking database, (2) System Engineering Notebooks and System Health Reports, and (3) discussions with site engineering personnel. This effort also included a review of applicable site correspondence (Licensee Event Reports, etc.). Plant-specific OE for the medium voltage cables is captured in the CR-3 response to Generic Letter 2007-01, Serial: 3F0507-06 dated May 3, 2007.

Industry OE that forms the basis for this program is included in the OE element of the corresponding NUREG-1801, Chapter XI, Program. Industry OE noted in NUREG-1801 has shown that cross-linked polyethylene or high molecular weight polyethylene insulation materials are most susceptible to water tree formation. The formation and growth of water trees varies directly with operating voltage; for example, treeing is much less prevalent in 4KV cables than those operated at 13KV or 33KV. Also, minimizing exposure to moisture minimizes the potential for the development of water treeing.

This review of plant-specific and industry-wide OE confirms that the OE discussed in the corresponding NUREG-1801, Chapter XI, Program is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the CR-3 Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the CR-3 QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that all electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will provide reasonable assurance that the intended functions of inaccessible non-EQ medium-voltage cables exposed to adverse localized equipment environments caused by moisture while energized will be maintained consistent with the CLB through the period of extended operation.

B.2.34 METAL ENCLOSED BUS PROGRAM

Program Description

The Metal Enclosed Bus (MEB) Program is credited for aging management of all non-segregated 4.16KV and 250/125VDC MEB within the scope of License Renewal. The Program involves various activities conducted at least once every 10 years to identify the potential existence of aging degradation. In this Program, a sample of accessible bolted connections will be checked for loose connection by using thermography or by measuring connection resistance using a low range ohmmeter. In addition, the internal portions of the bus enclosure will be visually inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion. The bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The internal bus supports will be visually inspected for structural integrity and signs of cracks. The first test and inspections for license renewal will be completed before the period of extended operation.

As an alternative to thermography or measuring connection resistance of bolted connections, for the accessible bolted connections that are covered with heat shrink tape, sleeving, insulating boots, etc., visual inspection of the insulation material may be used to detect surface anomalies, such as discoloration, cracking, chipping or surface contamination. If this alternative visual inspection is used to check bolted connections, the first inspection will be completed before the period of extended operation and every five years thereafter.

NUREG-1801 Consistency

The Metal Enclosed Bus Program is a new program consistent with NUREG-1801, Section XI.E4.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

This is a new aging management program for MEB. Therefore, there is no existing site specific OE to validate the effectiveness of this program. However, plant-specific and industry-wide OE was considered in the development of all electrical programs in Appendix B of the CR-3 LR Application. The review of plant-specific and industry-wide

OE ensures that the corresponding NUREG-1801, Chapter XI, Program will be an effective aging management program for the period of extended operation. Plant-specific OE for MEB has been captured by a review of one or more of the following: (1) the Action Tracking database, (2) System Engineering Notebooks and System Health Reports, and (3) discussions with site engineering personnel. This effort also included a review of work management records, applicable site correspondence (Licensee Event Reports, etc.).

Industry OE that forms the basis for this program is included in the OE element of the corresponding NUREG-1801, Chapter XI, Program. Industry OE noted in NUREG-1801 has shown that failures have occurred on MEBs caused by cracked insulation and moisture or debris buildup internal to the MEB. Industry OE noted in NUREG-1801 has also shown that MEB exposed to appreciable ohmic or ambient heating during operation may experience loosening of bolted connections related to the repeated cycling of connected loads or of the ambient temperature environment. This phenomenon can occur in heavily loaded circuits (i.e., those exposed to appreciable ohmic heating or ambient heating) that are routinely cycled.

This review of plant-specific and industry-wide OE confirms that the OE discussed in the corresponding NUREG-1801, Chapter XI, Program is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the CR-3 Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the CR-3 QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that all electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of the MEB Program will provide reasonable assurance that the intended functions of the non-segregated MEB within the scope of License Renewal will be maintained consistent with the CLB through the period of extended operation.

B.2.35 FUSE HOLDER PROGRAM

Program Description

The Fuse Holder Program is credited for the aging management of fuse holders located outside of active devices that are susceptible to aging effects. Fuse holders inside an active device, such as switchgear, power supplies, power inverters, battery chargers, control panels and circuit boards are not within the scope of this Program. The Program focuses on the metallic clamp (or clip) portion of the fuse holder. The parameters monitored include corrosion and oxidation. Identified fuse holders within the scope of License Renewal will be tested at least once every 10 years. Testing may include thermography, contact resistance testing, or other appropriate testing to be determined prior to Program implementation. The first test for License Renewal will be completed before the period of extended operation.

NUREG-1801 Consistency

The Fuse Holder Program is a new program consistent with NUREG-1801, Section XI.E4 with exceptions.

Exceptions to NUREG-1801

<u>Program Elements Affected</u>

- **Parameters Monitored/Inspected**
  Loss of continuity due to corrosion and oxidation will be managed by Fuse Holder Program. Fatigue due to ohmic heating, thermal cycling, electrical transients, frequent manipulation, vibration, and chemical contamination are not applicable aging effects for CR-3 fuse holders located outside of active devices.

Enhancements

None.

Operating Experience

This is a new aging management program for fuse holders. Therefore, there is no existing site specific OE to validate the effectiveness of this program. However, plant-specific and industry-wide OE was considered in the development of all electrical programs in Appendix B of the CR-3 LR Application. The review of plant specific and industry-wide OE ensures that the corresponding NUREG-1801, Chapter XI, Program will be an effective aging management program for the period of extended operation. Plant-specific OE for fuse holders has been captured by a review of one or more of the following: (1) the Action Tracking database, (2) System Engineering Notebooks and System Health Reports, and (3) discussions with site engineering personnel. This effort also included a review of work management records and applicable site correspondence (Licensee Event Reports, etc.).

Industry OE that forms the basis for this program is included in the OE element of the corresponding NUREG-1801, Chapter XI, Program. NUREG-1801 notes that loosening of fuse holders and corrosion of fuse clips are aging mechanisms that, if left unmanaged, can lead to a loss of electrical continuity function. Also, as stated in NUREG-1760, fuse holders experience a number of age-related failures. The major concern is that failures of a deteriorated cable system (cables, connections including fuse holders, and penetrations) might be induced during accident conditions since they are not subject to the environmental qualification requirements of 10 CFR 50.49.

This review of plant-specific and industry-wide OE confirms that the OE discussed in the corresponding NUREG-1801, Chapter XI, Program is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the CR-3 Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the CR-3 QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that all electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of the Fuse Holder Program will provide reasonable assurance that the electrical continuity function of fuse holders within the scope of License Renewal located outside of active devices will be maintained consistent with the current licensing basis through the period of extended operation.

B.2.36 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

Program Description

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is credited for aging management of cable connections not included in the CR-3 EQ Program. The Program will be implemented as a one-time inspection on a representative sample of non-EQ cables connections within the scope of License Renewal prior to the period of extended operation to provide an indication of the integrity of the cable connections. The specific type of test performed will be determined prior to testing, and is to be a proven test for detecting loose connections, such as thermography, contact resistance testing, or other appropriate testing judged to be effective in determining cable connection integrity. This Program does not include high-voltage (>35KV) switchyard connections. The aging effect/ mechanism of concern is:

- Loosening of Cable Connections

The factors considered for sample selection are application (high, medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.) in both indoor and outdoor environments. The technical basis for the sample selections of cable connections to be tested will be provided.

The metallic parts of Metal Enclosed Bus connections are managed by the Metal Enclosed Bus Program as delineated in NUREG-1801, XI.E4, "Metal Enclosed Bus," and are, therefore, not included within the scope of the Program.

NUREG-1801 Consistency

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program consistent with NUREG-1801, Section XI.E6, with exceptions.

Exceptions to NUREG-1801

<u>Program Elements Affected</u>

- **Scope of Program**
  NUREG-1801, Rev. 1, AMP XI.E6 states that connections associated with cables in scope of License Renewal are part of this Program, regardless of their association with active or passive components. However, CR-3 has applied the clarification provided in proposed LR-ISG-2007-02 dated August 29, 2007, that revises the scope to include only external cable connections terminating at an

active device such as motor, motor control center, switchgear, or of a passive device such as a fuse cabinet. Wiring connections internal to an active assembly installed by manufacturers are considered a part of the active assembly; and, therefore, are not within the scope of this Program.

- **Detection of Aging Effects**
  NUREG-1801, Rev. 1, AMP XI.E6 specifies periodic testing of connections using thermography, contact resistance testing, or other appropriate testing methods. However, consistent with the test frequency flexibility provided in proposed LR-ISG-2007-02 dated August 29, 2007, this element will be implemented as a one-time inspection on a representative sample of non-EQ cable connections within the scope of License Renewal prior to the period of extended operation. Inspection methods may include thermography, contact resistance testing, or other appropriate testing methods. This one-time inspection verifies that the loosening of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging effect that requires a periodic aging management program.

Enhancements

None.

Operating Experience

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program with no site specific OE history. However, plant-specific and industry-wide OE was considered in the development of all electrical programs in Appendix B of the CR-3 License Renewal Application. The review of plant-specific and industry-wide OE ensures that the corresponding NUREG-1801, Chapter XI, Program will be an effective aging management program for the period of extended operation. Plant-specific OE for cable connections has been captured by a review of one or more of the following: (1) the Action Tracking database, (2) System Engineering Notebooks and System Health Reports, and (3) discussions with site engineering personnel. This effort also included a review of work management records, applicable site correspondence (Licensee Event Reports, etc.), and Nuclear Assessment Section assessment records.

Industry OE that forms the basis for this program is included in the OE element of the corresponding NUREG-1801, Chapter XI, Program. NUREG-1801 notes that loose terminations were identified by several plants. Industry OE noted in NUREG-1801 has also 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.

This review of plant-specific and industry-wide OE confirms that the OE discussed in the corresponding NUREG-1801, Chapter XI, Program is bounding, i.e., that there is no unique, plant-specific OE in addition to that in NUREG-1801. Going forward, OE will be captured through the CR-3 Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the CR-3 QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that all electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will provide reasonable assurance that electrical connections within the scope of License Renewal will be maintained consistent with the current licensing basis through the period of extended operation. This AMP will be effective for managing aging effects since it will incorporate proven monitoring techniques, acceptance criteria, corrective actions, and administrative controls.

B.2.37 CARBORUNDUM (B_4C) MONITORING PROGRAM

Program Description

The CR-3 Carborundum (B_4C) Monitoring Program is an existing program that monitors the effects of aging on the Carborundum (B_4C) panels that are located in the high density spent fuel storage racks in Spent Fuel Pool A.

Carborundum (B_4C) is a boron carbide shielding material utilized as a neutron absorber for the CR-3 spent fuel storage racks. Stability of the Carborundum (B_4C) supports the fuel storage pool Technical Specification criticality analysis requirement that the effective neutron multiplication factor (K_{eff}) of ≤ 0.95 must be maintained for all postulated events. The condition of the Carborundum (B_4C) poison material in the high density spent fuel racks located in Pool A is an indication of the K_{eff} of the fuel in the pool. The Program periodically removes and examines Carborundum (B_4C) poison samples from the pool to ensure that the K_{eff} is maintained below 0.95.

Aging Management Program Elements

The results of an evaluation of the aging management activities for the Carborundum (B_4C) Monitoring Program against the ten elements described in Appendix A of NUREG-1800 is provided below.

- **Scope of Program**
  The CR-3 Carborundum (B_4C) Monitoring Program monitors the effects of aging on the Carborundum (B_4C) panels located in the high density spent fuel storage racks in Spent Fuel Pool A.

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

- **Parameters Monitored/Inspected**
  The parameters monitored or inspected will verify that: (1) Carborundum (B_4C) sample coupons meet visual acceptance criteria and will be managed during the period of extended operation, and (2) Carborundum (B_4C) sample weight loss shall be within acceptable criteria and will be managed during the period of extended operation. The inspections monitor Carborundum (B_4C) samples that have been exposed to either: (1) gamma radiation dose plus borated water or (2) borated water alone to determine percentage weight loss of the sample. Based on the low percentage weight loss of Carborundum (B_4C) for sample inspections performed every five years; the inspection interval has been increased to nominally every 10 years.

- **Detection of Aging Effects**
  The Carborundum (B_4C) panels within the scope of this Program are to be inspected nominally every 10 years. This is an adequate period to detect aging effects before a loss of component intended function, since experience has shown that aging degradation for the Carborundum (B_4C) is a slow process. A 5-year nominal testing interval had been utilized up to 2004, and enough data has been accumulated to determine that the degradation (loss of material) rate is low enough to satisfy acceptance criteria through the period of extended operation.

- **Monitoring and Trending**
  Monitoring and trend data is incorporated in test procedures to be used to project and compare for upcoming sample testing. Trending of discrepancies is also performed (as required) in accordance with the Corrective Action Program. The Corrective Action Program is implemented by the CR-3 QA Program in accordance with 10 CFR 50, Appendix B. Prior to the period of extended operation, Program administrative controls will be revised to include provisions in the implementing procedure to monitor and trend weight loss data to ensure the projections meet acceptance criteria.

- **Acceptance Criteria**
  Inspection findings are to be within the acceptance criteria to ensure that the structure or component intended function(s) are maintained under all CLB design conditions during the period of extended operation. Program administrative controls contain the applicable acceptance criteria. Prior to the period of extended operation, Program administrative controls will be revised to incorporate acceptance criteria tables for accumulated weight losses of monitored Carborundum samples.

- **Corrective Actions**
  Corrective actions will be implemented through the CR-3 Corrective Action Program when inspection results do not meet the acceptance criteria. This program element is addressed in Subsection B.1.3.

- **Confirmation Process**
  This program element is addressed in Subsection B.1.3.

- **Administrative Controls**
  This program element is addressed in Subsection B.1.3.

- **Operating Experience**
  NUREG-1801 is based on industry OE through January 2005. NUREG-1801 and recent industry OE has been reviewed for applicability to CR-3. More recent OE is captured through the normal Operating Experience Program where it is

screened for applicability. This process will continue through the period of extended operation.

Plant-specific OE has also been reviewed. At CR-3, carborundum neutron absorber has been tested since 1984. Tests have been carried out on Carborundum (B_4C) sample coupons exposed to gamma dose plus borated water. Also, samples were tested that had been exposed to only borated water. A 5-year testing interval had been utilized up to 2004, and enough data has been accumulated to determine that the degradation rate is low enough to satisfy acceptance criteria through the period of extended operation. Additionally, because of the low degradation rate, the inspection interval has been increased to nominally every 10 years.

During 2004, OE included a failed sample for a weight loss of 21% (compared to 4% to 5% for the comparable samples). It was determined that the weight loss was a result of the material loss adjacent to the sample packet vent hole. It was also determined that the vent holes were above the active fuel length, therefore degradation opposite the vent holes would not result in neutron streaming and would have no effect on reactivity. Therefore, there are no adverse consequences from material degradation opposite the holes. The Spent Fuel Pool A criticality analysis remains valid. Also during 2004, a report was initiated concerning sample dose exposure. In 2001, during fuel movement for Spent Fuel Pool B re-rack, fuel was inadvertently moved away from the gamma sample holder. The total missed dose was estimated to be about 1% of the accumulated total dose on the samples. Since samples are exposed to accelerated gamma dose compared to the racks themselves; this 1% is considered insignificant. Therefore, the loss of exposure does not invalidate the spent fuel rack poison surveillance program.

A criticality analysis was performed for Spent Fuel Pool A. The analysis demonstrates that for the defined acceptance criteria, the maximum k_{eff} is less than 0.95 without credit for soluble boron at a 95% probability with a 95% confidence level. Under accident conditions, it is demonstrated that 165 ppm of soluble boron is required to assure that the maximum k_{eff} is less than or equal to 0.945. Technical Specification 3.7.14 requires maintaining a concentration of dissolved boron in the fuel pool ≥ 1925 ppm. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

Conclusion

Continued implementation of the Carborundum (B_4C) Monitoring Program will provide reasonable assurance that the Carborundum (B_4C) panels that are located in the storage racks of Spent Fuel Storage Pool A will perform their intended function consistent with the CLB through the period of extended operation.

B.2.38 HIGH-VOLTAGE INSULATORS IN THE 230KV SWITCHYARD PROGRAM

Program Description

The High-Voltage Insulators in the 230KV Switchyard Program is credited for the aging management of the high-voltage insulators used in the power path for the overhead transmission conductors that connect CR-3 230KV Switchyard to the Backup Engineered Safeguards Transformer (BEST). The Program inspects the insulators for salt deposits or surface contamination and mechanical wear of the steel hardware connecting the insulators to one another. The high-voltage insulators within the scope of this Program are to be inspected at least once every four years. The first inspections for License Renewal are to be completed prior to the period of extended operation.

Aging Management Program Elements

The results of an evaluation of the aging management activities for the High-Voltage Insulators in the 230KV Switchyard Program against the ten elements described in Appendix A of NUREG-1800 is provided below.

- **Scope of Program**
  This Program applies to high-voltage insulators used in the power path for the overhead transmission conductors that connect CR-3 230KV Switchyard to the Backup Engineered Safeguards Transformer (BEST).

- **Preventive Actions**
  No actions are taken as part of this inspection Program to prevent or mitigate aging degradation.

- **Parameters Monitored/Inspected**
  The following parameters will be monitored/inspected to ensure component intended function during the period of extended operation: 1) evidence of salt deposits or surface contamination, and 2) mechanical wear of the steel hardware connecting the insulators to one another.

- **Detection of Aging Effects**
  The high-voltage insulators within the scope of this Program are to be inspected at least once every four years. This is an adequate frequency to detect aging effects before a loss of component intended function since experience has shown that aging degradation is a slow process. A four-year inspection interval will provide multiple data points during a 20-year period, which can be used to characterize the degradation rate. The first inspection for License Renewal is to be completed prior to the period of extended operation.

- **Monitoring and Trending**
  Trending actions are not part of this Program. However, trending of discrepancies is performed (as required) in accordance with the CR-3 Corrective Action Program. The Corrective Action Program is implemented by the CR-3 QA Program in accordance with 10 CFR 50, Appendix B.

- **Acceptance Criteria**
  Inspection results are to be within the acceptance criteria to ensure component intended function(s) are maintained under all CLB design conditions during the period of extended operation. Acceptance criteria will be delineated in the applicable inspection procedure.

- **Corrective Actions**
  Corrective actions will be implemented through the CR-3 Corrective Action Program when inspection results do not meet the acceptance criteria. Corrective actions may include, but are not limited to, washing, replacing or repairing affected high-voltage insulators. This program element is addressed in Subsection B.1.3.

- **Confirmation Process**
  This program element is addressed in Subsection B.1.3.

- **Administrative Controls**
  This program element is addressed in Subsection B.1.3.

- **Operating Experience**
  The High-Voltage Insulators in the 230KV Switchyard Program is a new program with no site specific OE history. However, plant-specific and industry-wide OE was considered in the development of all electrical programs in Appendix B of the CR-3 License Renewal Application. The review of plant-specific and industry-wide OE ensures that this will be an effective aging management program for the period of extended operation. Plant-specific OE for high voltage insulators has been captured by a review of one or more of the following: (1) the Action Tracking database, (2) System Engineering Notebooks and System Health Reports, and (3) discussions with site engineering personnel. This effort may have also included a review of work management records, applicable site correspondence (Licensee Event Reports, etc.), and Nuclear Assessment Section assessment records. The following are plant-specific OE applicable to this Program:

  - On March 17, 1993, CR-3 experienced a loss of the 230KV switchyard (i.e., a loss of offsite power) when a light rain caused arcing across salt-laden 230KV insulators and opened switchyard breakers. This event was the subject of NRC Information Notice NRC 93-95, dated December 13, 1993: "Storm-

Related Loss of Offsite Power Events Due to Salt Buildup on Switchyard Insulators."

- On September 6, 2004, CR-3 experienced phase-to-ground faults concurrently on a 230KV transmission line and a 230KV switchyard south bus breaker during Tropical Storm Frances. The transmission line fault was caused by mechanical failure of a carbon steel pin in a vertical string of insulators due to high wind conditions. The breaker fault was caused by flashover due to contamination from wind and salt spray.

Site- and industry-wide OE that forms the basis for this program has shown that various airborne materials such as dust, salt, and industrial effluents can contaminate insulator surfaces. A large buildup of contamination enables the conductor voltage to track along the surface more easily and can lead to insulator flashover. Operating experience has also shown that mechanical wear of the steel pins connecting the insulators to one another could lead to a loss of intended function if the insulators are subject to significant or prolonged movement.

Going forward, OE will be captured through the CR-3 Corrective Action and Operating Experience Programs implemented in accordance with Progress Energy corporate procedures. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. The administrative controls that implement the Corrective Action and Operating Experience Programs are implemented in accordance with the CR-3 QA Program, which is in conformance with 10 CFR 50, Appendix B. This process will verify that the electrical programs credited for License Renewal will continue to be effective in the management of aging effects.

Conclusion

Implementation of the High-Voltage Insulators in the 230KV Switchyard Program will provide reasonable assurance that high-voltage insulators will perform their intended function consistent with the current licensing basis through the period of extended operation.

B.3 TIME-LIMITED AGING ANALYSIS PROGRAMS

B.3.1 REACTOR COOLANT PRESSURE BOUNDARY FATIGUE MONITORING PROGRAM

Program Description

The Reactor Coolant Pressure Boundary (RCPB) Fatigue Monitoring Program includes preventive measures to mitigate fatigue cracking caused by anticipated cyclic strains in metal components of the RCPB. This is accomplished by monitoring and tracking the significant thermal and pressure transients for limiting RCPB components in order to prevent the fatigue design limit from being exceeded. The RCPB Fatigue Monitoring Program addresses the effects of the reactor coolant environment on component fatigue life by including, within the Program scope, environmental fatigue evaluations of the sample locations specified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." These locations were evaluated by applying environmental correction factors to ASME Section III, Class 1 fatigue analyses, as specified in NUREG/CR-6583,"Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," and NUREG/CR-6717, "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels." Prior to exceeding the design limit, preventive and/or corrective actions are triggered by the Program. CR-3 has ensured that the effects of the reactor water environment on fatigue-sensitive locations have been addressed and are managed for the period of extended operation.

NUREG-1801 Consistency

The RCPB Fatigue Monitoring Program is an existing program that is consistent with NUREG-1801, Section X.M1.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

A review of NRC Generic Communications (Information Notices, Bulletins, Generic Letters, and Draft Generic Communication), the INPO OE database, and Licensee Event Reports was performed, but no applicable OE items were identified that relate to fatigue monitoring or to exceeding fatigue design limits. Assessments on the implementation of EPRI "Good Practice" documents released in accordance with the

NEI 03-08 materials initiative protocol related to fatigue were reviewed. The assessments revealed that the Program is in accordance with the "Good Practice" recommendations. The Program has been effective documenting transients and cycles on applicable systems and components so that the limits imposed by Technical Specifications and the FSAR are not exceeded.

Conclusion

Continued use of the RCPB Fatigue Monitoring Program will provide reasonable assurance that the fatigue design limits for applicable components/commodities will not be exceeded such that the components/commodities will continue to perform their intended functions consistent with the CLB for the period of extended operation.

B.3.2 ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM

Program Description

The EQ Program is an existing CR-3 engineering program. The EQ Program manages component thermal, radiation, and cyclical aging through the use of aging analysis based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished or replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered Time-Limited Aging Analyses (TLAAs) for CR-3 License Renewal. Refer to Section 4.4 for a discussion of EQ Program reanalysis attributes.

NUREG-1801 Consistency

The EQ Program is an existing program consistent with NUREG-1801, Section X.E1.

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The CR-3 EQ Program has been effective at managing aging effects. As stated in NUREG-1801, EQ programs include consideration of OE 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. Although the EQ Program System Health Report shows the program has a "Yellow" indication status, the overall effectiveness of the program is demonstrated by the "Green" status under key performance areas such as regulatory compliance, overall program effectiveness, and performance. The green program status is the highest ranking available and shows the Program's strength in these areas. 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.

Conclusion

The overall effectiveness of the EQ Program is demonstrated by the excellent OE for systems and components in the Program. The Program has been subject to periodic internal and external assessments that facilitate continuous improvement. Continued implementation of the CR-3 EQ Program provides reasonable assurance that aging effects will be managed such that the components within the scope of License Renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

APPENDIX C

IDENTIFYING AGING EFFECTS BY MATERIAL AND ENVIRONMENT

Appendix C is not being used in this application.

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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 CR-3 Technical Specifications are required to support the License Renewal Application. Therefore, Appendix D is not being used in this application.

TABLE 3.2.2-3 (continued) ENGINEERED SAFETY FEATURES – SUMMARY OF AGING MANAGEMENT EVALUATION – DECAY HEAT REMOVAL SYSTEM

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping, piping components, and piping elements	M-1	Cast Austenitic Stainless Steel	Reactor Coolant (Inside)	Cumulative Fatigue Damage due to Fatigue	TLAA	IV.C2-10 (R-18)	3.1.1-07	A
				Loss of Material due to Crevice Corrosion Loss of Material due to Pitting Corrosion	Water Chemistry	IV.C2-15 (RP-23)	3.1.1-83	A
				Loss of Fracture Toughness due to Thermal Embrittlement	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	IV.C2-6 (R-08)	3.1.1-55	A
				Cracking due to SCC	ASME Section XI Inservice Inspection and Water Chemistry	IV.C2-5 (R-09)	3.1.1-68	A
			Air - Indoor Uncontrolled (Outside)	None	None	IV.E-2 (RP-04)	3.1.1-86	A
		Stainless Steel	Treated Water (Inside)	Cracking due to SCC	Water Chemistry	V.D1-31 (E-12)	3.2.1-48	A
				Cumulative Fatigue Damage due to Fatigue	TLAA	V.D1-27 (E-13)	3.2.1-01	A
				Loss of Material due to Crevice Corrosion Loss of Material due to Pitting Corrosion	Water Chemistry	V.D1-30 (EP-41)	3.2.1-49	A

TABLE 3.2.2-3 (continued) ENGINEERED SAFETY FEATURES – SUMMARY OF AGING MANAGEMENT EVALUATION – DECAY HEAT REMOVAL SYSTEM

Component/ Commodity	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping, piping components, and piping elements (continued)	M-1	Stainless Steel	Air - Indoor Uncontrolled (Outside)	None	None	V.F-12 (EP-18)	3.2.1-53	A
			Air - Outdoor (Outside)	Loss of Material due to Crevice Corrosion Loss of Material due to Pitting Corrosion	External Surfaces Monitoring			J

Notes for Tables 3.2.2-1 through 3.2.2-3:

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 AMP is credited or NUREG-1801 identifies a plant-specific AMP.
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:

201. A raw water environment has been assumed for abandoned equipment associated with Sodium Hydroxide tanks.
202. The components are located in areas that do not contain borated water systems.
203. The RB Spray and Decay Heat Pumps have jacketed bearing housings and stuffing box heads; these jackets are supplied with cooling from the DC System.
204. Aging effects are associated with the exposed dome of the BWST. The wall of the tank is sealed within a protective concrete cylinder, while the floor is in contact with a grouted base.
205. This item addresses nickel alloy nozzles and welds associated with the Core Flood Tanks.

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