Attachment - Fourteen

FPL Application for Renewed Operating Licenses
(802-pages)

APPLICATION FOR RENEWED OPERATING LICENSES

TURKEY POINT UNITS 3 & 4

TABLE OF CONTENTS

APPENDICES

LIST OF TABLES

PREFACE

The following discussion describes the content of the Turkey Point Units 3 and 4 License Renewal Application.

Chapter 1 provides the administrative information required by Part 54 of Title 10 of the Code of Federal Regulations; Sections 17 and 19 (10 CFR 54.17 and 10 CFR 54.19).

Chapter 2 provides the scoping and screening methodology. Chapter 2 describes and justifies the methodology used to determine the systems, structures, and components within the scope of license renewal and the structures and components subject to an aging management review. Tables 2.2-1, 2.2-2, and 2.2-3 provide a listing of the plant mechanical systems, structures, and electrical/I&C systems respectively, and these tables identify those plant systems and structures that are within the scope of license renewal. Chapter 2 provides a description of systems, intended functions, and references to system boundary drawings. Tables 2.3-1, 2.3-4, 2.3-5, and 2.3-6 show the drawing numbers for the mechanical systems in the scope of license renewal. The drawings are provided in a separate submittal. Tables in Chapter 3 are referenced in Chapter 2.

Chapter 3 describes the results of the aging management reviews of the components and structures requiring aging management reviews. Furthermore, Chapter 3:

- identifies the components and structures subject to aging management review and their intended functions,
- describes or references the processes used to identify aging effects requiring management (Appendix C summarizes the process used to identify aging effects associated with non-Class 1 components, which encompasses engineered safety features system components, auxiliary system components, steam and power conversion system components, and steel in fluid structural components),
- discusses the materials and environments which produce aging effects,
- identifies the aging effects requiring management,
- describes industry and plant-specific operating experiences with respect to the applicable aging effects, and
- identifies the aging management programs that will manage the aging effects requiring management.

The aging management programs and the information necessary to demonstrate that the aging effects requiring management will be adequately managed are described in Appendix B. The tables in Chapter 3 provide a comprehensive summary of information concerning the aging effects requiring management for component and commodity groupings in the scope of license renewal. For the component and commodity groupings that make up the system or structure, the tables list intended function, material, environment, aging effects, and the aging management programs and activities.

Chapter 4 includes a list of time-limited aging analyses, as defined by 10 CFR 54.3. It includes the identification of the component or subject, and an explanation of the time-dependent aspects of the calculation or analysis. Chapter 4 demonstrates that the analyses remain valid for the period of extended operation, the analyses have been projected to the end of the period of extended operation, or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Chapter 4 also states that no 10 CFR 50.12 exemption involving a time-limited aging analysis as defined in 10 CFR 54.3 is required during the period of extended operation.

Appendix A, Updated Final Safety Analysis Report Supplement, provides a summary description of the programs for managing the effects of aging for the period of extended operation. A summary description of the evaluation of time-limited aging analyses for the period of extended operation is also included.

Appendix B, Aging Management Programs, describes the aging management programs and activities and demonstrates that the aging effects on the components and structures within the scope of the License Renewal Rule will be managed such that they will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. The Turkey Point Units 3 and 4 programs and activities that are credited for managing aging are divided into new actions and existing actions.

Appendix C, Process for Identifying Aging Effects Requiring Management for Non-Class 1 Components, summarizes the process through which the applicable aging effects were identified and associated with the non-Class 1 components determined to be subject to an aging management review.

Appendix D, Technical Specification Changes, concludes that no technical specification changes are necessary to manage the effects of aging during the period of extended operation.

The information in Chapter 2, Chapter 3, and Appendix B fulfills the requirements in 10 CFR 54.21(a). Section 1.4 discusses how the requirements of 10 CFR 54.21(b) will be met.

The information in Chapter 4 fulfills the requirements in 10 CFR 54.21(c). The information in Appendix A and Appendix D fulfills the requirements in 10 CFR 54.21(d) and 10 CFR 54.22, respectively. The supplement to the Environmental Report, as required by 10 CFR 54.23, is provided with the Turkey Point Units 3 and 4 License Renewal Application as a separate document.

1.0 ADMINISTRATIVE INFORMATION

1.1 PURPOSE AND GENERAL INFORMATION

Pursuant to Part 54 of Title 10 of the Code of Federal Regulations (10 CFR 54), this Application seeks renewal for an additional 20-year term of the facility operating licenses for Turkey Point Unit 3 (DPR-31) and Unit 4 (DPR-41). The Unit 3 operating license (DPR-31) currently expires at midnight, July 19, 2012. The Unit 4 operating license (DPR-41) expires at midnight, April 10, 2013. The application includes renewal of the source, special nuclear, and byproduct materials licenses that are combined in the Unit 3 and Unit 4 licenses.

The application is organized in accordance with the United States Nuclear Regulatory Commission (NRC) “Draft Standard Format For License Renewal Application,” August 9, 1999, and is consistent with the guidance provided by NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR 54 - License Renewal Rule," Revision 1.

The environmental information required by 10 CFR 54.23 is provided as a separate report entitled, “Applicant’s Environmental Report – Operating License Renewal Stage, Turkey Point Units 3 and 4.”

This License Renewal Application and its supporting Environmental Report are intended to provide sufficient information for the NRC to complete its technical and environmental reviews. The License Renewal Application and Environmental Report are designed to allow the NRC to make the finding required by 10 CFR 54.29 in support of the issuance of renewed operating licenses for Turkey Point Units 3 and 4. Following is the general information required by 10 CFR 54.17 and 10 CFR 54.19.

1.1.1 NAME OF APPLICANT

Florida Power & Light Company

1.1.2 ADDRESS OF APPLICANT

Florida Power & Light Company
700 Universe Boulevard
Post Office Box 14000
Juno Beach, Florida 33408-0420

Address of the Turkey Point Nuclear Plant:

Florida Power & Light Company
Turkey Point Nuclear Plant
9760 SW 344th Street
Florida City, Florida 33035-1800

1.1.3 OCCUPATION OF APPLICANT

Florida Power & Light Company (FPL) is an investor-owned utility, primarily engaged in the generation, transmission, and distribution of electricity. The service territory covers the southern third and almost the entire eastern seaboard of the State of Florida. FPL supplies electric service to more than 3.7 million residential, commercial, and industrial customers. To service this area, FPL operates 14 electric generating facilities with an installed capacity of over 16,000 megawatts (MW) electric, including the Turkey Point Nuclear Plant.

1.1.4 ORGANIZATION AND MANAGEMENT OF APPLICANT

FPL is a public utility incorporated under the laws of the State of Florida, with its principal office located in Juno Beach, Florida.

FPL is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. FPL makes this application on its own behalf and is not acting as an agent or representative of any other person.

The names and business addresses of FPL's directors and principal officers are listed below. All persons listed are U.S. citizens.

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1.1.5 CLASS AND PERIOD OF LICENSE SOUGHT

FPL requests renewal of the Class 104b operating licenses for Turkey Point Units 3 and 4 (license numbers DPR-31 and DPR-41, respectively) for a period of 20 years beyond the expiration of the current licenses. For Turkey Point Unit 3 (DPR-31), license renewal would extend the operating license from midnight July 19, 2012, until midnight July 19, 2032. For Turkey Point Unit 4 (DPR-41), license renewal would extend the operating license from midnight April 10, 2013, until midnight April 10, 2033. This application includes a request for renewal of those NRC source material, special nuclear material, and byproduct material licenses that are currently subsumed into or combined with the current operating licenses.

The facility will continue to be known as the Turkey Point Nuclear Plant and will continue to generate electric power during the renewal period.

1.1.6 ALTERATION SCHEDULE

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

1.1.7 CONFORMING CHANGES TO THE STANDARD INDEMNITY AGREEMENT

The requirements at 10 CFR 54.19(b) state that license renewal applications include, "…conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." The current indemnity agreement for Turkey Point Units 3 and 4 states, in Article VII, that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the Attachment to the agreement, which is the last to expire. Item 3 of the Attachment to the indemnity agreement, as revised by Amendment No. 5, lists four license numbers. Should the license numbers be changed upon issuance of the renewed licenses, FPL requests that conforming changes be made to Item 3 of the Attachment, and any other sections of the indemnity agreement as appropriate.

1.1.8 RESTRICTED DATA AGREEMENT

This application does not contain any Restricted Data or National Security Information, and FPL does not expect that any activity under the renewed licenses for Turkey Point Units 3 and 4 will involve such information. However, if such information were to become involved, FPL agrees that it would appropriately safeguard such information and would not permit any individual to have access to, or any facility to possess, such information until the individual or facility had been approved under the provisions of 10 CFR 25 or 10 CFR 95, respectively.

1.2 DESCRIPTION OF TURKEY POINT NUCLEAR PLANT

The two nuclear power units designated as Turkey Point Units 3 and 4 are located adjacent to oil- and gas-fired Units 1 and 2 at the Turkey Point Plant. This is a steam electric generating facility situated on the shore of Biscayne Bay, about 25 miles south of Miami, Florida.

The Turkey Point Units 3 and 4 reactors are Westinghouse designed, pressurized light-water moderated and cooled systems. Each is designed to produce a core thermal power output of 2300 MWt. Each steam and power conversion system, including its turbine generator, is designed to permit generation of a net electrical output of approximately 693 MW. The units were uprated in 1996 from an initial core thermal output of 2200 MWt.

Descriptions of Turkey Point Units 3 and 4 systems and structures can be found in the Updated Final Safety Analysis Report (UFSAR). Additional descriptive information about Turkey Point Units 3 and 4 systems, structures, and components is provided in Chapters 2, 3, and 4 of this Application, and references to the UFSAR are provided where pertinent.

1.3 TECHNICAL INFORMATION REQUIRED FOR AN APPLICATION

In accordance with 10 CFR 54.21, four technical items are required to support an application for a renewed operating license. These are an integrated plant assessment (Chapters 2 and 3), an evaluation of time-limited aging analyses (Chapter 4), a supplement to the Turkey Point Units 3 and 4 UFSAR that contains a summary description of the programs and activities for managing the effects of aging and the evaluation of the time-limited aging analyses (Appendix A), and current licensing basis changes during NRC review (Section 1.4).

In addition to the technical information, 10 CFR 54.22 requires applicants to submit any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation (Appendix D). Also, 10 CFR 54.23 requires the Application to include a supplement to the Environmental Report (Applicant's Environmental Report – Operating License Renewal Stage).

The Integrated Plant Assessment (IPA), as defined by 10 CFR 54.3, is a licensee assessment that demonstrates that a nuclear power plant facility's structures and components requiring aging management review in accordance with 10 CFR 54.21(a) for license renewal have been identified. The IPA also demonstrates that the effects of aging on the functionality of such structures and components will be managed to maintain the current licensing basis during the period of extended operation. The Turkey Point Units 3 and 4 IPA includes:

- identification of the structures and components within the scope of license renewal that are subject to an aging management review;
- identification of the aging effects applicable to these structures and components;
- identification of plant-specific programs and activities that will manage these identified aging effects; and
- a demonstration that these programs and activities will be effective in managing the effects of aging during the period of extended operation.

The Turkey Point Units 3 and 4 IPA for license renewal, along with other information necessary to document compliance with 10 CFR 54, is maintained in an auditable and retrievable form in accordance with 10 CFR 54.37(a). The Turkey Point Units 3

and 4 IPA is documented with site-specific reports and calculations that were generated in accordance with FPL's Quality Assurance Program. Also, note that references to the Turkey Point Units 3 and 4 Technical Specifications and the UFSAR are as of Amendments 205/199 and Amendment 16, respectively.

1.4 CURRENT LICENSING BASIS CHANGES DURING NRC REVIEW

Each year, following the submittal of the Turkey Point Units 3 and 4 License Renewal Application and at least three months before the scheduled completion of the NRC review, Turkey Point will submit amendments to the Application pursuant to 10 CFR 54.21(b). These revisions will identify any changes to the current licensing basis that materially affect the contents of the License Renewal Application, including the UFSAR supplement and any other aspects of the Application.

2.0 STRUCTURES AND COMPONENTS SUBJECT TO AN AGING MANAGEMENT REVIEW

This chapter describes the process for the identification of structures and components subject to an aging management review in the Turkey Point integrated plant assessment (IPA). For those systems, structures, and components (SSCs) within the scope of license renewal, 10 CFR 54.21(a)(1) requires a license renewal applicant to identify and list the structures and components subject to an aging management review. Furthermore, 10 CFR 54.21(a)(2) requires that the methods used to identify and list these structures and components be described and justified. The technical information in this chapter serves to satisfy these requirements.

Turkey Point's IPA methodology follows the approach recommended in NEI 95-10 [Reference 2.1-1]. The methodology consists of scoping, screening, and aging management reviews. The methodology is implemented in accordance with FPL's Quality Assurance Program.

Scoping and screening methodology is described in Section 2.1. The results of the assessment to identify the systems and structures within the scope of license renewal (plant level scoping) are contained in Section 2.2. The results of the identification of the components and structural components subject to an aging management review (screening) are contained in Section 2.3 for mechanical systems, Section 2.4 for structures, and Section 2.5 for electrical/instrumentation and control (I&C) systems.

2.1 SCOPING AND SCREENING METHODOLOGY

Scoping is the evaluation performed to identify SSCs that satisfy the criteria in 10 CFR 54.4. Based on the nature and content of design information systems at Turkey Point, scoping as defined in 10 CFR 54.4 was performed in two steps: (1) plant level scoping, and (2) component and structural component scoping. For the first step, an evaluation was performed to identify systems and structures that satisfy the criteria in 10 CFR 54.4. This is designated as plant level scoping and is described in Subsection 2.1.1. For the second step, the systems and major structures identified as satisfying the criteria in 10 CFR 54.4 were further evaluated to identify the specific components and structural components that satisfy the criteria in 10 CFR 54.4 and, therefore, are in the scope of license renewal.

Once the in-scope components and structural components were identified, they were screened to identify those subject to an aging management review in accordance with 10 CFR 54.21(a)(1). The component and structural component scoping and screening process is described in Subsection 2.1.2.

2.1.1 PLANT LEVEL SCOPING

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

Specifically, 10 CFR 54.4 states that:

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

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

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR Part 100 guidelines.

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

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

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

The scoping process to identify systems and structures that satisfy the requirements of 10 CFR 54.4(a)(1), 10 CFR 54.4(a)(2), and 10 CFR 54.4(a)(3) is performed on systems and structures using documents which form the Current Licensing Basis (CLB) and other information sources. The CLB for Turkey Point Units 3 and 4 has been defined in accordance with the definition provided in 10 CFR 54.3. The key information sources that form the CLB include the UFSAR, Technical Specifications, and the docketed licensing correspondence. Other important information sources used for scoping are further described in Subsection 2.1.1.1.

The scoping process utilized by Turkey Point considers the guidance provided by the NRC in its letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI) dated August 5, 1999, entitled, “License Renewal Issue No. 98-0082, Scoping Guidance" [Reference 2.1-2].

The aspects of the scoping process used to identify systems and structures that satisfy the requirements of 10 CFR 54.4(a)(1), 10 CFR 54.4(a)(2), and 10 CFR 54.4(a)(3) are described in Subsections 2.1.1.2, 2.1.1.3, and 2.1.1.4 respectively.

2.1.1.1 INFORMATION SOURCES

In addition to the UFSAR, Technical Specifications, and docketed licensing correspondence, three information sources – the design basis documents, the component database, and piping and instrumentation diagrams (P&IDs) – were relied upon to a great extent in performing scoping and screening for Turkey Point. A brief discussion of these sources is provided.

2.1.1.1.1 DESIGN BASIS DOCUMENTS

In response to a NRC Safety System Functional Inspection on the Turkey Point Auxiliary Feedwater System, performed in August 1985, design basis documents were prepared for eighteen support and accident mitigation systems, selected licensing issues, and UFSAR Chapter 14 accident analyses. Design basis documents are a tool to explain the requirements behind the design rather than describing the design itself. Design basis documents are intended to complement other upper tier documents, such as the UFSAR and Technical Specifications, and are controlled and updated.

2.1.1.1.2 COMPONENT DATABASE

Specific component information for SSCs at Turkey Point can be found in the controlled component database. The controlled component database contains as-built information on a component level. The component database consists of multiple data fields for each component, such as design-related information, safety and seismic classifications, safety classification bases, and component tag, type, and description.

2.1.1.1.3 P&IDs

Turkey Point was designed and built prior to the issuance of present day nuclear power plant guidance documents for American Society of Mechanical Engineers (ASME) Code boundaries and quality group classifications. Quality group classifications for Turkey Point were established considering the Turkey Point CLB and various industry codes and standards, including Regulatory Guide 1.26, 10 CFR 50.55a, and ASME Section XI. Quality group classification boundaries for safety-related systems are delineated on P&IDs and provide a basis for ASME Section XI programs.

Various reference documents refer to “ASME Section III Code Class 1, 2, and 3,” or “Safety Class 1, 2, and 3” for safety-related components. The corresponding classifications reflected on the P&IDs for Turkey Point Units 3 and 4 are uniformly referred to as “Quality Group A, B, and C.” The classification "SR" has been used to identify those systems or portions of systems that are important to safety, but for which there are no specific commitments contained within the ASME Section XI program.

2.1.1.2 SAFETY-RELATED CRITERIA PURSUANT TO 10 CFR 54.4(a)(1)

10 CFR 54.4(a)(1) states that SSCs within the scope of license renewal include safety-related SSCs that are relied upon to remain functional during and following design basis events [as defined in 10 CFR 50.49(b)(1)] to ensure the following functions:

- the integrity of the reactor coolant pressure boundary;
- the capability to shut down the reactor and maintain it in a safe shutdown condition; or

- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR Part 100 guidelines.

In the mid-1980s, Turkey Point established safety classifications for systems and structures at the component level consistent with the definition of safety-related SSCs provided in the FPL Quality Assurance Program and the Turkey Point CLB. This definition of safety related encompasses the definition of safety related specified in 10 CFR 54.4(a)(1).

Safety classifications of SSCs were included in the component database and were established based on reliance on the SSCs during and following design basis events, which include design basis accidents, anticipated operational occurrences, natural phenomena, and external events. The design basis events considered are consistent with the Turkey Point CLB. UFSAR Chapter 14 provides the design basis event accident analyses for Turkey Point Units 3 and 4.

Natural phenomena and external events are described in Chapter 2 of the UFSAR and in appropriate sections of the design basis documents. Structures designed to withstand design basis events, natural phenomena, and external events are described in UFSAR Chapter 5.

Two of the design basis events, Accidental Liquid Release and Accidental Gas Release, are analyzed for offsite radiological consequences and do not involve analyses related to the reactor coolant pressure boundary or the capability to shut down the reactor and maintain it in a safe shutdown condition. Table 2.1-1 provides the radiological consequences of these design basis events from UFSAR Subsections 14.2.2 and 14.2.3 The offsite dose analyses indicate that the radiological consequences of Accidental Liquid Release and Accidental Gas Release are small fractions of 10 CFR 100 limits. As a result, the SSCs related to the prevention and/or mitigation of these design basis events do not meet the scoping criteria of 10 CFR 54.4 (a)(1)(iii). However, these SSCs were evaluated for possible inclusion in the license renewal scope relative to the criteria of 10 CFR 54.4(a)(2) and (a)(3).

The steps to identify systems and structures at Turkey Point that meet the criteria of 10 CFR 54.4(a)(1) are outlined below:

- The UFSAR, Technical Specifications, licensing correspondence, design basis documents, component database, and design drawings were reviewed, as applicable.
- Based on the above, license renewal intended functions relative to the criteria of 10 CFR 54.4(a)(1) were identified for each system and structure determined to be safety related.

The scoping process to identify safety-related systems and structures for Turkey Point is consistent with and satisfies the criteria in 10 CFR 54.4(a)(1).

2.1.1.3 NON-SAFETY RELATED CRITERIA PURSUANT TO 10 CFR 54.4(a)(2)

10 CFR 54.4(a)(2) states that SSCs within the scope of license renewal include non-safety related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified for safety-related SSCs.

At Turkey Point, non-safety related SSCs whose failure could impact safety-related SSCs carry an augmented quality classification (Quality Related) and are included in the FPL Quality Assurance Program. The non-safety related SSCs that are within the scope of license renewal for Turkey Point fall into two categories:

- Non-safety related SSCs that functionally support the operation of safety-related SSCs, and
- Non-safety related SSCs whose failure could cause an interaction with safety-related SSCs and potentially result in the failure of the safety-related SSCs to perform their intended safety function(s).

With regard to non-safety related SSCs that functionally support the operation of safety-related SSCs, there are several systems and structures in this category, including non-safety related ventilation systems that cool safety-related areas, and non-safety related piping segments that provide structural support at safety-related/non-safety related boundaries. SSCs associated with these systems and structures are classified “Quality Related” in the component database.

For non-safety related piping segments, safety-related/non-safety related functional boundaries for piping systems are made at system pressure boundary valves. The structural integrity boundary may extend beyond the system pressure boundary valve. The structural integrity support system includes the piping segments and supports that provide structural support for the boundary valve. These components ensure the integrity of the safety-related/non-safety related functional system

pressure boundary under all design basis loading conditions and are conservatively assumed to meet the scoping criteria of 10 CFR 54.4(a)(2).

The second category involves the potential for non-safety related systems or structures to impact the ability of safety-related systems or structures to perform their intended functions. To complete this portion of the scoping effort, a systematic review of potential non-safety related/safety-related interactions was performed. The UFSAR, licensing correspondence, and design basis documents were relied upon in addressing these interactions. For most of the potential interactions, failure of the non-safety related system or structure is assumed to occur, and design features are provided to accommodate the failure. Examples include internal flooding (protective design feature: sump pumps and drainage), and internal missiles (protective design feature: buildings, missile barriers, and enclosures). In these situations, the design features are considered to be in the scope of license renewal, not the non-safety related system or structure that is assumed to fail.

For other potential interactions, the non-safety related system or structure has the design capability to preclude the interaction with safety-related systems or structures. The primary interaction for this case is related to seismic design. Turkey Point's approach to scoping and screening of non-safety related systems or structures that have the potential for seismic interaction with safety related systems or structures is described in more detail below.

Non-seismic systems or structures that are positioned above or in close proximity to safety-related systems or structures, and whose failure during a seismic event could cause the subsequent failure of the safety-related systems or structures, are commonly referred to as "seismic II over I" or seismic interaction. It is important to note that Turkey Point Units 3 and 4 were not originally licensed for "seismic II over I." However, "seismic II over I" was considered for license renewal scoping.

For seismic interactions, Turkey Point has chosen an area-based approach to scoping, because the seismic interaction design feature is dependent upon the location of the non-safety related system or structure relative to safety-related systems and structures. The approach utilized identifies the major structures of the plant containing both safety-related and non-safety related systems and structures. Component and structural component level scoping performed as part of the screening process (see Subsection 2.1.2.2) then establishes the specific non-safety related seismic interaction component/structural component types located within these structures for inclusion in the license renewal scope. Based on this approach,

non-safety related components and structural components with the potential for seismic interactions are identified as in the scope of license renewal.

The steps to identify non-safety related systems and structures at Turkey Point that meet the criteria of 10 CFR 54.4(a)(2) are outlined below:

- The UFSAR, Technical Specifications, licensing correspondence, design basis documents, component database, pipe stress analyses, and design drawings were reviewed, as applicable.
- Based on the above, license renewal intended functions relative to the criteria of 10 CFR 54.4(a)(2) were identified for each system and structure determined to be non-safety related whose failure could affect safety-related SSCs.

The scoping process to identify non-safety related systems and structures whose failure can affect safety-related systems and structures for Turkey Point is consistent with and satisfies the criteria in 10 CFR 54.4(a)(2).

2.1.1.4 OTHER SCOPING PURSUANT TO 10 CFR 54.4(a)(3)

10 CFR 54.4(a)(3) states that SSCs within the scope of license renewal include all systems and structures relied on in safety analyses or plant evaluations to demonstrate compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

Scoping based on each of these regulations is described in the following sections.

2.1.1.4.1 FIRE PROTECTION (FP)

Fire protection features and commitments are described in detail in Appendix 9.6A of the UFSAR and the design basis documents. The systems and structures at Turkey Point that support the multiple levels of protection for postulated fires are considered within the scope of license renewal. At Turkey Point, non-safety related SSCs relied on for fire protection carry an augmented quality classification (Quality Related) and are included in the FPL Quality Assurance Program.

In addition to the Turkey Point UFSAR, licensing correspondence, and design basis documents, two primary information sources utilized in performing this portion of the

scoping effort were the Turkey Point Safe Shutdown Analysis and the Essential Equipment List.

With regard to the Safe Shutdown Analysis, Section III.G.1 of Appendix R to 10 CFR 50 requires that fire protection features be provided for systems, structures, and components important to safe shutdown. In order to meet these requirements for Turkey Point, equipment required for safe shutdown, including the associated power and control cables, and equipment that could adversely affect safe shutdown if spuriously actuated by fire-induced faults, have been identified for every fire area in the plant in order to assess the fire protection required.

The Essential Equipment List was developed as the first step of the Turkey Point safe shutdown analysis process. This list, which defines the minimum equipment necessary to bring the plant to cold shutdown, contains all power generation and distribution equipment (e.g., diesel generators, batteries, switchgear, motor control centers, power panels) that is required for the operation of the listed equipment. In addition, the list includes equipment that, although not required for safe shutdown, could adversely affect safe shutdown if spuriously actuated by a fire-induced electrical fault. One feature of Turkey Point's Essential Equipment List is that no equipment in storage is credited for safe shutdown.

The steps to identify systems and structures relied upon for Fire Protection at Turkey Point that meet the associated criterion of 10 CFR 54.4(a)(3) are outlined below:

- The UFSAR, Technical Specifications, Essential Equipment List, Safe Shutdown Analysis, licensing correspondence, design basis documents, component database, and design drawings were reviewed, as applicable.
- Based on the above, license renewal intended functions relative to the criterion of 10 CFR 54.4(a)(3) for fire protection were identified for each system and structure determined to meet this criterion.

The scoping process to identify systems and structures relied upon and/or specifically committed to for fire protection for Turkey Point is consistent with and satisfies the associated criterion in 10 CFR 54.4(a)(3).

2.1.1.4.2 ENVIRONMENTAL QUALIFICATION (EQ)

Certain safety-related electrical components are required to withstand environmental conditions that may occur during or following a design basis accident per 10 CFR 50.49. The criteria for determining which equipment requires environmental

qualification are indicated in UFSAR Appendix 8A.3 and are identified on the Turkey Point Environmental Qualification (EQ) List for 10 CFR 50.49. Time-Limited Aging Analyses associated with environmentally qualified equipment are discussed in Subsection 4.4.1.

For non-safety related electrical components whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions, Turkey Point elected not to differentiate between safety-related and non-safety related components. If failure of an electrical component can affect safety-related functions, that electrical component is treated as safety-related for environmental qualification purposes.

The steps to identify systems and structures subject to environmental qualification at Turkey Point that meet the associated criterion of 10 CFR 54.4(a)(3) are outlined below:

- The UFSAR, Technical Specifications, licensing correspondence, Environmental Qualification List, and design basis documents were reviewed, as applicable.
- Based on the above, license renewal intended functions relative to the criterion of 10 CFR 54.4(a)(3) for environmental qualification were identified for each system and structure determined to meet this criterion.

The scoping process to identify systems and structures relied upon and/or specifically committed to for environmental qualification for Turkey Point is consistent with and satisfies the associated criterion in 10 CFR 54.4(a)(3).

2.1.1.4.3 PRESSURIZED THERMAL SHOCK (PTS)

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", requires that licensees evaluate the reactor vessel beltline materials against specific criteria to ensure protection against brittle fracture. See references 2.1-4 through 2.1-7 for a listing of Turkey Point licensing correspondence related to pressurized thermal shock.

The steps to identify systems and structures relied upon for protection against pressurized thermal shock at Turkey Point that meet the associated criterion of 10 CFR 54.4(a)(3) are outlined below:

- The UFSAR, Technical Specifications, licensing correspondence, design basis documents, component database, and design drawings were reviewed, as applicable.
- Based on the above, the only components relied upon for protection against pressurized thermal shock are the reactor vessels. Analyses applicable to pressurized thermal shock have been reevaluated and demonstrated that the reactors vessels meet the screening criteria at the end of the extended period of operation (see Subsection 4.2.1).

The scoping process to identify systems and structures relied upon and/or specifically committed to for pressurized thermal shock for Turkey Point is consistent with and satisfies the associated criterion in 10 CFR 54.4(a)(3).

2.1.1.4.4 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Turkey Point design features related to anticipated transients without scram events are described in detail in UFSAR Section 7.2.4.

The steps to identify systems and structures relied upon for anticipated transients without scram at Turkey Point that meet the associated criterion of 10 CFR 54.4(a)(3) are outlined below:

- The UFSAR, Technical Specifications, licensing correspondence, design basis documents, component database, and design drawings were reviewed, as applicable.
- Based on the above, license renewal intended functions relative to the criterion of 10 CFR 54.4(a)(3) for anticipated transients without scram events were identified for each system and structure determined to meet this criterion.

The scoping process to identify systems and structures relied upon and/or specifically committed to for anticipated transient without scram events for Turkey Point is consistent with and satisfies the associated criterion in 10 CFR 54.4(a)(3).

2.1.1.4.5 STATION BLACKOUT (SBO)

The UFSAR and design basis documents provide the licensing criteria that are the bases for Turkey Point's resolution to station blackout. Design features to satisfy the Station Blackout Rule are described in UFSAR Section 8.2.2.2. Turkey Point

licensing correspondence related to station blackout are listed as references 2.1-8 through 2.1-10.

The steps to identify systems and structures relied upon for station blackout at Turkey Point that meet the associated criterion of 10 CFR 54.4(a)(3) are outlined below:

- The UFSAR, Technical Specifications, licensing correspondence, design basis documents, component database, and design drawings were reviewed, as applicable.
- Based on the above, license renewal intended functions relative to the criterion of 10 CFR 54.4(a)(3) for station blackout were identified for each system and structure determined to meet this criterion.

The scoping process to identify systems and structures relied upon and/or specifically committed to for station blackout for Turkey Point is consistent with and satisfies the associated criterion in 10 CFR 54.4(a)(3).

2.1.2 COMPONENT/STRUCTURAL COMPONENT SCOPING AND SCREENING

This subsection discusses the process used at Turkey Point to: (1) identify components and structural components (collectively abbreviated as SCs) within the scope of license renewal for in-scope systems and structures; and (2) identify which of the SCs determined to be in-scope require an aging management review.

The requirement to identify SCs subject to an aging management review is specified in 10 CFR 54.21(a)(1) that states:

"Each application must contain the following information:

(a) An integrated plant assessment (IPA). The IPA must--

(1) For those systems, structures, and components within the scope of this part, as delineated in §54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components--

(i) That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

(ii) That are not subject to replacement based on a qualified life or specified time period."

This portion of Turkey Point's IPA methodology is divided into three engineering disciplines; mechanical, civil/structural, and electrical/I&C. The relevant aspects of the component/structural component scoping and screening process for mechanical systems, civil structures, and electrical/I&C systems are described in Subsections 2.1.2.1, 2.1.2.2, and 2.1.2.3, respectively.

For mechanical systems and civil structures, this process establishes evaluation boundaries, determines the SCs that compose the system or structure, determines which of those SCs support system/structure intended functions, and identifies specific SC intended functions. Consequently, not all of the SCs for in-scope systems or structures are in the scope of license renewal. Once these in-scope SCs are identified, the process then determines which SCs are subject to an aging management review per the criteria of 10 CFR 54.21(a)(1). Note that screening for Turkey Point is consistent with the NRC Staff's guidance on consumables provided in the NRC's March 10, 2000, letter from Christopher I. Grimes to Douglas J. Walters [Reference 2.1-3].

For electrical/I&C systems, a bounding approach as described in NEI 95-10 [Reference 2.1-1] is taken. This approach establishes evaluation boundaries, determines the electrical and I&C component commodity groups that compose in-scope systems, identifies specific component and commodity intended functions, and then determines which component commodity groups are subject to an aging management review per the criteria of 10 CFR 54.21(a)(1). This approach calls for component scoping after screening has been performed.

2.1.2.1 MECHANICAL SYSTEMS

For mechanical systems, the component/structural component scoping and screening process is performed on each system identified to be within the scope of license renewal. This process evaluates the individual SCs included within in-scope mechanical systems to identify specific SCs or SC groups that require an aging management review.

Mechanical system evaluation boundaries were established for each system within the scope of license renewal. These boundaries were determined by mapping the pressure boundary associated with license renewal system intended functions onto the system flow diagrams. License renewal system intended functions are the functions a system must perform relative to the scoping criteria of 10 CFR 54.4(a)(1), 10 CFR 54.4(a)(2), and 10 CFR 54.4(a)(3). The flow diagram

boundary drawings associated with each mechanical system within the scope of license renewal are identified with the mechanical system screening results described in Section 2.3.

The sequence of steps performed on each mechanical system determined to be within the scope of license renewal is as follows:

- Based on a review of design drawings and the system component list from the component database, SCs that are included within the system are identified.
- Based on the plant level scoping results, the pressure boundary associated with license renewal system intended functions is mapped onto the system's flow diagrams.
- The system SCs that are within the scope of license renewal (i.e., required to perform a license renewal system intended function) are identified.
- Component intended functions for in-scope SCs are identified. The component intended functions identified are based on the guidance of NEI 95-10 [Reference 2.1-1].
- The in-scope SCs that perform an intended function without moving parts or without a change in configuration or properties [screening criterion of 10 CFR 54.21(a)(1)(i)] are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10 [Reference 2.1-1].
- The passive, in-scope SCs that are not subject to replacement based on a qualified life or specified time period [screening criterion of 10 CFR 54.21(a)(1)(ii)] are identified as requiring an aging management review. The determination of whether passive, in-scope SC has a qualified life or specified replacement time period was based on a review of plant-specific information, including the component database, maintenance programs, and procedures.

2.1.2.2 CIVIL STRUCTURES

For structures, the component/structural component scoping and screening process is performed on each structure identified to be within the scope of license renewal. This method evaluates the individual SCs included within in-scope structures to identify specific SCs or SC groups that require an aging management review.

The sequence of steps performed on each structure determined to be within the scope of license renewal is as follows:

- Based on a review of design drawings, the structure component list from the component database, and plant walkdowns, SCs that are included within the structure are identified. These SCs include items such as walls, supports, and non-current carrying electrical and instrumentation and control components, i.e., conduit, cable trays, electrical enclosures, instrument panels, and related supports.
- The SCs that are within the scope of license renewal (i.e., required to perform a license renewal system intended function) are identified.
- Design features and associated SCs that prevent potential seismic interactions for in-scope structures housing both safety-related and non-safety related systems are identified. This includes a walkdown of each plant area containing both safety-related and non-safety related SSCs.
- Component intended functions for in-scope SCs are identified. The component intended functions identified are based on the guidance of NEI 95-10 [Reference 2.1-1].
- The in-scope SCs that perform an intended function without moving parts or without a change in configuration or properties [screening criterion of 10 CFR 54.21(a)(1)(i)] are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10 [Reference 2.1-1].
- The passive, in-scope SCs that are not subject to replacement based on a qualified life or specified time period [screening criterion of 10 CFR 54.21(a)(1)(ii)] are identified as requiring an aging management review. The determination of whether a passive, in-scope SC has a qualified life or specified replacement time period was based on a review of plant-specific information, including the component database, maintenance programs and procedures, vendor manuals, and plant experience.

2.1.2.3 ELECTRICAL AND I&C SYSTEMS

The method used to determine which electrical and I&C components are subject to an aging management review is organized based on component commodity groups. The primary difference in this method versus the one used for mechanical systems and structures is the order in which the component scoping and screening steps are performed. This method was selected for use with the electrical and I&C components since most electrical and I&C components are active. Thus, the method selected provides the most efficient means for determining electrical and I&C components that require an aging management review. The method employed is consistent with the guidance in NEI 95-10 [Reference 2.1-1].

The sequence of steps for identification of electrical and I&C components that require an aging management review is as follows:

- Electrical and I&C component commodity groups associated with electrical, instrumentation and control, and mechanical systems within the scope of license renewal are identified. This step includes a complete review of design drawings and electrical and I&C component commodity groups in the component database.
- A description and function for each of the electrical and I&C component commodity groups are identified.
- The electrical and I&C component commodity groups that perform an intended function without moving parts or without a change in configuration or properties [screening criterion of 10 CFR 54.21(a)(1)(i)] are identified. Active/passive screening determinations are based on the guidance in Appendix B to NEI 95-10 [Reference 2.1-1].
- For the passive electrical and I&C component commodity groups, component commodity groups that are not subject to replacement based on a qualified life or specified time period [screening criterion of 10 CFR 54.21(a)(1)(ii)] are identified as requiring an aging management review. Electrical and I&C component commodity groups covered by the 10 CFR 50.49 Environmental Qualification Program are considered to be subject to replacement based on qualified life.
- Certain passive, long-lived electrical and I&C component commodity groups that do not support license renewal system intended functions are eliminated.

2.1.3 GENERIC SAFETY ISSUES

In accordance with the guidance in NEI 95-10 [Reference 2.1-1] and Appendix A of the draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" [Reference 2.1-11], review of NRC generic safety issues (GSIs) as part of the license renewal process is required to satisfy a finding per 10 CFR 54.29. GSIs that involve issues related to license renewal aging management reviews or time-limited aging analysis evaluations are to be addressed in the License Renewal Application. Based on NEI and NRC guidance, NUREG-0933 [Reference 2.1-12], and previous license renewal applicants, Turkey Point has identified the following GSIs to be addressed:

- GSI 168, Environmental Qualification of Electrical Equipment – This GSI is related to aging concerns with respect to environmental qualification of electrical equipment. Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for Turkey Point Units 3 and 4. Accordingly, this GSI is addressed in Subsection 4.4.2.
- GSI 190, Fatigue Evaluation of Metal Components for 60-year Plant Life – This GSI addresses fatigue life of metal components and was recently closed by the NRC [Reference 2.1-13]. In the closure letter, however, the NRC concluded that licensees should address the effects of reactor coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. Accordingly, the issue of environmental effects on component fatigue life is addressed in Subsection 4.3.5.

2.1.4 CONCLUSION

The methods described in Subsections 2.1.1 and 2.1.2 were used for the Turkey Point Units 3 and 4 IPA to identify the systems, structures, and components that are within the scope of license renewal and require an aging management review. The methods are consistent with and satisfy the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.1.5 REFERENCES

2.1-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 1, Nuclear Energy Institute, January 2000.

2.1-2 C. I. Grimes (NRC) letter to D. J. Walters (NEI), “License Renewal Issue No. 98-0082, Scoping Guidance," August 5, 1999.

2.1-3 C. I. Grimes (NRC) letter to D. J. Walters (NEI), “License Renewal Issue No. 98-12, Consumables," March 10, 2000.

2.1-4 C. O. Woody (FPL) letter to H. G. Thompson (NRC), "Turkey Points Units 3 and 4 -10 CFR 50.61(b)(1) Report," January 23, 1986.

2.1-5 D. G. McDonald (NRC) letter to C. O. Woody (FPL), "Projected Values of Material Properties For Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events - Turkey Point Plant, Units 3 and 4," March 11, 1987.

2.1-6 T. F. Plunkett (FPL) letter to U. S. Nuclear Regulatory Commission, "Turkey Points Units 3 and 4 - 10 CFR 50.61(b)(1) Report," February 13, 1992.

2.1-7 The NRC Safety Evaluation on the amendment to recapture the construction period for Turkey Point, April 20, 1994.

2.1-8 W. F. Conway (FPL) letter to U. S. Nuclear Regulatory Commission, "Information to Resolve Station Blackout," April 17, 1989.

2.1-9 G. E. Edison (NRC) letter to J. H. Goldberg (FPL),"Turkey Point Units 3 and 4 - Safety Evaluation for Proposed Implementation of the Station Blackout Rule (10 CFR 50.63) (TAC Nos. 68618 and 68619)," June 15, 1990.

2.1-10 R. Auluck (NRC) letter to J. H. Goldberg (FPL),"Turkey Point Units 3 and 4 - Supplemental Safety Evaluation for Proposed Implementation of the Station Blackout Rule (10 CFR 50.63) (TAC Nos. 81159 and 81160)," July 31, 1991.

2.1-11 NRC draft, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” April 21, 2000.

2.1-12 NUREG-0933, “A Prioritization of Generic Safety Issues,” Supplement 23, April 1999.

2.1-13 Memorandum, Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to William D. Travers, Executive Director of Operations - Closeout of Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60 Year Plant Life," U. S. Nuclear Regulatory Commission, December 26, 1999.

TABLE 2.1-1
RADIOLOGICAL CONSEQUENCES OF ACCIDENTAL RELEASES

Design Basis Event	Whole Body		Thyroid	
	EB[1]	LPZ[2]	EB[1]	LPZ[2]
10 CFR 100 Limits	25 Rem	25 Rem	300 Rem	300 Rem
Accidental Liquid Release	Negligible	Negligible	Negligible	Negligible
Accidental Gas Release	.064 Rem	.0062 Rem	Negligible	Negligible

NOTES: 1. Exclusion Boundary, 0-2 hours
2. Low Population Zone, 0-2 hours

2.2 PLANT LEVEL SCOPING RESULTS

Turkey Point's Integrated Plant Assessment (IPA) methodology consists of scoping, screening, and aging management reviews. This section provides the plant level scoping results achieved when applying the scoping methodology described in Subsection 2.1.1 to plant systems and structures. Tables 2.2-1, 2.2-2, and 2.2-3 provide the plant level scoping results for mechanical systems, structures, and electrical/I&C systems, respectively. If a system or structure, in whole or in part, meets one or more of the license renewal scoping criteria, the system or structure is considered to be within the scope of license renewal. Also included in the tables are references to the sections in the application that discuss screening results for in-scope systems and structures.

Figure 2.2-1 provides a layout of Turkey Point Units 3 and 4 and identifies the structures within the scope of license renewal in bold. Figure 2.2-2 provides a layout of the structural components included in the structure identified as "Yard Structures."

TABLE 2.2-1
LICENSE RENEWAL SCOPING RESULTS FOR MECHANICAL SYSTEMS

System Name	System in License Renewal Scope?	Screening Results Application Subsection
Amertap	No	
Auxiliary Building Ventilation	Yes	2.3.3.10
Auxiliary Feedwater and Condensate Storage	Yes	2.3.4.3
Auxiliary Steam	No	
Circulating Cooling Water	No	
Component Cooling Water	Yes	2.3.3.2
Condensate	No	
Condensate Polishing	No	
Condensate Recovery	No	
Condenser	No	
Containment Isolation	Yes	2.3.2.3
Containment Post-Accident Monitoring and Control	Yes	2.3.2.7
Containment Spray	Yes	2.3.2.2
Control Building Ventilation	Yes	2.3.3.11
Chemical and Volume Control	Yes	2.3.3.4
Electrical Equipment Room Ventilation	Yes	2.3.3.10
Emergency Containment Cooling	Yes	2.3.2.1
Emergency Containment Filtration	Yes	2.3.2.6
Emergency Diesel Generator and Support Systems	Yes	2.3.3.15
Emergency Diesel Generator Building Ventilation	Yes	2.3.3.12
Environmental Monitoring	No	
Extraction Steam	No	
Feedwater and Blowdown	Yes	2.3.4.2
Feedwater Heaters, Drains, and Vents	No	
Fire Protection	Yes	2.3.3.14
Gland Steam and Drains	No	
Instrument Air	Yes	2.3.3.8
Intake Cooling Water	Yes	2.3.3.1
Main Steam and Turbine Generators	Yes	2.3.4.1
Metal Impact Monitoring	No	

TABLE 2.2-1 (continued)
LICENSE RENEWAL SCOPING RESULTS FOR MECHANICAL SYSTEMS

System Name	System in License Renewal Scope?	Screening Results Application Subsection
New Fuel Storage Area Ventilation	No	
Normal Containment and Control Rod Drive Mechanism Cooling	Yes	2.3.3.9
Penetration Cooling	No	
Primary Water Makeup	Yes	2.3.3.5
Radwaste Building Ventilation	No	
Reactor Coolant	Yes	2.3.1
Residual Heat Removal	Yes	2.3.2.5
Safety Injection	Yes	2.3.2.4
Sample System - NSSS and Secondary	Yes	2.3.3.6
Screen Wash and Chlorination	No	
Secondary Wet Layup	No	
Security	No	
Service (City) Water	No	
Spent Fuel Pool Cooling	Yes	2.3.3.3
Spent Fuel Storage Area Ventilation	No	
Steam Generator Wet Layup	No	
Turbine Building Ventilation	Yes	2.3.3.13
Turbine Lube Oil	No	
Turbine Plant Chemical Addition	No	
Turbine Plant Cooling Water	No	
Waste Disposal	Yes	2.3.3.7
Water Treatment Plant	No	

TABLE 2.2-2
LICENSE RENEWAL SCOPING RESULTS FOR STRUCTURES

Structure Name	Structure in License Renewal Scope?	Screening Results Application Subsection
Access Dress Facility	No	
Auxiliary Building (includes Fuel Handling Building and New Electrical Equipment Room)	Yes	2.4.2.1
"C" Bus Electrical Switchgear Enclosures	No	
Cafeteria	No	
Chemical Storage Building	No	
Cold Chemistry Lab	Yes	2.4.2.2
Containments	Yes	2.4.1
Control Building	Yes	2.4.2.3
Cooling Water Canals	Yes	2.4.2.4
Diesel Driven Fire Pump Enclosure	Yes	2.4.2.5
Discharge Structure	Yes	2.4.2.6
Dry Storage Warehouse	No	
Electrical Penetration Rooms	Yes	2.4.2.7
Emergency Diesel Generator Buildings	Yes	2.4.2.8
Fire Protection Monitoring Station	Yes	2.4.2.9
Fire Rated Assemblies	Yes	2.4.2.10
Hazardous Materials Storage Facility	No	
Health Physics Control Building	No	
Health Physics Truck Monitoring Building	No	
I&C Repair Facility	No	
Intake Structure	Yes	2.4.2.11
Machine Shop	No	
Main Steam and Feedwater Platforms	Yes	2.4.2.12
Main Truck Gate House	No	
Meteorological Towers	No	
New Fuel Storage and Handling	No	
Nuclear Administration Building	No	
Nuclear Administration Building Vault	No	
Nuclear Entrance Building	No	
Nuclear Maintenance Building	No	

TABLE 2.2-2 (continued)
LICENSE RENEWAL SCOPING RESULTS FOR STRUCTURES

Structure Name	Structure in License Renewal Scope?	Screening Results Application Subsection
Offsite Communications Tower	No	
Operator Radiation Controlled Area (RCA) Access Station	No	
Other Miscellaneous Buildings	No	
Plant Vent Stack	Yes	2.4.2.13
Polar Cranes	Yes	2.4.1
Radwaste Building	No[1]	
Satellite Security Stations	No	
Self-Contained Breathing Apparatus (SCBA) Facility	No	
Security Barriers	No	
Spare Main Transformer	No	
Spent Fuel Storage and Handling	Yes	2.4.1 & 2.4.2.14
Steam Generator Storage Facility	No	
Switchyard Relay Enclosure	No	
Technical Support Center	No	
Turbine Building	Yes	2.4.2.15
Turbine Gantry Cranes	Yes	2.4.2.16
Turkey Point Units 1 and 2 Chimneys	Yes	2.4.2.17
Warehouse	No	
Water Treatment Plant	No	
Yard Structures (includes equipment foundations, concrete footings for structural steel supports, pipe trenches, and duct banks)	Yes	2.4.2.18

NOTE: 1. UFSAR Section 5.3-3 and Appendix 5A classify the Radwaste Building as Seismic Class 1. Considering the Radwaste Building does not house or protect safety-related SSCs and that the radiological consequences of accidental releases from postulated failures are a small fraction of 10 CFR 100 limits (see Table 2.1-1), the building does not meet the criteria of 10 CFR 54.4.

TABLE 2.2-3
LICENSE RENEWAL SCOPING RESULTS FOR ELECTRICAL/I&C SYSTEMS

System Name	System in License Renewal Scope?	Screening Results Application Section
125 VDC and 120 VAC	Yes	2.5
240 kV Switchyard	No	
4.16 kV	Yes	2.5
480 V Switchgear and Motor Control Centers	Yes	2.5
Annunciators	No	
Area Radiation Monitoring	No	
ATWS Mitigating System Actuation Circuitry (AMSAC)	Yes	2.5
Communications	Yes	2.5
Containment Electrical Penetrations (conductor and non-metallic portions)	Yes	2.5
Emergency Load Sequencer	Yes	2.5
Emergency Response Facility and Plant Computer	Yes	2.5
Engineering Safeguards	Yes	2.5
Fire and Smoke Detection	Yes	2.5
Lightning Protection	Yes	2.5
Main and Auxiliary Transformers	No	
Nuclear Instrumentation (Incore and Excore)	Yes	2.5
Plant Lighting	Yes	2.5
Process Radiation Monitoring	Yes	2.5
Qualified Safety Parameter Display System (QSPDS)	Yes	2.5
Reactivity Computer	No	
Reactor Protection	Yes	2.5
Start-Up Transformers	No	
Underwater TV Camera	No	

FIGURE 2.2-1
TURKEY POINT PLANT STRUCTURES

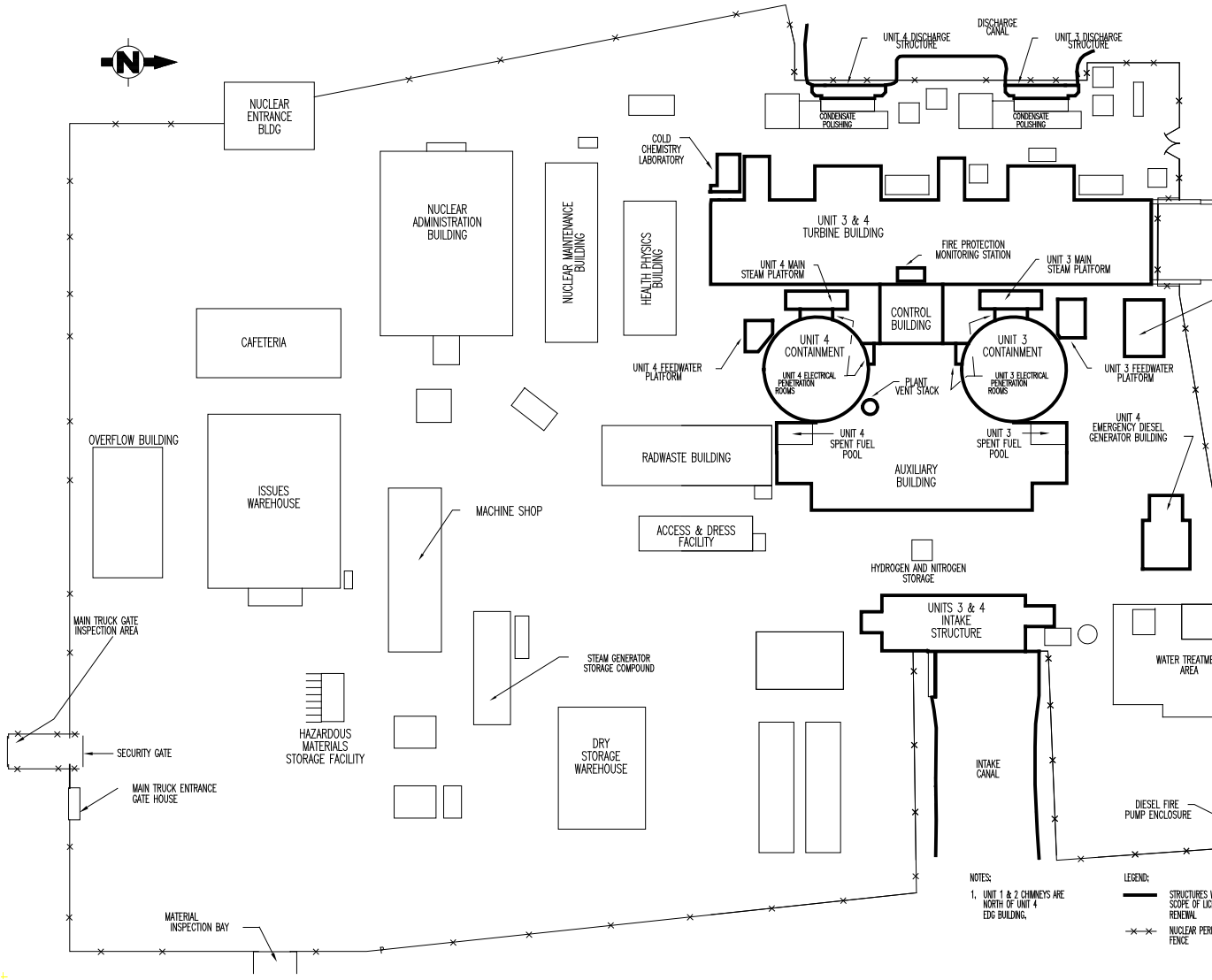
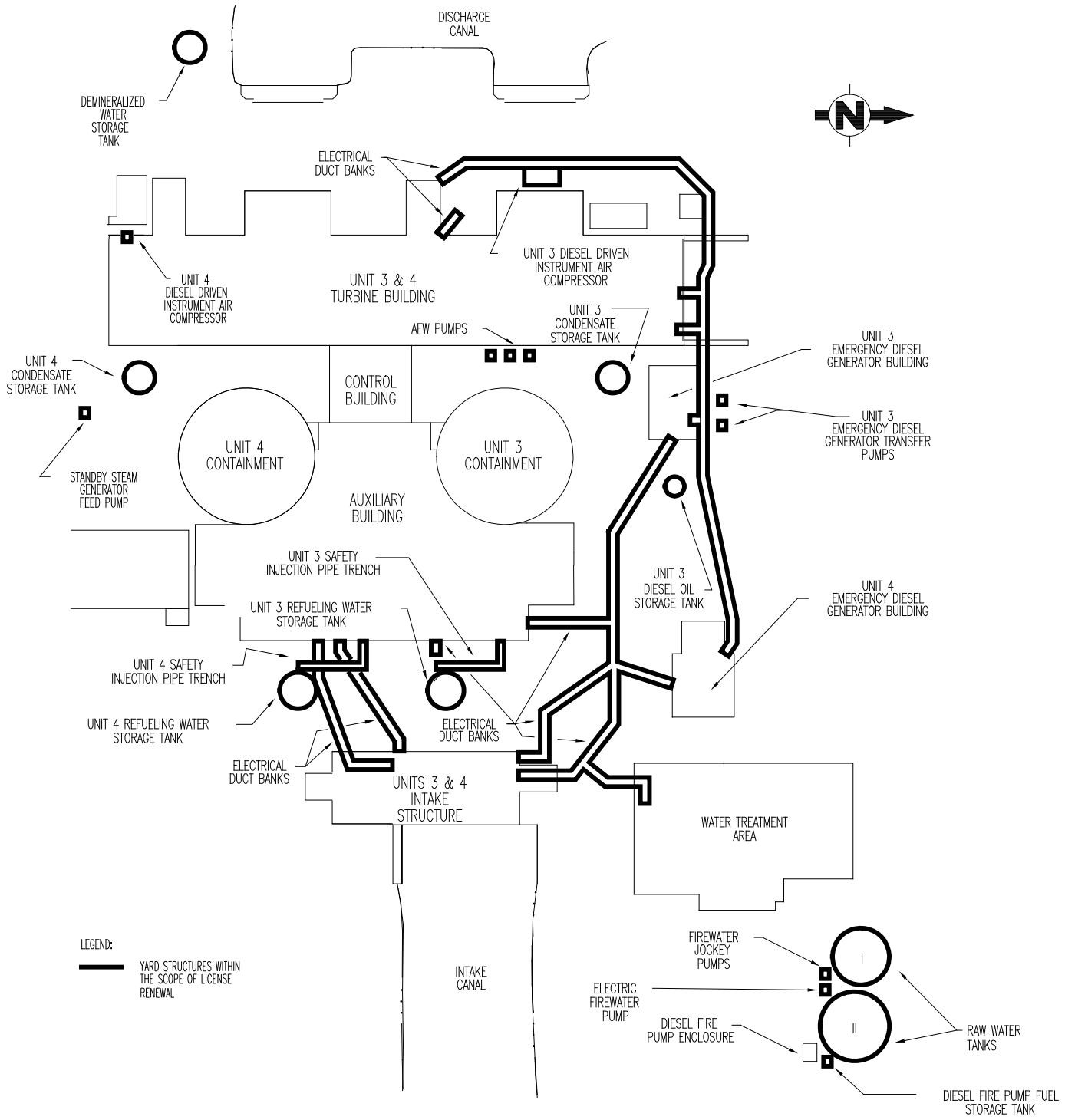


FIGURE 2.2-2
TURKEY POINT YARD STRUCTURES



2.3 SYSTEM SCOPING AND SCREENING RESULTS – MECHANICAL SYSTEMS

The determination of mechanical systems within the scope of license renewal is made by initially identifying Turkey Point mechanical systems and then reviewing them to determine which ones satisfy one or more of the criteria contained in 10 CFR 54.4. This process is described in Section 2.1 and the results of the mechanical systems review are contained in Section 2.2.

Section 2.1 also provides the methodology for determining the components within the scope of 10 CFR 54.4 that meet the requirements contained in 10 CFR 54.21(a)(1). The components that meet these screening requirements are identified in this section. These identified components subsequently require an aging management review for license renewal.

The screening results are provided below in four subsections:

- Reactor Coolant Systems
- Engineered Safety Features Systems
- Auxiliary Systems
- Steam and Power Conversion Systems.

2.3.1 REACTOR COOLANT SYSTEMS

The Reactor Coolant Systems consist of the systems and components designed to contain and support the nuclear fuel, contain the reactor coolant, and transfer the heat produced in the reactor to the steam and power conversion systems for the production of electricity.

Unless noted otherwise, the Reactor Coolant Systems for Turkey Point Units 3 and 4 are the same, with no components common to both units. The Reactor Coolant Systems are described in UFSAR Chapters 3 and 4. The following components are included in this subsection:

- Reactor Coolant Piping
- Regenerative and Excess Letdown Heat Exchangers
- Pressurizers
- Reactor Vessels
- Reactor Vessel Internals
- Reactor Coolant Pumps
- Steam Generators

The license renewal flow diagrams listed in Table 2.3-1 show the evaluation boundaries for the portions of Reactor Coolant Systems that are within the scope of license renewal.

Reactor Coolant System components subject to aging management review include the reactor vessel and control rod drive mechanism pressure boundary, pressurizers, steam generators, reactor vessel internals, reactor coolant pumps (pressure boundary only), and reactor coolant piping, valves (pressure boundary only), and fittings. The regenerative and excess letdown heat exchangers that are part of the Chemical and Volume Control System are also addressed in this subsection because they form a part of the Reactor Coolant System pressure boundary.

Class 1 as used in this application means the Safety Class 1 definition per American Nuclear Society (ANS) Standard N46.2.

The design code for reactor coolant piping is the 1955 Edition of American National Standards Institute (ANSI) B31.1 with the exception of the pressurizer surge lines that were analyzed to the 1986 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. Class 1 piping starts at, and includes, the circumferential welds joining the piping to the Class 1 components and typically ends at the second normally closed valve from the Reactor Coolant System or the 3/8-inch flow restrictor in the piping.

The regenerative heat exchangers were designed and fabricated in accordance with the requirements of Tubular Exchanger Manufacturers Association (TEMA) Class R and the ASME Boiler and Pressure Vessel Code, Section III, Class C. The excess letdown heat exchangers were designed and fabricated in accordance with the requirements of TEMA Class R, the ASME Boiler and Pressure Vessel Code, Section III, Class C (tube side), and the ASME Boiler and Pressure Vessel Code, Section VIII (shell side).

The pressurizers were designed and fabricated in accordance with the requirements of the 1965 Edition of the ASME Boiler and Pressure Vessel Code.

The reactor vessels were manufactured by Babcock & Wilcox Co. in accordance with the design and fabrication requirements of the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III, through the Summer 1966 Addenda.

The reactor vessel internals were designed prior to the creation of ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, using internal Westinghouse design criteria that effectively evolved to become the original NG criteria. The reactor vessel internals were designed using the allowable stress levels of the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Article 4, through the Summer 1966 Addenda.

The reactor coolant pump casings, main flanges, and main flange bolts were analyzed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 4.

The original steam generator components were designed and analyzed to the 1965 Edition of the ASME Boiler and Pressure Vessel Code, through Summer 1965 Addenda. The replacement steam generator components were constructed in accordance with the 1974 Edition of the ASME Boiler and Pressure Vessel Code, through Summer 1976 Addenda.

2.3.1.1 WESTINGHOUSE OWNERS GROUP GENERIC TECHNICAL REPORTS

Turkey Point actively participated in a Westinghouse Owners Group effort that developed a series of generic technical reports whose purpose was to demonstrate that the aging effects for Reactor Coolant System components are adequately managed for the period of extended operation. The following generic technical reports, applicable to Westinghouse Reactor Coolant Systems, have been submitted to the NRC for approval by Westinghouse:

- WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components" [References 2.3-1 through 2.3-3]. Draft NRC Safety Evaluation dated February 10, 2000 [Reference 2.3-4] has been issued.
- WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers" [References 2.3-3, 2.3-5, and 2.3-6]. Draft NRC Safety Evaluation dated August 7, 2000 [Reference 2.3-7] has been issued.
- WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals" [References 2.3-8 and 2.3-9]. Draft NRC Safety Evaluation has not been issued.
- WCAP-14422, "License Renewal Evaluation: Aging Management for Reactor Coolant System Supports" [Reference 2.3-10]. Draft NRC Safety Evaluation dated February 25, 2000 [Reference 2.3-11] has been issued. Note that the Reactor Coolant System supports are discussed in Section 2.4.1.

NRC-approved generic technical reports may be incorporated by reference in the Application pursuant to 10 CFR 54.17(e) provided the conditions of approval contained in the safety evaluation of the specific report are met. These reports are not incorporated by reference in the Turkey Point License Renewal Application because, as of September 1, 2000, none has received a final safety evaluation. However, to facilitate NRC review of these particular components, this Application addresses the applicability of these reports to the associated components at Turkey Point.

2.3.1.1.1 PROCESS FOR ESTABLISHING WESTINGHOUSE GENERIC TECHNICAL REPORT APPLICABILITY TO TURKEY POINT

Turkey Point used the following process to establish Westinghouse generic technical report applicability to the components.

1) Comparison of the component intended functions for the Reactor Coolant System components under review - The Turkey Point-specific component screening review first identifies the component intended functions and then compares these functions to those identified in the generic technical reports. Differences are noted and justification for the variances provided.

2) Identification of the items that are subject to aging management review - Turkey Point drawings and pertinent design and field change data are reviewed. The process establishes the full extent to which plant identified scope matches the scope identified in the generic technical reports. For those components that require an aging management review, a comparison of the component material and environment is considered in determining the extent to which the plant scope is bounded by the generic technical report. Areas not bounded are noted and evaluated.

3) Identification of the applicable aging effects - An independent assessment of the applicable aging effects is performed by reviewing plant operating environment, operating stresses, and plant-specific operating experience. This assessment reveals potential aging effects not identified in the generic technical reports. Aging effects for items that are determined to be subject to aging management review, that were not identified in the generic technical reports, are evaluated.

4) Review of Open Items and Applicant Action Items - In order to facilitate NRC review, open items and applicant action items are addressed if available prior to August 1, 2000.

Note that items (1), (2), and (4) are addressed in Sections 2.3.1 and 2.4.1. Item (3) is addressed in Sections 3.2 and 3.6.

2.3.1.2 REACTOR COOLANT PIPING

Reactor coolant piping consists of piping (including fittings, branch connections, safe ends, thermal sleeves, flow restrictors, and thermowells), pressure retaining parts of valves, and bolted closures and connections. Reactor coolant piping is presented in two parts:

- Class 1 piping
- Non-Class 1 piping.

2.3.1.2.1 CLASS 1 PIPING

Class 1 piping includes the main coolant piping; pressurizer surge, spray, safety, and relief lines; vents, drains, instrumentation lines; and Class 1 portions of ancillary systems attached to the Reactor Coolant System. Ancillary systems attached to the Reactor Coolant System include Residual Heat Removal, Safety Injection, Nuclear Steam Supply System Sampling, and Chemical and Volume Control. Reactor coolant piping is described in UFSAR Section 4.2.2.

The NRC issued a draft safety evaluation [Reference 2.3-4] on Westinghouse Owners Group generic technical report WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components" [References 2.3-1 through 2.3-3], on February 10, 2000.

Turkey Point reviewed the current design and operation of the reactor coolant piping using the process described in Subsection 2.3.1.1.1 and confirmed that the Turkey Point Class 1 piping is bounded by the description of Class 1 piping contained in WCAP-14575 with regard to design criteria and features, materials of construction, fabrication techniques, installed configuration, modes of operation, and environments/exposures. The component intended functions for Class 1 piping are inclusive of the intended functions identified in WCAP-14575. In addition to the functions identified in WCAP-14575, Turkey Point has identified an additional function for flow-restricting orifices and reducers. These orifices and reducers provide throttling to limit the maximum flow through a postulated break in an attached non-Class I line to a value within the makeup capability of the Chemical and Volume Control System.

As a result of the NRC review of WCAP-14575, several open items and applicant action items were identified and documented in the NRC draft safety evaluation.

The Turkey Point-specific responses to those open items and applicant action items relevant to the identification of reactor coolant piping components subject to aging management review are provided in Tables 2.3-2 and 2.3-3.

2.3.1.2.2 NON-CLASS 1 PIPING

Non-Class 1 piping is not within the scope of WCAP-14575. However, several non-Class 1 components are within the scope of license renewal. The component intended function of these in-scope non-Class 1 components is pressure boundary integrity. The non-Class 1 reactor coolant components requiring an aging management review include:

- Instrumentation tubing and fittings downstream of flow restrictors
- Inner reactor vessel flange O-ring leak detection line tubing, fittings and valves (pressure boundary only)
- Reactor vessel head vent piping, fittings, and valves (pressure boundary only) downstream of the restricting orifices
- Instrument air/nitrogen supply piping, tubing, fittings, accumulators, and valves (pressure boundary only) to the power operated relief valves
- Reactor coolant pump motor upper bearing oil heat exchanger and lower bearing oil cooling coil (the heat exchanger and cooling coil form a portion of the Component Cooling Water pressure boundary)

2.3.1.3 REGENERATIVE AND EXCESS LETDOWN HEAT EXCHANGERS

The regenerative and excess letdown heat exchangers are a part of Chemical and Volume Control. They are addressed in this subsection, however, because they are within the Reactor Coolant System pressure boundary. The regenerative and excess letdown heat exchangers are described in UFSAR Section 9.2.

The regenerative heat exchangers are of a multiple shell and U-tube design, each consisting of three heat exchangers interconnected in series by piping and mounted on a common support frame. The heat exchangers are designed to recover heat from the letdown stream by heating the charging stream, thus minimizing reactivity effects due to injection of cold water and minimizing thermal stress on the charging line penetrations in the reactor coolant loop piping. The letdown stream flows through the shell of the heat exchangers, and the charging stream flows through the tubes.

The excess letdown heat exchangers are of the U-tube design. Their function is to cool reactor coolant letdown flow equivalent to that portion of the nominal seal injection flow that enters the Reactor Coolant System through the labyrinth of the reactor coolant pump seals. They may be used when the normal letdown path is temporarily out of service or for supplementing the maximum letdown during heatup. The letdown is a four-pass flow through the tubes, while Component Cooling Water System flow is a single pass through the shells.

The component intended functions of the regenerative and excess letdown heat exchangers are pressure boundary integrity and heat transfer.

2.3.1.4 PRESSURIZERS

The pressurizers are vertical cylindrical vessels containing electric heaters in the lower heads and water spray nozzles in the upper heads. Since sources of heat in the Reactor Coolant Systems are interconnected by piping with no intervening isolation valves, relief protection for the Reactor Coolant Systems is provided on the pressurizers. Overpressure protection consists of three code safety valves and two power operated relief valves on each pressurizer. Piping attached to the pressurizer is Class 1 up to and including the second isolation valve (with the exception of the pressurizer code safety valves) and is discussed in Subsection 2.3.1.2. The pressurizers are described in UFSAR Section 4.2.2.

A draft safety evaluation for Westinghouse Owners Group generic technical report WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers" [References 2.3-3, 2.3-5, and 2.3-6], was issued on August 7, 2000 [Reference 2.3-7]. Turkey Point reviewed the current design and operation of the pressurizers using the process described in Subsection 2.3.1.1.1 and has confirmed that the Turkey Point pressurizers are bounded by the description contained in WCAP-14574. The component intended functions for the pressurizers are consistent with the intended functions identified in WCAP-14574.

2.3.1.5 REACTOR VESSELS

The reactor vessels consist of cylindrical vessel shells, lower vessel heads, closure heads, nozzles, interior attachments, and associated pressure-retaining bolting. The vessels are fabricated of low alloy steel with austenitic stainless steel cladding on internal surfaces exposed to the reactor coolant fluid. Coolant flow for each reactor vessel enters through three inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward, through the annular space between

the vessel wall and the core barrel into a plenum at the bottom of the vessel, where it reverses direction, passes up through the core into the upper plenum, and then flows out of the vessel through three exit nozzles located on the same plane as the inlet nozzles. The component intended functions of the reactor vessels include pressure boundary integrity and structural support. The reactor vessels are described in UFSAR Chapter 3.

Control rod drive mechanism housings are attached to flanged nozzles, which penetrate the closure heads. The active portions of the control rod drive mechanisms do not require an aging management review per 10 CFR 54.21(a)(1)(i). The part-length control rod drive mechanisms, although they remain installed, are not being used at Turkey Point. Note that two of the part-length control rod drive mechanism housings on each reactor vessel have been modified for the installation of the Reactor Vessel Level Indication System. The control rod drive mechanism housings are threaded and seal welded to the reactor vessel head penetrations. The component intended function of the control rod drive mechanism housings is pressure boundary integrity. The control rod drive mechanisms are described in UFSAR Sections 3.1.3 and 3.2.3.

Bottom mounted instrumentation penetrates the reactor vessel lower head domes. The fifty (50) bottom head instrumentation tubes and attached bottom mounted guide tubes, flux thimble tubes, and seal table for each reactor vessel provide the capability of monitoring core flux distribution. The component intended function of the bottom mounted instrumentation is pressure boundary integrity. The bottom mounted instrumentation is described in UFSAR Section 3.2.3.

2.3.1.6 REACTOR VESSEL INTERNALS

The reactor vessel internals are designed to support, align, and guide the core components, and to support and guide incore instrumentation. The reactor vessel internals consist of two basic assemblies for each reactor vessel: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core; and a lower internals assembly that can be removed, if desired, following a complete core unload. The reactor vessel internals are described in UFSAR Chapter 3.

Each lower internals assembly is supported in the vessel by resting on a ledge below the vessel-head mating surface and is closely guided at the bottom by radial support/clevis assemblies. Each upper internals assembly is clamped at this same

ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The lower internals comprise the core barrel, thermal shield, core baffle assembly, lower core plate, intermediate diffuser plate, bottom support casting, and supporting structures. The upper internals assembly (upper core support structure) is a rigid member composed of the top support plate and deep beam section, support columns, control rod guide tube assemblies, and the upper core plate. Upon upper internals assembly installation, the last three parts are physically located inside the core barrel.

The component intended functions of the reactor vessel internals are core support, coolant distribution, guidance and support of instrumentation and control rods, and vessel shielding.

A draft safety evaluation for Westinghouse Owners Group generic technical report WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," [References 2.3-8 and 2.3-9], has not been issued. Turkey Point reviewed the current design and operation of the reactor vessel internals using the process described in Subsection 2.3.1.1.1 and has confirmed that the Turkey Point reactor vessel internals are bounded by the description contained in WCAP-14577. The component intended functions for the reactor vessel internals are consistent with the intended functions identified in WCAP-14577.

2.3.1.7 REACTOR COOLANT PUMPS

Each of the three reactor coolant loops for Turkey Point Units 3 and 4 contains a vertically mounted, single stage centrifugal reactor coolant pump that employs a controlled leakage seal assembly. The reactor coolant pumps provide the motive force for circulating the reactor coolant through the reactor core, piping, and steam generators. The reactor coolant pumps used at Turkey Point are Westinghouse Model 93. The component intended function of the reactor coolant pumps is pressure boundary integrity. The components that support this function include the casing, cover, pressure-retaining bolting, and integral thermal barrier heat exchanger. Non-Class 1 piping, instrumentation, and other components attached to the reactor coolant pumps are addressed in Subsection 2.3.1.2.2. The reactor coolant pump seals are not subject to an aging management review for the following reasons:

- Seal leakoff is closely monitored in the control room, and a high leakoff flow is alarmed as an abnormal condition requiring corrective action.
- The reactor coolant pump seal package and its constituent parts are routinely inspected and parts replaced, as required based on condition, for each reactor coolant pump.
- Plant operating experience has demonstrated the effectiveness of these activities.

Class 1 reactor coolant piping connected to the pumps, including the welded joints, is discussed in Subsection 2.3.1.2.1. The portions of the reactor coolant pump rotating elements above the pump coupling, including the electric motor and the flywheel, are not subject to aging management review in accordance with 10 CFR 54.21(a)(1)(i). The reactor coolant pumps are described in UFSAR Section 4.2.2.

The reactor coolant pumps are within the scope of WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components" [Reference 2.3-1 through 2.3-3]. The NRC draft safety evaluation [Reference 2.3-4] for WCAP-14575 was issued on February 10, 2000.

Turkey Point reviewed the current design and operation of the reactor coolant pumps using the process described in Subsection 2.3.1.1.1 and confirmed that the reactor coolant pumps are bounded by the description contained in WCAP-14575 with regard to design criteria and features, materials of construction, fabrication techniques, installed configuration, modes of operation, and environments/exposures. The component intended function for the reactor coolant pumps is also consistent with the intended function identified in WCAP-14575.

As a result of the NRC review of WCAP-14575, several open items and applicant action items were identified and documented in the NRC draft safety evaluation. The Turkey Point-specific responses to those open items and applicant action items relevant to the identification of reactor coolant pump components subject to aging management review are provided in Tables 2.3-2 and 2.3-3.

2.3.1.8 STEAM GENERATORS

There are three steam generators installed in each unit. One steam generator is installed in each reactor coolant loop. Each steam generator is a vertical shell and

tube heat exchanger, which transfers heat from a single-phase fluid at high temperature and pressure (the reactor coolant) in the tube side, to a two-phase (steam-water) mixture at lower temperature and pressure in the shell side. The steam generators are described in the UFSAR Section 4.2.2.

The reactor coolant enters and exits the tube side of each steam generator through nozzles located in the lower hemispherical head. The Reactor Coolant System fluid flows through inverted U-tubes connected to the tube sheet. The lower head is divided into inlet and outlet chambers by a vertical partition plate extending from the lower head to the tube sheet. The steam-water mixture is generated on the secondary, or shell side, and flows upward through moisture separators and dryers to the outlet nozzle at the top of the vessel, providing essentially dry, saturated steam. Manways are provided to permit access to both sides of the lower head and to the U-tubes and moisture separating equipment on the shell side of the steam generators.

The component intended functions of the steam generators include pressure boundary integrity, heat transfer, flow distribution, structural support, and throttling.

2.3.1.9 SUMMARY

The Reactor Coolant Systems are in the scope of license renewal because they contain:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are a part of the Environmental Qualification Program
- SCs that are relied on during certain postulated fire, station blackout, pressurized thermal shock, and anticipated transients without scram events

The Reactor Coolant System components subject to an aging management review and the component intended functions are provided in Table 3.2-1. The aging management review for the Reactor Coolant Systems is discussed in Section 3.2.

2.3.2 ENGINEERED SAFETY FEATURES SYSTEMS

Engineered Safety Features Systems consist of systems and components designed to function under accident conditions to minimize the severity of an accident or to mitigate the consequences of an accident. In the event of a loss-of-coolant accident, the Engineered Safety Features Systems provide emergency coolant to assure structural integrity of the core, to maintain the integrity of the containment, and to reduce the concentration of fission products expelled to the containment building atmosphere. Unless noted otherwise, the Engineered Safety Features Systems for Turkey Point Units 3 and 4 are the same.

The following systems are included in this subsection:

- Emergency Containment Cooling
- Containment Spray
- Containment Isolation
- Safety Injection
- Residual Heat Removal
- Emergency Containment Filtration
- Containment Post Accident Monitoring and Control

2.3.2.1 EMERGENCY CONTAINMENT COOLING

Emergency Containment Cooling is designed to remove sufficient heat to maintain the containment below its structural design pressure and temperature during a loss-of-coolant accident or main steamline break. In addition, the emergency fan cooling units continue to remove heat after the maximum hypothetical accident and reduce containment pressure to atmospheric. Heat removed from the containment is transferred to Component Cooling Water. Emergency Containment Cooling consists of three fan cooling units that are located above the refueling floor, around the inside of each containment. Emergency Containment Cooling is described in UFSAR Section 6.3.

The flow diagrams listed in Table 2.3-4 show the evaluation boundaries for the portions of Emergency Containment Cooling that are within the scope of license renewal.

Emergency Containment Cooling is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are a part of the Environmental Qualification Program

Emergency Containment Cooling components subject to an aging management review include the emergency fan cooler units (pressure boundary only) and associated heat exchanger coils. The intended functions for Emergency Containment Cooling components subject to an aging management review include pressure boundary integrity and heat transfer. A complete list of Emergency Containment Cooling components requiring an aging management review and the component intended functions are provided in Table 3.3-1. The aging management review for Emergency Containment Cooling is discussed in Section 3.3.

2.3.2.2 CONTAINMENT SPRAY

Containment Spray is designed to remove sufficient heat to maintain the containment below its design pressure and temperature during a loss-of-coolant accident or main steam line break. Containment Spray is composed of two motor-driven horizontal centrifugal pumps, each discharging to two spray lateral headers located near the top of the containment structure. The system also utilizes the residual heat removal pumps and heat exchangers for the long-term recirculation phase of containment spray, as described in Subsection 2.3.2.5. Additionally, Containment Spray provides a source of water for Emergency Containment Filtration spray (see Subsection 2.3.2.6). Components associated with this function are included in the scope of Emergency Containment Filtration. Containment Spray is described in UFSAR Section 6.4.

The flow diagrams listed in Table 2.3-4 show the evaluation boundaries for the portions of Containment Spray that are within the scope of license renewal.

Containment Spray is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are a part of the Environmental Qualification Program

Containment Spray components subject to an aging management review include the pumps and valves (pressure boundary only), heat exchangers, cyclone separators, piping, tubing, fittings, orifices, and spray nozzles. The intended functions for Containment Spray components subject to an aging management review include pressure boundary integrity, spray, throttling, filtration, and heat transfer. A complete list of Containment Spray components requiring an aging management review and the component intended functions are provided in Table 3.3-2. The aging management review for Containment Spray is discussed in Section 3.3.

2.3.2.3 CONTAINMENT ISOLATION

Containment Isolation is an engineered safety feature that provides for the closure or integrity of containment penetrations to prevent leakage of uncontrolled or unmonitored radioactive materials to the environment. Containment Isolation is described in UFSAR Section 6.6.

Process systems that have license renewal system intended functions in addition to the containment isolation function are included in the system screening results described elsewhere in Section 2.3.

The pressure boundary (metallic) portions of electrical penetrations and miscellaneous/spare mechanical penetrations that are not associated with a process system are included in the civil/structural screening described in Section 2.4.

The non-metallic and conductor portions of containment electrical penetrations are included in the electrical screening described in Section 2.5.

Note that all containment penetrations and associated containment isolation valves and components that ensure containment integrity, regardless of where they are described, require an aging management review.

Breathing Air, Nitrogen and Hydrogen, and Containment Purge are the process systems whose only license renewal intended function is containment isolation. The flow diagrams listed in Table 2.3-4 show the evaluation boundaries for the portions of Breathing Air, Nitrogen and Hydrogen, and Containment Purge that are within the scope of license renewal.

Breathing Air, Nitrogen and Hydrogen, and Containment Purge are in the scope of license renewal because they contain:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are a part of the Environmental Qualification Program
- SCs that are relied on during station blackout events

Breathing Air, Nitrogen and Hydrogen, and Containment Purge components within the scope of license renewal and subject to aging management review include valves (pressure boundary only), piping, tubing, fittings, and debris screens (Containment Purge). The intended functions for Breathing Air, Nitrogen and Hydrogen, and Containment Purge components subject to an aging management review include pressure boundary integrity and filtration. Breathing Air, Nitrogen and Hydrogen, and Containment Purge components requiring an aging management review and the component intended functions are listed in Table 3.3-3. The aging management review for Containment Isolation is discussed in Section 3.3.

2.3.2.4 SAFETY INJECTION

Safety Injection, which includes the safety injection accumulators, provides emergency core cooling and reactivity control during and following design basis accidents. Safety Injection is described in UFSAR Section 6.2.

The flow diagrams listed in Table 2.3-4 show the evaluation boundaries for the portions of Safety Injection that are within the scope of license renewal. Insulation is not within the scope of license renewal for Safety Injection because the systems do not contain boric acid solutions at concentrations that require heat tracing, tank heaters, and/or insulation to prevent precipitation.

Safety Injection is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are a part of the Environmental Qualification Program

- SCs that are relied on during certain postulated fire and station blackout events

Safety Injection components subject to an aging management review include the refueling water storage tanks, accumulators, pumps and valves (pressure boundary only), heat exchanger tubes, orifices, piping, tubing, and fittings. The intended functions for Safety Injection components subject to an aging management review include pressure boundary integrity, heat transfer, and throttling. A complete list of Safety Injection components requiring an aging management review and the component intended functions are provided in Table 3.3-4. The aging management review for Safety Injection is discussed in Section 3.3.

2.3.2.5 RESIDUAL HEAT REMOVAL

Residual Heat Removal delivers borated water to the Reactor Coolant Systems during the injection phase of a design basis accident. Following a loss-of-coolant accident, Residual Heat Removal cools and recirculates water that is collected in the containment recirculation sumps and returns it to the Reactor Coolant, Containment Spray, and Safety Injection Systems to maintain reactor core and containment cooling functions. In addition, during normal plant operations, Residual Heat Removal removes residual and sensible heat from the core during plant shutdown, cooldown, and refueling operations. Residual Heat Removal is described in UFSAR Section 6.2.

The flow diagrams listed in Table 2.3-4 show the evaluation boundaries for the portions of Residual Heat Removal that are within the scope of license renewal.

Residual Heat Removal is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are a part of the Environmental Qualification Program
- SCs that are relied on during certain postulated fire events

Residual Heat Removal components subject to an aging management review include pumps and valves (pressure boundary only), heat exchangers, orifices, piping, tubing, and fittings. The intended functions for Residual Heat Removal

components subject to an aging management review include pressure boundary integrity, heat transfer, and throttling. A complete list of Residual Heat Removal components requiring an aging management review and the component intended functions are provided in Table 3.3-5. The aging management review for Residual Heat Removal is discussed in Section 3.3.

2.3.2.6 EMERGENCY CONTAINMENT FILTRATION

Emergency Containment Filtration serves to reduce the iodine concentration in the containment atmosphere, following a loss-of-coolant accident with failed fuel, to levels ensuring that the offsite dose will not exceed the guidelines of 10 CFR 100 at the site boundary and to assist in limiting the dose to the control room operators to less than 10 CFR 50, Appendix A, General Design Criterion (GDC) 19 limits. Emergency Containment Filtration consists of three filter units, each containing a moisture separator, a high-efficiency particulate filter bank, an impregnated charcoal filter bank, and a fan. Included in the scope of Emergency Containment Filtration are components carrying water from Containment Spray to Emergency Containment Filtration for filter spray. Filter spray provides cooling of the filter in the unlikely event of a post-accident fan trip. Emergency Containment Filtration is described in UFSAR Section 6.3.

The flow diagrams listed in Table 2.3-4 show the evaluation boundaries for the portions of Emergency Containment Filtration that are within the scope of license renewal.

Emergency Containment Filtration is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are a part of the Environmental Qualification Program

Emergency Containment Filtration components subject to an aging management review include the filter units and valves (pressure boundary only), piping, tubing, fittings, and spray nozzles. The intended functions for Emergency Containment Filtration components subject to an aging management review include pressure boundary integrity and spray. A complete list of Emergency Containment Filtration

components requiring an aging management review and the component intended functions are provided in Table 3.3-6. The aging management review for Emergency Containment Filtration is discussed in Section 3.3.

2.3.2.7 CONTAINMENT POST ACCIDENT MONITORING AND CONTROL

Containment Post Accident Monitoring and Control includes the following subsystems:

- Post Accident Hydrogen Monitoring
- Containment Pressure Monitoring
- Post Accident Sampling
- Post Accident Hydrogen Control
- Containment Air Particulate and Gas Monitoring

This subsection addresses the mechanical SCs that are required to support the system intended functions of these subsystems. The screening results for electrical/I&C SCs are provided in Section 2.5 of this Application. Two subsystems of the Containment Post Accident Monitoring and Control System, Containment Water Level Monitoring and Containment High Range Radiation Monitoring, do not contain mechanical SCs required to support the intended functions of these subsystems. Therefore, SCs associated with the Containment Water Level Monitoring and Containment High Range Radiation Monitoring subsystems are addressed in Section 2.5.

The flow diagrams listed in Table 2.3-4 show the evaluation boundaries for the portions of the Containment Post Accident Monitoring and Control System that are within the scope of license renewal.

Post accident hydrogen monitoring provides indication of the hydrogen gas concentration in the containment atmosphere following a loss-of-coolant accident. The mechanical portions of post accident hydrogen monitoring provide a flow path from the containment to the hydrogen monitors and then back to containment. Post accident hydrogen monitoring is described in UFSAR Section 9.14.

Containment pressure monitoring consists of redundant containment pressure signals that are provided to isolate the containment and initiate several reactor safeguard actions. The mechanical portions of containment pressure monitoring

provide sensing lines from the containment to the containment pressure monitors. Containment pressure monitoring is described in UFSAR Section 7.5.

The only mechanical portion of post accident sampling in the scope of license renewal is the sample cooler because it forms a part of the Component Cooling Water pressure boundary. Component Cooling Water is described in UFSAR Section 9.3.

Post accident hydrogen control provides the means for achieving and maintaining containment post accident hydrogen control. Post accident hydrogen control is described in UFSAR Section 9.12.

Containment air particulate and gas monitoring measures radioactivity in the containment air. The mechanical portions of containment air particulate and gas monitoring provide a flow path from the containment to the monitors and then back to the containment. Containment air particulate and gas monitoring is described in UFSAR Section 11.2.3.

Containment Post Accident Monitoring and Control is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are a part of the Environmental Qualification Program
- SCs that are relied on during station blackout

Containment Post Accident Monitoring and Control components subject to an aging management review include pumps and valves (pressure boundary only), orifices, piping, tubing, and fittings. The intended functions for Containment Post Accident Monitoring and Control components subject to an aging management review include pressure boundary integrity and throttling. A complete list of Containment Post Accident Monitoring and Control components requiring an aging management review and the component intended functions are provided in Table 3.3-7. The aging management review for Containment Post Accident Monitoring and Control is discussed in Section 3.3.

2.3.3 AUXILIARY SYSTEMS

Auxiliary Systems are those systems used to support normal and emergency plant operations. The systems provide cooling, ventilation, sampling, and other required functions. Unless noted otherwise, the Auxiliary Systems for Turkey Point Units 3 and 4 are the same. The following systems are included in this subsection:

- Intake Cooling Water
- Component Cooling Water
- Spent Fuel Pool Cooling
- Chemical and Volume Control
- Primary Water Makeup
- Sample Systems
- Waste Disposal
- Instrument Air
- Normal Containment And Control Rod Drive Mechanism Cooling
- Auxiliary Building Ventilation
- Control Building Ventilation
- Emergency Diesel Generator Building Ventilation
- Turbine Building Ventilation
- Fire Protection
- Emergency Diesel Generators and Support Systems

2.3.3.1 INTAKE COOLING WATER

Intake Cooling Water removes heat from Component Cooling Water and Turbine Plant Cooling Water. The Intake Cooling Water pumps supply salt water from the plant's intake area through two redundant piping headers to the tube side of the Component Cooling Water and Turbine Plant Cooling Water heat exchangers. Flow is routed from the heat exchangers to the plant discharge canal. Intake Cooling Water is described in UFSAR Section 9.6.2.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Intake Cooling Water that are within the scope of license renewal. Note: The Component Cooling Water heat exchangers were considered to be part of Component Cooling Water and were screened with that system. (See Subsection 2.3.3.2.)

Intake Cooling Water is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires and station blackout events

Intake Cooling Water components subject to an aging management review include: pumps and valves (pressure boundary only), strainers, orifices, piping, tubing, and fittings. The intended functions for Intake Cooling Water components subject to an aging management review are pressure boundary integrity, filtration, structural integrity, structural support, and throttling. For a complete list of Intake Cooling Water components that require aging management review and the component intended functions, see Table 3.4-1. The aging management review for Intake Cooling Water is discussed in Section 3.4.

2.3.3.2 COMPONENT COOLING WATER

Component Cooling Water removes heat from safety-related and non-safety related components during normal and emergency operation. The component cooling water pumps circulate component cooling water through heat exchangers and coolers that are associated with other systems to transfer heat from those systems to Component Cooling Water. The component cooling water heat exchangers transfer heat from Component Cooling Water to Intake Cooling Water. Component Cooling Water is described in UFSAR Section 9.3.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Component Cooling Water that are within the scope of license renewal. Note: Other coolers and heat exchangers cooled by Component Cooling Water were considered part of their respective systems. Accordingly, these coolers and heat exchangers were screened with those systems.

Component Cooling Water is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the Environmental Qualification Program
- SCs that are relied on during postulated fires and station blackout events

Component Cooling Water components subject to an aging management review include: pumps and valves (pressure boundary only), heat exchangers, tanks, orifices, piping, tubing, and fittings. The intended functions for Component Cooling Water components subject to an aging management review include pressure boundary integrity, heat transfer, and throttling. For a complete list of Component Cooling Water components that require aging management review and the component intended functions, see Table 3.4-2. The aging management review for Component Cooling Water is discussed in Section 3.4.

2.3.3.3 SPENT FUEL POOL COOLING

Spent Fuel Pool Cooling removes decay heat from the spent fuel pool and filters and demineralizes the water in the spent fuel pool. There are two spent fuel pools and Spent Fuel Pool Cooling Systems. Spent Fuel Pool Cooling consists of three separate loops: cooling, purification, and skimmer loops.

The cooling loop removes heat from the spent fuel pool by circulating water through the spent fuel pool heat exchanger. The heat from the spent fuel pool is transferred to Component Cooling Water. The purification loop filters and demineralizes the spent fuel pool water by circulating a portion of the cooling loop flow through a filter and demineralizer. The skimmer loop removes dust and debris from the water surface of the spent fuel pool by taking a suction on the skimmer and circulating the water through strainers and filters. Spent Fuel Pool Cooling is described in UFSAR Section 9.3 and Appendix 14D.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Spent Fuel Pool Cooling that are within the scope of license renewal.

Spent Fuel Pool Cooling is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during station blackout events

Spent Fuel Pool Cooling components subject to an aging management review include: pumps and valves (pressure boundary only), heat exchangers, filters, demineralizers, orifices, piping, tubing, and fittings. The intended functions for Spent Fuel Pool Cooling components subject to an aging management review include pressure boundary integrity, heat transfer, and throttling. For a complete list of Spent Fuel Pool Cooling components that require aging management review and the component intended functions, see Table 3.4-3. The aging management review for Spent Fuel Pool Cooling is discussed in Section 3.4.

2.3.3.4 CHEMICAL AND VOLUME CONTROL

Chemical and Volume Control provides a continuous feed and bleed for the Reactor Coolant System to maintain proper water level and to adjust boron concentration. Chemical and Volume Control includes Boron Addition and Supply, which provides makeup, transfers boric acid solution, and maintains reactor water purity. Some components of Boron Addition and Supply are common to Turkey Point Units 3 and 4. Chemical and Volume Control is described in UFSAR Section 9.2.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Chemical and Volume Control that are within the scope of license renewal. Insulation is not within the scope of license renewal for Chemical and Volume Control because the systems do not contain boric acid solutions at concentrations that require heat tracing, tank heaters, and/or insulation to prevent precipitation.

Chemical and Volume Control is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions

- SCs that are part of the Environmental Qualification Program
- SCs that are relied on during postulated fires and station blackout events

Chemical and Volume Control components subject to an aging management review include: pumps and valves (pressure boundary only), tanks, heat exchangers, orifices, piping, tubing, and fittings. The intended functions for Chemical and Volume Control components subject to an aging management review include pressure boundary integrity, heat transfer, and throttling. For a complete list of Chemical and Volume Control components that require aging management review and the component intended functions, see Table 3.4-4. The aging management review for Chemical and Volume Control is discussed in Section 3.4.

2.3.3.5 PRIMARY WATER MAKEUP

Primary Water Makeup provides demineralized and deaerated water for makeup to various systems throughout the plant. The Turkey Point Units 3 and 4 Primary Water Makeup Systems are operated independently, however, the systems can be cross-connected. Primary Water Makeup is described in UFSAR Section 9.6.2.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Primary Water Makeup that are within the scope of license renewal.

Primary Water Makeup is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires and station blackout events

Primary Water Makeup components subject to an aging management review include: valves (pressure boundary only), piping, tubing, and fittings. The intended function for Primary Water Makeup components subject to an aging management review is pressure boundary integrity. For a complete list of Primary Water Makeup components that require aging management review and the component intended functions, see Table 3.4-5. The aging management review for Primary Water Makeup is discussed in Section 3.4.

2.3.3.6 SAMPLE SYSTEMS

Turkey Point Units 3 and 4 Sample Systems each consist of two subsystems: Sample System - Nuclear Steam Supply System and Sample System - Secondary. Both subsystems are designed to operate manually, on an intermittent basis. Samples can be obtained under conditions ranging from full power to cold shutdown.

The Sample System - Nuclear Steam Supply System permits remote sampling of fluids of the primary plant systems. The subsystem is used to evaluate fluid chemistry in the Reactor Coolant, Emergency Core Cooling, and Chemical and Volume Control Systems. The Sample System - Nuclear Steam Supply System is described in UFSAR Section 9.4.

The Sample System - Secondary permits remote sampling of fluids of the secondary systems. The subsystem is used to evaluate fluid chemistry in the Feedwater, Condensate/Condenser Hotwell, Steam Generator Blowdown, Main Steam, and Heater Drain Systems. A description of the Sample System – Secondary is not included in the UFSAR.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of the Sample Systems that are within the scope of license renewal.

Sample Systems are in the scope of license renewal because they contain:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the Environmental Qualification Program
- SCs that are relied on during postulated fires, anticipated transients without scram and station blackout events

Sample Systems components subject to an aging management review include: valves and coolers (pressure boundary only), piping, tubing, and fittings. The intended functions for Sample Systems components subject to an aging management review include pressure boundary integrity and throttling. For a complete list of Sample Systems components that require aging management review and the component intended functions, see Table 3.4-6. The aging management review for Sample Systems is discussed in Section 3.4.

2.3.3.7 WASTE DISPOSAL

Waste Disposal collects and processes potentially radioactive reactor plant wastes prior to release or removal from the plant site. The system is common to Units 3 and 4 except for the components associated with each containment. Waste Disposal consists of three subsystems: liquid; solid; and gaseous waste disposal systems. Waste Disposal is described in UFSAR Section 11.1.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Waste Disposal that are within the scope of license renewal.

Waste Disposal is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the Environmental Qualification Program
- SCs that are relied on during postulated fires and station blackout events

Waste Disposal components subject to an aging management review include: pumps, valves and heat exchangers (pressure boundary only), piping, tubing, and fittings. The intended function for Waste Disposal components subject to an aging management review is pressure boundary integrity. For a complete list of Waste Disposal components that require aging management review and the component intended functions, see Table 3.4-7. The aging management review for Waste Disposal is discussed in Section 3.4.

2.3.3.8 INSTRUMENT AIR

Instrument Air provides a reliable source of dry, oil-free air for instrumentation and controls and pneumatic valves. Instrument Air provides motive power and control air to safety-related and non-safety related components. Instrument Air contains both electric driven and diesel driven air compressors. Instrument Air is described in UFSAR Section 9.17.

Safety-related air operated valves, normally supplied by Instrument Air, which are required to operate following design basis events are provided with backup sources

of either air or nitrogen. These backup sources are considered safety related and were screened with the particular valves they serve.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Instrument Air that are within the scope of license renewal. Note that some of the license renewal boundaries have been established at normally open valves. This approach is considered acceptable for Instrument Air for the following reasons:

- Instrument Air supplies air to many active components required for normal plant operation, and loss or reduction of air pressure due to degraded conditions is detected early.
- Instrument Air is predominantly constructed of galvanized carbon steel and bronze with an internal environment of dry air, making it very resistant to general corrosion.
- The limited number of valves that rely on Instrument Air are only required for maintaining hot standby conditions for SBO events, or achieving cold shutdown during and following design basis fires. Both of these situations would permit ample time for manual isolation of portions of Instrument Air not within the scope of license renewal, if required.

Instrument Air is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the Environmental Qualification Program
- SCs that are relied on during postulated fires and station blackout events

Instrument Air components subject to an aging management review include: valves (pressure boundary only), flasks/tanks, filters, strainers, heat exchangers, orifices, piping, tubing, and fittings. The intended functions for Instrument Air components subject to an aging management review include pressure boundary integrity, heat transfer, filtration, and throttling. For a complete list of Instrument Air components that require aging management review and the component intended functions, see Table 3.4-8. The aging management review for Instrument Air is discussed in Section 3.4.

2.3.3.9 NORMAL CONTAINMENT AND CONTROL ROD DRIVE MECHANISM COOLING

Normal Containment and Control Rod Drive Mechanism Cooling provides air circulation and cooling to maintain containment bulk ambient temperature below design limits and to remove heat from the Control Rod Drive Mechanisms.

Normal Containment and Control Rod Drive Mechanism Cooling is described in UFSAR Section 9.10.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Normal Containment and Control Rod Drive Mechanism Cooling that are within the scope of license renewal.

Normal Containment and Control Rod Drive Mechanism Cooling is in the scope of license renewal because it contains:

- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires and station blackout events

Normal Containment and Control Rod Drive Mechanism Cooling components subject to an aging management review include: heat exchangers, coolers, ductwork, tubing, and fittings. The intended functions for Normal Containment and Control Rod Drive Mechanism Cooling components subject to an aging management review include pressure boundary integrity, heat transfer, and structural support. For a complete list of Normal Containment and Control Rod Drive Mechanism Cooling components that require aging management review and the component intended functions, see Table 3.4-9. The aging management review for Normal Containment and Control Rod Drive Mechanism Cooling is discussed in Section 3.4.

2.3.3.10 AUXILIARY BUILDING VENTILATION

Auxiliary Building Ventilation provides adequate heat removal to ensure proper operation of safety-related equipment in the auxiliary building. Auxiliary Building Ventilation includes Electrical Equipment Room Ventilation.

Auxiliary Building Ventilation is common to both units. The system provides clean air to the operating areas of the auxiliary building and exhausts air from the

equipment rooms and open areas of the auxiliary building. Auxiliary Building Ventilation is described in UFSAR Section 9.8.1.

Electrical Equipment Room Ventilation is the same for Turkey Point Units 3 and 4. Electrical Equipment Room Ventilation provides cooling for the electrical equipment room under normal and emergency conditions. During normal operations, non-safety related chillers maintain the desired room temperature. In the event of a failure of the non-safety related system or a loss of offsite power, safety-related air conditioners will perform the same function. Electrical Equipment Room Ventilation is described in UFSAR Section 9.8.2.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Auxiliary Building Ventilation and Electrical Equipment Room Ventilation that are within the scope of license renewal.

Auxiliary Building Ventilation and Electrical Equipment Room Ventilation are in the scope of license renewal because they contain:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires and station blackout events

Auxiliary Building Ventilation and Electrical Equipment Room Ventilation components subject to an aging management review include: air handlers (pressure boundary only), filters, ductwork, tubing, and fittings. The intended function for Auxiliary Building Ventilation and Electrical Equipment Room Ventilation components subject to an aging management review is pressure boundary integrity. For a complete list of Auxiliary Building Ventilation and Electrical Equipment Room Ventilation components that require aging management review and the component intended functions, see Table 3.4-10. The aging management review for Auxiliary Building Ventilation and Electrical Equipment Room Ventilation is discussed in Section 3.4.

2.3.3.11 CONTROL BUILDING VENTILATION

Control Building Ventilation provides a temperature controlled environment to ensure proper operation of equipment in the control building. Control Building Ventilation is

composed of three subsystems: Control Room Ventilation; Computer/Cable Spreading Room Ventilation; and DC Equipment/Inverter Room Ventilation. These subsystems are common for Turkey Point Units 3 and 4.

Control Room Ventilation circulates air from the control room and the control room offices through roughing filters to the air handling units. Conditioned air is returned and distributed throughout the control room. Control Room Ventilation maintains the habitability of the control room following design basis events. Control Room Ventilation is described in UFSAR Section 9.9.1.

Computer/Cable Spreading Room Ventilation maintains the temperature and humidity requirements of the vital electrical equipment installed in the computer and cable spreading rooms. It also provides sufficient ventilation for intermittent occupancy by operations and maintenance personnel. Computer/Cable Spreading Room Ventilation is described in UFSAR Section 9.9.3.

DC Equipment/Inverter Room Ventilation provides cooling to the rooms that house the safety-related battery banks, battery chargers, inverters, and DC load centers. DC Equipment/Inverter Room Ventilation is described in UFSAR Section 9.9.2.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Control Building Ventilation that are within the scope of license renewal.

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

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires and station blackout events

Control Building Ventilation components subject to an aging management review include: air handling units and valves (pressure boundary only), heat exchangers, ductwork, piping, tubing, and fittings. The intended functions for Control Building Ventilation components subject to an aging management review include pressure boundary integrity, throttling, and heat transfer. For a complete list of Control Building Ventilation components that require aging management review and the component intended functions, see Table 3.4-11. The aging management review for Control Building Ventilation is discussed in Section 3.4.

2.3.3.12 EMERGENCY DIESEL GENERATOR BUILDING VENTILATION

Emergency Diesel Generator Building Ventilation is required to provide cooling functions for the emergency diesel generators and associated equipment. Emergency Diesel Generator Building Ventilation is different for Turkey Point Units 3 and 4. Emergency Diesel Generator Building Ventilation is necessary to ensure proper operation of the emergency diesel generators and other safety-related electrical equipment.

Unit 3 Emergency Diesel Generator Building Ventilation consists of one wall mounted exhaust fan and associated ductwork for each emergency diesel generator. The fan operates to maintain cooling in the room when its associated emergency diesel generator is running.

Unit 4 Emergency Diesel Generator Building Ventilation includes the following subsystems: Emergency Diesel Generator Room Ventilation; Diesel Control Room Ventilation; and 3D and 4D Switchgear Room Ventilation. Unit 4 Emergency Diesel Generator Building Ventilation is described in UFSAR Section 8.2.2.1.1.3.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Emergency Diesel Generator Building Ventilation that are within the scope of license renewal. Note: There is no flow diagram for Unit 3 Emergency Diesel Generator Building Ventilation, however, all components associated with this system are in the scope of license renewal.

Emergency Diesel Generator Building Ventilation is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Emergency Diesel Generator Building Ventilation components subject to an aging management review include: filters (pressure boundary only), ductwork, tubing, and fittings. The intended function for Emergency Diesel Generator Building Ventilation components subject to an aging management review is pressure boundary integrity.

For a complete list of Emergency Diesel Generator Building Ventilation components that require aging management review and the component intended functions, see Table 3.4-12. The aging management review for Emergency Diesel Generator Building Ventilation is discussed in Section 3.4.

2.3.3.13 TURBINE BUILDING VENTILATION

Turbine Building Ventilation provides a temperature controlled environment to ensure proper operation of equipment in the turbine building. Turbine Building Ventilation consists of two subsystems: Load Center and Switchgear Rooms Ventilation; and Steam Generator Feed Pump Ventilation.

Load Center and Switchgear Rooms Ventilation provides a temperature controlled environment for the safety-related 4160V switchgear and 480V load centers, located in the rooms, during normal and emergency conditions. Load Center and Switchgear Rooms Ventilation is described in UFSAR Section 9.16.

Steam Generator Feed Pump Ventilation provides cooling to the steam generator feed pump room. The subsystem is non-safety related, performs no safety-related functions, and is not in the scope of license renewal.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Turbine Building Ventilation that are within the scope of license renewal.

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

- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires and station blackout events

Turbine Building Ventilation components subject to an aging management review include: pumps, valves, and air handling units (pressure boundary only); heat exchangers, piping, tubing, and fittings. The intended functions for Turbine Building Ventilation components subject to an aging management review include pressure boundary integrity, throttling, and heat transfer. For a complete list of Turbine Building Ventilation components that require aging management review and the component intended functions, see Table 3.4-13. The aging management review for Turbine Building Ventilation is discussed in Section 3.4.

2.3.3.14 FIRE PROTECTION

Fire Protection protects plant equipment in the event of a fire, to ensure safe plant shutdown, and minimizes the risk of a radioactive release to the environment. Fire Protection consists of Fire Water Supply including sprinklers, Halon Suppression, Fire Dampers, RCP Oil Collection, Alternate Shutdown, Safe Shutdown, and Fire Detection and Protection. Individual components that constitute Alternate Shutdown and Safe Shutdown were screened with their respective systems. Fire Detection and Protection was screened with Electrical and Instrumentation and Controls (See Section 2.5). Fire Protection is described in UFSAR Appendix 9.6A. The majority of Fire Protection is common to Units 3 and 4.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Fire Protection that are within the scope of license renewal.

Fire Protection is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires

Fire Protection components subject to an aging management review include the raw water tanks, pumps and valves (pressure boundary only), tanks, heat exchangers, hose stations, flame arrestors, sprinklers, strainers, orifices, piping, tubing, and fittings. The intended functions for Fire Protection components subject to an aging management review are pressure boundary integrity, heat transfer, filtration, throttling, fire spread prevention, and spray. For a complete list of the Fire Protection components that require aging management review and the component intended functions, see Tables 3.4-14 and 3.6-12. The aging management reviews for Fire Protection are discussed in Section 3.4 and Subsection 3.6.2. Fire extinguishers, fire hoses, and air packs are not subject to an aging management review because they are replaced based on condition in accordance with National Fire Protection Association (NFPA) standards and plant surveillance procedures for fire protection equipment. This position is consistent with the NRC Staff's guidance on consumables provided in the NRC's March 10, 2000, letter from Christopher I. Grimes to Douglas J. Walters [Reference 2.3-12].

2.3.3.15 EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEMS

The Emergency Diesel Generators provide AC power to the onsite electrical distribution system to assure the capability for a safe and orderly shutdown. The Emergency Diesel Generators Support Systems necessary to ensure proper operation of the Emergency Diesel Generators are:

- Air Intake and Exhaust
- Air Start
- Fuel Oil
- Cooling Water
- Lube Oil

The Emergency Diesel Generators are described in UFSAR Section 8.2.2.1.1.1 and the Emergency Diesel Generators Support Systems are described in Section 9.15.

The Unit 3 emergency diesel generator fuel oil storage tank is a free-standing steel tank. The Unit 4 emergency diesel generator fuel oil storage tank is a concrete structure with a steel liner that is an integral part of the Unit 4 emergency diesel generator building.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of Emergency Diesel Generators and Support Systems that are within the scope of license renewal.

Emergency Diesel Generators and Support Systems are in the scope of license renewal because they contain:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Emergency Diesel Generators and Support Systems components subject to an aging management review include: two Diesel Oil Storage Tanks, pumps and valves (pressure boundary only), tanks, heat exchangers, flame arrestors, filters, strainers,

piping, tubing, and fittings. The intended functions for Emergency Diesel Generators and Support Systems components subject to an aging management review include pressure boundary integrity, filtration, heat transfer, throttling, and fire spread prevention. For a complete list of the Emergency Diesel Generators and Support Systems components that require aging management review and component intended functions, see Table 3.4-15. The aging management review for the Emergency Diesel Generators and Support Systems are discussed in Section 3.4.

2.3.4 STEAM AND POWER CONVERSION SYSTEMS

The Steam and Power Conversion Systems act as a heat sink to remove heat from the reactor and convert the heat generated in the reactor to the plant's electrical output. Unless noted otherwise, the Steam and Power Conversion Systems for Turkey Point Units 3 and 4 are the same. The following systems are included in this subsection:

- Main Steam and Turbine Generators
- Feedwater and Blowdown
- Auxiliary Feedwater and Condensate Storage

2.3.4.1 MAIN STEAM AND TURBINE GENERATORS

Main Steam transports saturated steam from the steam generators to the main turbine and other secondary steam system components. Main Steam provides the principal heat sink for the Reactor Coolant System protecting the Reactor Coolant System and the steam generators from overpressurization, provides isolation of the steam generators during a postulated steam line break, and provides steam supply to the Auxiliary Feedwater pump turbines.

Turbine Generators convert the steam input from Main Steam to the plant's electrical output, provide first-stage pressure input to the reactor protection system, and provide isolation under certain postulated steam line break scenarios. Main Steam and Turbine Generators are described in UFSAR Section 10.2.2.

The flow diagrams listed in Table 2.3-6 show the evaluation boundaries for the mechanical portions of Main Steam and Turbine Generators that are within the scope of license renewal.

Main Steam is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the Environmental Qualification Program

- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Turbine Generators are in the scope of license renewal because they contain:

- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during anticipated transients without scram events

Main Steam and Turbine Generators components subject to an aging management review include: valves (pressure boundary only), steam traps, flow elements, piping, tubing, and fittings. The intended functions for Main Steam and Turbine Generators components subject to an aging management review are pressure boundary integrity and throttling. For a complete list of Main Steam and Turbine Generators components that require aging management review and the component intended functions, see Table 3.5-1. The aging management review for Main Steam and Turbine Generators is discussed in Section 3.5.

2.3.4.2 FEEDWATER AND BLOWDOWN

Feedwater and Blowdown provide sufficient water flow to the steam generators to maintain an adequate heat sink for the Reactor Coolant System, provide for feedwater and blowdown isolation following a postulated loss-of-coolant accident or steam line break event, and assist in maintaining steam generator water chemistry. Feedwater and Blowdown consists of three subsystems: Main Feedwater; Steam Generator Blowdown; and Standby Steam Generator Feedwater.

Main Feedwater supplies pre-heated, high-pressure feedwater to the steam generators at a rate equal to main steam and the steam generator blowdown flows. The feedwater flow rate is controlled by the Steam Generator Level Control System which determines the desired feedwater flow by comparing the feed flow, steam flow, and steam generator level. Main Feedwater is described in UFSAR Section 10.2.2.

Steam Generator Blowdown assists in maintaining required steam generator chemistry by providing a means for removal of foreign matter that concentrates in the evaporator section of the steam generator. Steam Generator Blowdown is fed by three independent blowdown lines (one per steam generator), which tie to a common blowdown flask. Steam generator blowdown is continuously monitored for

radioactivity during plant operation. Steam Generator Blowdown is described in UFSAR Section 10.2.4.3.

Standby Steam Generator Feedwater is common to Turkey Point Units 3 and 4. Standby Steam Generator Feedwater supplies steam generator feedwater during normal startup, shutdown, and hot standby conditions. Standby Steam Generator Feedwater delivers sufficient feedwater to maintain one unit at hot standby while providing makeup for maximum blowdown. The Standby Steam Generator Feedwater pumps take suction from the demineralized water storage tank and discharge to a common header upstream of the feedwater regulating valves. Standby Steam Generator Feedwater is described in UFSAR Section 9.11.

The flow diagrams listed in Table 2.3-6 show the evaluation boundaries for the portions of Feedwater and Blowdown that are within the scope of license renewal.

Main Feedwater is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Steam Generator Blowdown is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the Environmental Qualification Program
- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Standby Steam Generator Feedwater is in the scope of license renewal because it contains:

- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions

- SCs that are relied on during postulated fires

Feedwater and Blowdown components subject to an aging management review include the Demineralized Water Storage Tank, pumps and valves (pressure boundary only), orifices, piping, tubing, and fittings. The intended functions for the Feedwater and Blowdown components subject to an aging management review are pressure boundary integrity and throttling. For a complete list of Feedwater and Blowdown components that require aging management review and the component intended functions, see Table 3.5-2. The aging management review for Feedwater and Blowdown is discussed in Section 3.5.

2.3.4.3 AUXILIARY FEEDWATER AND CONDENSATE STORAGE

Auxiliary Feedwater supplies feedwater to the steam generators when normal feedwater sources are not available, provides for auxiliary feedwater steam and feedwater isolation during a postulated steam generator tube rupture event, and provides for auxiliary feedwater isolation to the faulted steam generator and limits feedwater flow to the steam generators to limit positive reactivity insertion during a postulated steam line break event. Auxiliary Feedwater is a shared system between Turkey Point Units 3 and 4.

Auxiliary Feedwater contains three steam turbine driven pumps. The pumps can be supplied steam from the steam generators in either unit. The pumps take suction from either condensate storage tank and discharge to one of two redundant headers. Each header can supply each steam generator. Auxiliary Feedwater is normally maintained in standby with one pump aligned to one discharge header and two pumps aligned to the other header. Upon initiation, all three pumps start to supply the affected steam generator with feedwater. Auxiliary Feedwater is described in UFSAR Section 9.11.

Condensate Storage stores water for use by Auxiliary Feedwater to support safe shutdown of the plant. Condensate Storage consists of a condensate storage tank on each unit with piping that feeds all three auxiliary feedwater pumps. The tank outlet piping is cross-connected between the units so that either tank can supply the water required by Auxiliary Feedwater. Condensate Storage is described in UFSAR Section 9.11.3.

The flow diagrams listed in Table 2.3-6 show the evaluation boundaries for the portions of Auxiliary Feedwater and Condensate Storage that are within the scope of license renewal.

Auxiliary Feedwater is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are non-safety related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the Environmental Qualification Program
- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Condensate Storage is in the scope of license renewal because it contains:

- SCs that are safety related and are relied upon to remain functional during and following design basis events
- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Auxiliary Feedwater and Condensate Storage components subject to an aging management review include: Condensate Storage Tanks, pumps and valves (pressure boundary only), coolers, orifices, piping, tubing, and fittings. The intended functions for Auxiliary Feedwater and Condensate Storage components subject to an aging management review are pressure boundary integrity, heat transfer, and throttling. For a complete list of Auxiliary Feedwater and Condensate Storage components that require aging management review and the component intended functions, see Table 3.5-3. The aging management review for Auxiliary Feedwater and Condensate Storage is discussed in Section 3.5.

2.3.5 REFERENCES

2.3-1 WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," Revision 1, August 1996.

2.3-2 R. A. Newton (WOG) letter to U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated April 18, 1997) for Additional Information on WOG Generic Technical Report WCAP-14575, License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," June 13, 1997.

2.3-3 R. A. Newton (WOG) letter to U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated June 4, 1999) for Additional Information on WOG Generic Technical Reports WCAP-14574, Aging Management Evaluation for Pressurizers, and WCAP-14575, Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," July 19, 1999.

2.3-4 C. I. Grimes (NRC) letter to R. A. Newton (WOG), "Draft Safety Evaluation Concerning Westinghouse Owners Group License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, WCAP-14575, Revision 1, August 1996," February 10, 2000.

2.3-5 WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," Revision 0, July 1996.

2.3-6 R. A. Newton (WOG) letter to U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated May 6, 1997) for Additional Information on WOG Generic Technical Report WCAP-14574, License Renewal Evaluation: Aging Management for Pressurizers," May 30, 1997.

2.3-7 C. I. Grimes (NRC) letter to R. A. Newton (WOG), “Draft Safety Evaluation Concerning the Westinghouse Owners Group License Renewal Evaluation: Aging Management Evaluation for Pressurizers, WCAP-14574, Revision 0, July 1996,” August 7, 2000.

2.3-8 WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," Revision 0, June 1997.

2.3-9 R. A. Newton (WOG) letter to U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated June 14, 1999) for Additional Information on WOG Generic Technical Reports: WCAP-14577, License Renewal Evaluation: Aging Management for Reactor Vessel Internals," November 24, 1999.

2.3-10 WCAP-14422, "License Renewal Evaluation: Aging Management for Reactor Coolant System Supports," Revision 2, March 1997.

2.3-11 C. I. Grimes (NRC) to letter R. A. Newton (WOG), "Draft Safety Evaluation Concerning the Westinghouse Owners Group License Renewal Evaluation: Aging Management for Reactor Coolant System Supports, WCAP-14422, Revision 2," February 25, 2000.

2.3-12 Christopher I. Grimes (NRC) letter to Douglas J. Walters (NEI), “License Renewal Issue No. 98-12, Consumables," March 10, 2000.

2.3-13 Richard P. Croteau (NRC) letter to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping," June 23, 1995.

2.3-14 FPL letter L-2000-176 to U. S. Nuclear Regulatory Commission, “License Renewal Boundary Drawings."

TABLE 2.3-1
REACTOR COOLANT SYSTEM EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Reactor Coolant	
3-RCS-01	0
3-RCS-02	0
3-RCS-03	0
3-RCS-04	0
4-RCS-01	0
4-RCS-02	0
4-RCS-03	0
4-RCS-04	0
Safety Injection	
3-SI-01	0
3-SI-03	0
4-SI-01	0
4-SI-03	0
Residual Heat Removal	
3-RHR-01	0
4-RHR-01	0
Sample System – Nuclear Steam Supply System	
3-SAMP-03	0
4-SAMP-03	0
Chemical and Volume Control	
3-CVCS-02	0
3-CVCS-03	0
4-CVCS-02	0
4-CVCS-03	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-2
CLASS 1 PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS - APPLICANT ACTION ITEMS FROM SECTION 4.1 OF WCAP-14575 DRAFT SAFETY EVALUATION

Renewal Applicant Action Item	Turkey Point-Specific Response
(1) The license renewal applicant is to verify that its plant is bounded by the technical report. Further, the renewal applicant is to commit to programs described as necessary in the technical report to manage the effects of aging during the period of extended operation on the functionality of the reactor coolant system piping. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within this technical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor coolant system piping and associated pressure boundary components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).	As summarized in Subsections 2.3.1.2 and 2.3.1.7, the Turkey Point Class 1 piping and reactor coolant pumps are bounded by the topical report with regard to design criteria and features, materials of construction, fabrication techniques, installed configuration, modes of operation and environments/exposures. Programs necessary to manage the effects of aging are described in Subsections 3.2.1 and 3.2.6 and are summarized in Table 3.2-1. Program commitments to manage the effects of aging for Class 1 piping and reactor coolant pumps are described in Appendix B and are summarized in the proposed UFSAR supplement provided in Appendix A. Deviations from the aging management programs included in the topical report are described in Subsections 3.2.1 and 3.2.6.
(2) Summary description of the programs and evaluation of Time-Limited Aging Analyses are to be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).	A summary of the programs identified to manage the effects of aging for Class 1 piping and reactor coolant pumps is included in the proposed UFSAR supplement in Appendix A. A markup of the UFSAR sections affected by the TLAA evaluations is also included in the proposed UFSAR supplement.

TABLE 2.3-2 (continued)
CLASS 1 PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS - APPLICANT ACTION ITEMS FROM SECTION 4.1 OF WCAP-14575 DRAFT SAFETY EVALUATION

Renewal Applicant Action Item	Turkey Point-Specific Response
(3) Applicants must provide a description of all insulation used on austenitic stainless steel Nuclear Steam Supply System piping to ensure the piping is not susceptible to stress-corrosion cracking from halogens.	During construction, the Class 1 piping was insulated in accordance with the applicable Westinghouse Equipment Specification. The specification listed specific tradenames that were approved, by Westinghouse, for use on austenitic stainless steel. As described in the Turkey Point UFSAR, Section 4.2.5 "…external corrosion resistant surfaces in the reactor coolant system are insulated with low halide or halide free insulating material…" During 1979 the insulation on the reactor coolant piping was changed to reflective insulation. The insulation is made of austenitic stainless steel; any nonmetallics comply with NRC Regulatory Guide 1.36. Subsequent additions of insulation were done in accordance with the applicable Bechtel specification, which also imposes the requirements of Regulatory Guide 1.36. Since all the insulation that was used on the reactor coolant piping is low halide or halide free, the piping is not susceptible to stress corrosion cracking initiated by such halides.
(4) The license renewal applicant should describe how each plant-specific AMP addresses the following 10 elements: (1) scope of the program, (2) preventive actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.	Programs necessary to manage the effects of aging for Class 1 piping and reactor coolant pumps address the 10 elements identified. These programs are described in Appendix B.
(5) The license renewal applicant should perform additional fatigue evaluations or propose an AMP to address the components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575.	Turkey Point has performed a plant-specific fatigue evaluation for Turkey Point Class 1 piping and reactor coolant pumps. This evaluation is included in Section 4.3.

TABLE 2.3-2 (continued)
CLASS 1 PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS - APPLICANT ACTION ITEMS FROM SECTION 4.1 OF WCAP-14575 DRAFT SAFETY EVALUATION

Renewal Applicant Action Item	Turkey Point-Specific Response
(6) The staff recommendation for the closure of GSI-190 "Fatigue Evaluation of Metal Components for 60-Year Plant Life" is contained in a December 26, 1999, memorandum from Ashok Thadani to William Travers. The license renewal applicant should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. The evaluation of a sample of components with high-fatigue usage factors using the latest available environmental fatigue data is an acceptable method to address the effects of the coolant environment on component fatigue life.	Turkey Point has performed a plant-specific evaluation for Turkey Point Class 1 piping and reactor coolant pumps with regard to environmental effects on fatigue. This evaluation is included in Subsection 4.3.5.

TABLE 2.3-3
CLASS 1 PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS - OPEN ITEMS FROM SECTION 4.2 OF WCAP-14575 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(1) Westinghouse Owners Group should complete the specific revisions to the subject topical report that it has committed to perform in response to the staff's requests for additional information discussed in Section 3.1 of the safety evaluation. As described by WOG in its letter to the staff, dated July 19, 1999, these planned modifications are limited to Section 2.3.2.2, "Branch Line Restrictors," Section 2.3.2.4, "Thermal Barrier and RCP Seals," and the "summary" sections of the topical report.	The Turkey Point Class 1 piping aging management review includes branch line restrictors and their associated license renewal component intended function of throttling. The aging management review of the Class 1 piping is addressed in Subsection 3.2.1 and summarized in Table 3.2-1. The Turkey Point position regarding reactor coolant pump seals is summarized in Subsection 2.3.1.7.
(2) Westinghouse Owners Group should complete the updated review of generic communications and revise Section 3.1 of the topical report to describe the process used by the Westinghouse Owners Group to perform the review and to capture any additional items not identified by the original review.	Turkey Point has completed an updated review of generic communications for applicability to Class 1 piping and reactor coolant pumps. All generic communications applicable to aging effects are summarized in Subsections 3.2.1 and 3.2.6.
(3) The topical report indicates that thermal aging-related cracking of austenitic steel castings is an aging effect that the Westinghouse Owners Group considers potentially significant for the Reactor Coolant System piping and associated components. Thermal aging does not cause cracking, it causes a reduction in the fracture toughness of the material. The reduction in fracture toughness of the material results in a reduction in the critical flaw size that could lead to component failure. The Westinghouse Owners Group should revise the topical report, accordingly.	Turkey Point's aging management review methodology identifies reduction in fracture toughness as the aging effect related to thermal aging. Reduction in fracture toughness for Class 1 piping and reactor coolant pumps is addressed in Subsections 3.2.1 and 3.2.6.

TABLE 2.3-3 (continued)
CLASS 1 PIPING AND ASSOCIATED PRESSURE BOUNDARY COMPONENTS - OPEN ITEMS FROM SECTION 4.2 OF WCAP-14575 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(4) Components that have delta ferrite levels below the susceptibility screening criteria have adequate fracture toughness and do not require supplemental inspection. As a result of thermal embrittlement, components that have delta ferrite levels exceeding the screening criterion may not have adequate fracture toughness and do require additional evaluation or examination. Westinghouse Owners Group should address thermal-aging issues in accordance with the staff's comments in Section 3.3.3 of this evaluation.	As noted above for Open Item 3, reduction in fracture toughness for Class 1 piping and reactor coolant pumps is addressed in Subsections 3.2.1 and 3.2.6. The Turkey Point methodology is consistent with the staff's comments.
(5) Westinghouse Owners Group should propose to perform additional inspection of small-bore Reactor Coolant System piping, that is, less than 4-inch-size piping, for license renewal to provide assurance that potential cracking of small-bore Reactor Coolant System piping is adequately managed during the period of extended operation.	The aging management review and specific program commitments for Class 1 small bore piping are addressed in Subsection 3.2.1 and summarized in Table 3.2-1.
(6) Westinghouse Owners Group should revise AMP-3.6 to include an assessment of the margin on loads in conformance with the staff guidance provided in Reference 11. In addition, AMP-3.6 should be revised to indicate If the CASS component is repaired or replaced per ASME Code, Section XI IWB-4000 or IWB-7000, a new LBB analysis based on the material properties of the repaired or replaced component (and accounting for its thermal aging through the period of extended operation, as appropriate), is required to confirm the applicability of LBB. The inservice examination/flaw evaluation option is, per the basis on which the NRC staff has approved LBB in the past, insufficient to reestablish LBB approval.	The original Turkey Point Leak-Before-Break (LBB) analysis was performed consistent with the criteria specified in NUREG-1061, Volume 3, and utilized the modified limit load method as specified in the draft Standard Review Plan, Section 3.6.3. The NRC review and safety evaluation of the original Turkey Point LBB analysis is documented in the June 23, 1995, NRC letter to FPL [Reference 2.3-13]. The revised Turkey Point LBB analysis, which addresses the extended period of operation, utilizes a methodology consistent with the original LBB analysis. If Class 1 piping CASS components are repaired or replaced, Turkey Point design control procedures would require a new LBB analysis based on replacement material properties.

TABLE 2.3-4
ENGINEERED SAFETY FEATURES EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Emergency Containment Cooling	
3-E/NCC-01	0
4-E/NCC-01	0
Containment Spray	
3-CS-01	0
4-CS-01	0
Containment Isolation	
3-BA-01	0
4-BA-01	0
0-N2H2-01	0
3-CP-01	0
4-CP-01	0
Safety Injection	
3-SI-01	0
3-SI-02	0
3-SI-03	0
4-SI-01	0
4-SI-02	0
4-SI-03	0
Residual Heat Removal	
3-RHR-01	0
4-RHR-01	0
Emergency Containment Filtration	
3-ECF-01	0
4-ECF-01	0
Containment Post Accident Monitoring and Control	
0-PAMC-01	0
0-PAMC-02	0
3-PAMC-01	0
4-PAMC-01	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-5
AUXILIARY SYSTEMS
EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Intake Cooling Water	
3-ICW-01	0
3-ICW-02	0
4-ICW-01	0
4-ICW-02	0
Component Cooling Water	
3-CCW-01	0
3-CCW-02	0
3-CCW-03	0
3-CCW-04	0
3-CCW-05	0
4-CCW-01	0
4-CCW-02	0
4-CCW-03	0
4-CCW-04	0
Spent Fuel Pool Cooling	
3-SFP-01	0
3-SI-01	0
4-SFP-01	0
4-SI-01	0
Chemical and Volume Control	
0-CVCS-01	0
0-CVCS-02	0
3-CVCS-01	0
3-CVCS-02	0
3-CVCS-03	0
4-CVCS-01	0
4-CVCS-02	0
4-CVCS-03	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-5 (continued) AUXILIARY SYSTEMS EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Primary Water Makeup	
3-PW-01	0
3-RCS-03	0
3-CVCS-01	0
4-PW-01	0
4-RCS-03	0
4-CVCS-01	0
Sample System – Nuclear Steam Supply System	
3-SAMP-03	0
3-RCS-01	0
3-RCS-02	0
3-CVCS-01	0
3-CVCS-02	0
3-SI-03	0
4-SAMP-03	0
4-RCS-01	0
4-RCS-02	0
4-CVCS-01	0
4-CVCS-02	0
4-SI-03	0
Sample System - Secondary	
3-SAMP-01	0
3-SAMP-02	0
3-FW-04	0
3-MS-01	0
4-SAMP-01	0
4-SAMP-02	0
4-FW-04	0
4-MS-01	0
Waste Disposal	
0-WD-01	0
0-WD-02	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-5 (continued) AUXILIARY SYSTEMS EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Waste Disposal (continued)	
3-WD-01	0
3-RCS-02	0
4-WD-01	0
4-RCS-02	0
Instrument Air	
0-IA-01	0
3-IA-01	0
3-IA-02	0
3-IA-03	0
3-IA-04	0
3-IA-05	0
3-IA-06	0
3-IA-07	0
3-IA-08	0
3-IA-09	0
3-CVCS-02	0
3-CVCS-03	0
3-CP-01	0
3-FW-03	0
3-RHR-01	0
3-MS-01	0
4-IA-01	0
4-IA-02	0
4-IA-03	0
4-IA-04	0
4-IA-05	0
4-CVCS-02	0
4-CVCS-03	0
4-CP-01	0
4-FW-03	0
4-RHR-01	0
4-MS-01	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-5 (continued) AUXILIARY SYSTEMS EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Normal Containment And Control Rod Drive Mechanism Cooling	
3-E/NCC-01	0
4-E/NCC-01	0
Auxiliary Building Ventilation	
0-ABVAC-01	0
0-ABVAC-02	0
Control Building Ventilation	
0-CBVAC-01	0
0-CBVAC-02	0
0-CBVAC-03	0
Emergency Diesel Generator Building Ventilation	
4-EDVAC-01	0
Turbine Building Ventilation	
3-TBVAC-01	0
3-TBVAC-02	0
4-TBVAC-01	0
4-TBVAC-02	0
Fire Protection	
0-FP-01	0
0-FP-02	0
0-FP-03	0
0-FP-04	0
0-FP-05	0
0-FP-06	0
0-FP-07	0
0-FP-08	0
0-FP-09	0
0-FP-10	0
3-RCS-03	0
4-RCS-03	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-5 (continued) AUXILIARY SYSTEMS EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Emergency Diesel Generators and Support Systems	
3-EDG-01	0
3-EDG-02	0
3-EDG-03	0
3-EDG-04	0
3-EDG-05	0
3-EDG-06	0
4-EDG-01	0
4-EDG-02	0
4-EDG-03	0
4-EDG-04	0
4-EDG-05	0
4-EDG-06	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-6
STEAM AND POWER CONVERSION SYSTEMS EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Main Steam and Turbine Generators	
3-MS-01	0
3-MS-02	0
3-MS-03	0
3-SAMP-02	0
3-TG-01	0
4-MS-01	0
4-MS-02	0
4-MS-03	0
4-SAMP-02	0
4-TG-01	0
Feedwater and Blowdown	
0-FW-01	0
0-FW-02	0
3-FW-01	0
3-FW-02	0
3-FW-03	0
3-FW-04	0
4-FW-01	0
4-FW-02	0
4-FW-03	0
4-FW-04	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

TABLE 2.3-6 (continued)
STEAM AND POWER CONVERSION SYSTEMS EVALUATION BOUNDARIES[1]

Drawing Number	Revision
Auxiliary Feedwater and Condensate Storage	
0-AFW-01	0
0-AFW-02	0
3-AFW-01	0
3-AFW-02	0
3-AFW-03	0
3-COND-01	0
4-AFW-01	0
4-AFW-02	0
4-AFW-03	0
4-COND-01	0

NOTE: 1. Drawings submitted separately [Reference 2.3-14].

2.4 SCOPING AND SCREENING RESULTS - STRUCTURES

The determination of structures within the scope of license renewal is made by initially identifying Turkey Point structures and then reviewing them to determine which ones satisfy one or more of the criteria contained in 10 CFR 54.4. This process is described in Section 2.1 and the results of the structures review are contained in Section 2.2.

Section 2.1 also provides the methodology for determining the components within the scope of 10 CFR 54.4 that meet the requirements contained in 10 CFR 54.21(a)(1). The components that meet these screening requirements are identified in this section. These identified components subsequently require an aging management review for license renewal.

The screening results are provided below in two subsections:

- Containments
- Other structures

2.4.1 CONTAINMENTS

Each Turkey Point Containment is a domed concrete, steel reinforced structure that houses the reactor vessel, Reactor Coolant System, Reactor Coolant System supports, and other important systems which interface with the Reactor Coolant System. Additionally, each Containment houses and supports components required for plant refueling. This includes the polar crane, refueling cavity, and portions of the fuel handling system. The Containment for each unit is the third and final barrier against the possible release of radioactive material to the environment during the unlikely event of a failure of the Reactor Coolant System.

The Turkey Point Units 3 and 4 UFSAR classifies the Containments as Class I structures designed to prevent the uncontrolled release of radioactivity. Class I structures have been determined to meet the criteria of 10 CFR 54.4 and are within the scope of license renewal. For screening, each Containment was divided into two categories, Containment Structure and Containment Internal Structural Components. Each Containment category was then subdivided into component/commodity sets to determine those structures and components requiring an aging management review. The component/commodity sets were developed based on a review of Turkey Point plant-controlled drawings, the UFSAR, the plant equipment database, and guidance from NEI 95-10 [Reference 2.4-1]. The Containments are described in UFSAR Section 5.1.

Turkey Point actively participated in a Westinghouse Owners Group effort that developed a series of generic technical reports whose purpose was to demonstrate that aging effects are adequately managed for the period of extended operation. WCAP-14756, “License Renewal Evaluation: Aging Management Evaluation for Pressurized Water Reactor Containment Structures,” was submitted by Westinghouse to the NRC for approval . The draft NRC Safety Evaluation has not been issued for WCAP-14756. Turkey Point does not utilize or credit this generic technical report in this Application.

Containment structural components requiring an aging management review are identified in the following subsections. Note that the discussions below apply to the Containments for both Units 3 and 4.

2.4.1.1 CONTAINMENT STRUCTURE

2.4.1.1.1 CONCRETE

Each Containment structure consists of a post-tensioned reinforced concrete cylinder and a shallow dome connected to and supported by a reinforced concrete foundation slab. The combined strength provided by the concrete, conventional reinforcing steel, and the post-tensioning system is used to satisfy the design loads. Although these components act together as one composite system, the post-tensioning system is described as a separate component/commodity because it is installed and stressed after the reinforced concrete components are complete and because of the unique tendon surveillance program.

DOME AND CYLINDER WALLS

Cast-in-place concrete is used for each containment dome and shell (cylinder wall). In general, the concrete placement in the walls is done in ten-foot high lifts with vertical joints at the radial centerline of each of the six buttresses. Intermediate grade steel is used for bonded reinforcing throughout the cylinder and dome as crack control reinforcing. At areas of discontinuities, higher strength reinforcing steel is used to provide an additional margin of elastic strain capability. The Containment structures are described in UFSAR Section 5.1.2.

Turkey Point uses a waterproofing membrane underneath the foundation mat and outside the lower portions of the Containment Structure wall as a design feature to act as a measure of protection to manage or inhibit the intrusion of groundwater. The exterior containment walls located below grade also have embedded waterstops installed to inhibit the intrusion or seepage of groundwater. Although the membrane system provides a measure of protection, no credit was taken for the continued performance in the determination of aging effects. The waterproofing membrane and waterstops are piece parts and are not identified as unique structures or unique components. The waterproofing membrane and waterstops are considered design features that perform no intended function, and thus, are not in the scope of license renewal.

FLOORS

A reinforced concrete floor is provided in each Containment, above the embedded floor liner, to protect the liner plate from punctures and corrosion that could breach

the essentially leaktight barrier. The containment floors are described in UFSAR Sections 5.1.2 and 5.1.6.

Moisture barriers consisting of 1½"-deep sealing compound are provided between the liner air test system and concrete floors at elevation 14' to prevent intrusion of moisture between the concrete and liner surfaces. The sealing compound is Nukem 720 Elastik made by Amercoat Corporation. The space between the concrete and liner is filled with mastic material ¼" thick.

FOUNDATION SLABS

The conventionally reinforced concrete foundation slab serves as the structural foundation support for each Containment. The vertical tendons extend through the foundation slab and are anchored on the underside of the slab. A reinforced concrete enclosure (tendon access gallery) is provided for each Containment at the underside of the foundation slab perimeter for access to the lower vertical tendon anchorage for tendon inspection and surveillance purposes. High-strength reinforcing steel, mechanically spliced with T-Series Cadwelds, is used throughout the base slabs. The containment foundation slab is described in UFSAR Section 5.1.2.

The function of the tendon access galleries is to provide access to the bottom of the vertical tendons and to the horizontal lower tendons so they can be tested. Loss of function of the tendon access galleries is highly unlikely and to consider such is hypothetical. Per NEI 95-10 [Reference 2.4-1], consideration of hypothetical failures that could result from system interdependencies that are not part of the current licensing basis and that have not been previously experienced is not required. Accordingly, the lower tendon access galleries and the inspection pits do not support the intended function of the Containment structures.

2.4.1.1.2 STEEL

ANCHORS, EMBEDMENTS, AND ATTACHMENTS

Structural steel commodities include anchors, embedments, and attachments such as angles and anchor studs that are welded to the liner and serve to anchor each liner to the containment shells. In addition, other anchors, embedments, and attachments are provided that serve to transfer loads into the concrete cylinder walls or foundation mats from attachments to the liners.

Each liner plate is attached to the concrete by means of an angle grid system welded to the liner plate and embedded in the concrete. The anchor spacing is designed to maintain the essentially leaktight barrier by preserving the integrity of the liner. The load-carrying capacity of these anchorages is also required to assure that supported equipment, such as the polar cranes, can perform safely as required. The polar crane bracket attachments are welded to the liner and embedded in the concrete shell. Containment anchors, embedments, and attachments are described in UFSAR Section 5.1.2.

STEEL LINER PLATES

The interior of each Containment is lined with steel plates that are welded together. The liner plate covers the dome and cylinder walls and runs between the floor and the foundation slab to form an essentially leaktight barrier. The liners help assure leak tightness of the Containments. The liner plate for the floors is placed on top of the foundation concrete pour and is covered with an additional concrete floor cover. The liner plate for the walls is welded to an angle grid system embedded in concrete. The grid system is designed to allow for distortion during accident conditions without compromising the essentially leaktight barrier of the Containments. The liner plates are thickened at attachments such as the polar crane brackets to reduce predicted stress level in the plane of the liner plate. The containment steel liner plates are described in UFSAR Section 5.1.2.

The external surface of the liner plates, except for the floor liners, is coated on the inside with inorganic zinc primer and painted. The application of protective coatings on the liner surfaces ensures that the external metal surface is not in contact with a moist environment for extended periods of time. Although coated, no credit is taken for protective coatings applied to the liner in the determination of aging effects.

Coatings qualified for use in the Turkey Point Units 3 and 4 Containments are adequate to resist exposures due to both normal operating and design basis accident conditions. These exposures include ionizing radiation, high temperature and pressure, impingement from jets or sprays, and abrasion due to traffic. Generic Letter 98-04, “Potential for Degradation of the Emergency Core Cooling System and The Containment Spray System after a Loss-Of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,” was issued to alert licensees to the problems associated with the material condition of protective coatings inside containments. The generic letter was issued to request information to evaluate plant programs for ensuring coatings inside

containments do not detach from their substrate and interfere with operation of accident mitigation systems. Turkey Point's response to Generic Letter 98-04 [Reference 2.4-2] indicated that coating logs are maintained and documented in controlled calculations. The logs are reviewed and updated after each refueling outage. In addition, an assessment of the overall condition of coatings is performed prior to unit restart after each refueling outage to ensure that coatings will have no effect on operation of accident mitigation systems.

CATHODIC PROTECTION

Structural steel components such as the tendon trumplates, reinforcing bars, and liner plates are interconnected to form an electrically continuous cathodic structure. Cathodic protection is provided to make these steel components electrically negative with respect to the soil to prevent galvanic corrosion. Cathodic protection is a non-safety related design feature that does not perform intended functions as defined by 10 CFR 54.4(a). Additionally, no credit is taken for the cathodic protection system in the determination of aging effects. Accordingly, the cathodic protection system is not within the scope of license renewal.

FUEL TRANSFER TUBES

The fuel transfer tube for each Containment penetrates the Containment to link the refueling canal inside the Containment and the spent fuel pool in the auxiliary building. The fuel transfer tubes serve as the underwater pathways for moving the fuel assemblies into and out of the Containments for refueling operations during plant shutdown. As part of the containment pressure boundary, the fuel transfer tubes must assure the essentially leaktight barrier function of the Containment.

During normal operation, blind flanges are installed on the fuel transfer tube penetrations and serve as part of the Containment essentially leaktight barrier. The fuel transfer tubes are described in UFSAR Sections 6.6.2.1 and 6.6.3.

PENETRATIONS

All penetrations are designed to maintain an essentially leaktight barrier to prevent the uncontrolled release of radioactivity. In addition to supporting the essentially leaktight barrier function, each penetration performs service-related functions. Penetrations may also serve as support points for systems such as piping passing through the Containment boundary.

PENETRATION ASSEMBLIES (MECHANICAL)

Mechanical penetrations provide the means for passage of process piping transmitting liquids or gases across the Containment boundary. The piping and ventilation penetrations are of the rigid welded type and are solidly anchored to the containment wall, thus precluding any requirement for expansion bellows. Mechanical penetrations are described in UFSAR Section 5.1.5.2.

PENETRATION ASSEMBLIES (ELECTRICAL)

Electrical penetrations provide the means for electrical and instrumentation conductors to cross the Containment boundary while maintaining the essentially leaktight barrier. Electrical penetrations consist of carbon steel pipe canisters with stainless steel or carbon steel header plates welded to each other. Each canister affords a double barrier against leakage. The non-pressure boundary (non-metallic) portions of electrical penetrations are addressed in Section 2.5. Electrical penetrations are described in UFSAR Section 5.1.5.3.

EQUIPMENT HATCHES

The equipment hatch on each Containment is a large flanged penetration that provides access to the containment interior at the mezzanine level. A double gasketed dished head made of steel plate seals the opening. This head is bolted to the liner with 48 one-inch diameter bolts. A double O-ring seal, with the O-rings in grooves in the head flange, makes up the final seal. Leaktightness of the seals can be checked from outside Containment by pressurizing the annular space between the two O-rings. The equipment hatches are described in UFSAR Section 5.1.5.1.

CONTAINMENT PERSONNEL HATCHES

The containment personnel hatch on each Containment is a cylindrical tube that passes through the concrete wall of the Containment and is welded to the steel liner. The cylinder has a door at each end, mechanically interlocked so that one door cannot be opened unless the other is closed. A mechanical interlock defeat will permit both doors to be open at the same time. The doors also have interlocks to prevent opening a door unless the differential pressure is essentially zero and to prevent concurrently opening the equalizing valves on both doors.

The doors are pressure seating type, opening towards the inside of the Containment. Each door is provided with double gaskets. The door seal is made of

two O-rings installed in machined grooves on the bulkhead face. The machined surface of the doorplate seals against the O-rings when the door is locked. The door locks pull the door to a metal-to-metal contact on either side of the seals, thereby compressing the O-rings. The containment personnel hatches are described in UFSAR Section 5.1.5.1.

CONTAINMENT PERSONNEL ESCAPE HATCHES

The containment personnel escape hatch on each Containment is a cylindrical tube that passes through the concrete wall of the Containment and is welded to the liner. The tube has a circular door opening at each end. Each door is provided with double gaskets. The door seal is made of two O-rings installed in machined grooves on the bulkhead face. The machined surface of the doorplate seals against the O-rings when the door is locked. The door locks pull the door to a metal-to-metal contact on either side of the seal, thereby compressing the O-rings. The containment personnel escape hatches are described in UFSAR Section 5.1.5.1.

2.4.1.1.3 POST-TENSIONING SYSTEMS

The post-tensioning system for each Containment consists of numerous tendons placed around the containment walls, both vertically and horizontally, and over the containment dome. The tendons are enclosed in a sheathing system that consists of spirally wound sheet metal tubing that acts as housing for the tendons. The tendons are installed in the sheathing system, then tensioned to the prestress values required for containment integrity.

Each containment cylinder wall is prestressed by 180 vertical tendons, anchored at the top surface of the ring girder and at the bottom of the foundation slab, and 489 hoop tendons, each enclosing 120° of arc anchored in six vertical buttresses. Each dome is prestressed by three groups of 55 tendons, oriented at 120° to each other, for a total of 165 tendons anchored at the vertical face of the dome ring girder. Each tendon consists of 90 wires bundled together with buttonheaded tendon wires (Birkenmeier Brandestinin Ros Vogt or BBRV system) anchorage. The containment post tensioning system is described in UFSAR Sections 5.1.2 and 5.1.6.

2.4.1.2 CONTAINMENT INTERNAL STRUCTURAL COMPONENTS

The Containment internal structural components for each Containment consist mainly of the reactor primary shield wall, the lower secondary compartments, the

upper secondary compartments, the refueling cavity, and system/component supports.

2.4.1.2.1 CONCRETE

The primary shield walls are thick cylindrical walls that enclose the reactor vessels and provide biological shielding and structural support. The primary shield walls also act as part of the missile barrier. The lower secondary compartments for each Containment enclose the reactor coolant loops and consist of the secondary shield walls that support the intermediate floor. The upper secondary compartments for each Containment consist of four compartments. Three of these compartments enclose one reactor coolant loop each and another encloses the pressurizer. The compartment walls provide secondary biological shielding and structural support for the operating floor.

The refueling cavity/refueling canal for each Containment is a stainless steel lined reinforced concrete structure that forms a pool, above the reactor, when it is filled with borated water for refueling. It is irregularly shaped, formed by the upper portions of the primary shield concrete and other sidewalls of varying thicknesses and contains space for storing the upper and lower reactor internals packages and miscellaneous refueling tools.

Barriers surround all high-pressure equipment, i.e., high-energy reactor coolant system piping and components, which could generate missiles as a result of a design basis accident. These barriers, principally the primary and secondary shield walls, prevent such missiles from damaging the containment liner, piping penetrations, and the required engineered safeguards systems. A removable reinforced concrete shield, located above each reactor vessel head, provides missile protection for any missile that could be generated by the control rod drive mechanisms.

Concrete walls, floors, beams, equipment pads, and other miscellaneous concrete components are of conventional design using intermediate grade reinforcing steel. The concrete Containment internal structural components are described in UFSAR Section 5.1.2.

2.4.1.2.2 STEEL

POLAR CRANES

The reactor polar cranes and associated rails are seismically qualified Class I structures in the unloaded configuration. The cranes provide a means for lifting and handling heavy loads inside the Containment structures. The polar crane is described in UFSAR Sections 5A-1.2.8, 5E-2.8, and 5I-3.7.

SPENT FUEL STORAGE AND HANDLING

Spent fuel handling equipment located inside each Containment includes the reactor cavity seal ring, the manipulator crane, and portions of the fuel transfer system. The spent fuel handling equipment inside Containment is evaluated with the remainder of the Spent Fuel Storage and Handling System in Subsection 2.4.2. The spent fuel handling equipment is described in UFSAR Section 9.5.

CONTAINMENT SUMPS

For each Containment, there are two containment recirculation sumps, each with a line that leads from the containment sump to the suction of the residual heat removal pumps. Following a postulated loss-of-coolant accident, the pumps take suction on the sumps and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat removal heat exchangers.

Filtration of the water entering the residual heat removal pump suction piping is accomplished by screens located over the sumps. The containment recirculation sumps are described in UFSAR Sections 6.2.1 and 6.4.2.

REACTOR COOLANT SYSTEM SUPPORTS

Reactor Coolant System supports that are subject to aging management review include the reactor vessel supports, steam generator supports, pressurizer supports, and reactor coolant pump supports. The Reactor Coolant System supports are designed to resist operating loads, pipe rupture loads, and seismic loads.

The NRC issued a draft safety evaluation for Westinghouse Owners Group generic technical report WCAP-14422, “License Renewal Evaluation: Aging Management for Reactor Coolant System Supports,” Revision 2 [Reference 2.4-3], on February 25, 2000 [Reference 2.4-4].

Turkey Point reviewed the current design and operation of the Reactor Coolant System piping using the process described in Subsection 2.3.1.1.1 and confirmed that the operating environments used in the design of the Turkey Point Reactor Coolant System supports are consistent with the description contained in WCAP-14422. The Reactor Coolant System supports contain materials beyond those identified in Table 2-4 of WCAP-14422. These additional materials were included in the Reactor Coolant System supports aging management review described in Subsection 3.6.1.5. The component intended functions for the Turkey Point Reactor Coolant System supports are consistent with the intended functions identified in WCAP-14422.

As a result of the NRC review of WCAP-14422, several open items and license renewal applicant action items were identified. These open items and applicant action items are described in the draft NRC safety evaluation of WCAP-14422. The Turkey Point-specific responses to those open items and applicant action items relevant to the identification of Reactor Coolant System supports subject to aging management review are provided in Tables 2.4-1 and 2.4-2. Major component supports are described in UFSAR Section 5.1.9.

In addition to the Reactor Coolant System supports addressed by WCAP-14422, other supports are provided for various systems and components inside Containment. These supports are addressed as steel commodities below.

STEEL COMMODITIES

Structural and miscellaneous steel are provided in each Containment structure to allow access to the various elevations and areas inside Containment for inspection and maintenance. The steel also provides support for safety-related and non-safety related systems and components, including: piping, ducts, miscellaneous equipment, electrical cable tray and conduit, instruments and tubing, electrical and instrumentation enclosures and racks, steel beams and columns, stairways, ladders, and attachments to the concrete walls and liner. Containment internal structural steel is discussed in UFSAR Section 5.1.

Similar to the liner plates discussed in Subsection 2.4.1.1, the external surfaces of steel are coated. The application of protective coatings on steel surfaces ensures that the external metal surface is not in contact with a moist environment for extended periods of time. Although coated, no credit is taken for protective coatings in the determination of aging effects.

2.4.1.3 CONCLUSION

The Containments are in the scope of license renewal because they:

- Provide pressure boundary and/or fission product barrier
- Provide structural support to safety-related components
- Provide shelter/protection to safety-related components (including radiation shielding)
- Provide fire barriers to retard spreading of a fire
- Provide missile barriers
- Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provide flood protection barriers
- Provide filtration of process fluid to protect downstream equipment
- Provide structural support and shelter to components relied on during certain postulated fire, anticipated transients without scram, and station blackout events

A complete list of Containment structural components requiring an aging management review and the component intended functions are provided in Table 3.6-2. The aging management review of the Containments is discussed in Subsection 3.6.1. Note that only the fuel transfer tube blind flanges are included with the Containment aging management review. The aging management review of the fuel transfer tubes, penetration sleeves, and gate valves is discussed in Subsection 3.6.2 as part of Spent Fuel Storage and Handling.

2.4.2 OTHER STRUCTURES

The following structures are included in this subsection:

- Auxiliary Building
- Cold Chemistry Lab
- Control Building
- Cooling Water Canals
- Diesel Driven Fire Pump Enclosure
- Discharge Structure
- Electrical Penetration Rooms
- Emergency Diesel Generator Buildings
- Fire Protection Monitoring Station
- Fire Rated Assemblies
- Intake Structure
- Main Steam and Feedwater Platforms
- Plant Vent Stack
- Spent Fuel Storage and Handling
- Turbine Building
- Turbine Gantry Cranes
- Turkey Point Units 1 and 2 Chimneys
- Yard Structures

2.4.2.1 AUXILIARY BUILDING

The Auxiliary Building is a reinforced concrete structure that houses safety-related systems, structures, and components. Failure of the Auxiliary Building, or certain portions thereof, to adequately resist the applicable design loads could result in adverse interaction with SSCs important to nuclear safety. The fuel handling building structure (including the concrete spent fuel pool and the concrete sliding door) is part of the Auxiliary Building.

The building is constructed on a foundation mat with concrete bearing walls and slabs. Earthquake, wind, and other appropriate lateral loads are resisted by diaphragm action of the walls and slabs. Ductile behavior of all the walls and slabs is maintained for better resistance of dynamic loads. The Auxiliary Building is shown in Figure 2.2-1 and is described in UFSAR Section 5.2.

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

- Provides structural support to safety-related components
- Provides shelter/protection to safety-related components (including radiation shielding)
- Provides rated fire barriers to retard spreading of a fire
- Provides missile barriers
- Provides structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provides flood protection barriers
- Provides structural support and shelter to components relied on during certain postulated fire and station blackout events
- Provide pipe whip restraint and/or jet impingement protection

A complete list of Auxiliary Building structural components requiring an aging management review and the component intended functions are provided in Table 3.6-3. The aging management review for the Auxiliary Building is discussed in Subsection 3.6.2.

2.4.2.2 COLD CHEMISTRY LAB

The Cold Chemistry Lab is a concrete building with a concrete roof. It is located southwest of the turbine building. The Cold Chemistry Lab does not perform any safety-related functions, or directly protect safety-related equipment. The Cold Chemistry Lab is shown in Figure 2.2-1.

The Cold Chemistry Lab is in the scope of license renewal because:

- It is a non-safety related structure whose failure could prevent satisfactory accomplishment of required safety-related functions

A complete list of Cold Chemistry Lab structural components requiring an aging management review and the component intended functions are provided in Table 3.6-4. The aging management review for the Cold Chemistry Lab is discussed in Subsection 3.6.2.

2.4.2.3 CONTROL BUILDING

The Control Building is a three-story reinforced concrete structure housing safety related systems, structures, and components. The Control Building walls and roof are designed to withstand missile effects. The Control Building is shown in Figure 2.2-1 and is described in UFSAR Section 5.3-1.

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

- Provides structural support to safety-related components
- Provides shelter/protection to safety-related components (including radiation shielding)
- Provides rated fire barriers to retard spreading of a fire
- Provides missile barriers
- Provides structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provides structural support and shelter to components relied on during certain postulated fire, anticipated transients without scram, and station blackout events

A complete list of Control Building structural components requiring an aging management review and the component intended functions are provided in Table 3.6-5. The aging management review for the Control Building is discussed in Subsection 3.6.2.

2.4.2.4 COOLING WATER CANALS

The Cooling Water Canals serve as the plant ultimate heat sink. The Cooling Water Canals constitute a closed cooling system made up of earthen canals that provide cooling of discharged water prior to reuse at the intake structure. The Cooling Water Canals are shown in Figure 3.1-2 of the License Renewal Application Environmental Report.

The Cooling Water Canals are in the scope of license renewal because they:

- Provide a source of cooling water for plant shutdown

A complete list of Cooling Water Canals structural components requiring an aging management review and the component intended functions are provided in Table 3.6-6. The aging management review for the Cooling Water Canals is discussed in Subsection 3.6.2.

2.4.2.5 DIESEL DRIVEN FIRE PUMP ENCLOSURE

The Diesel Driven Fire Pump is protected from the external environment by a prefabricated enclosure. The enclosure is designed in accordance with the South Florida Building Code. The structure is anchor bolted to a reinforced concrete foundation. Access is provided through double doors at both ends of the building. The Diesel Driven Fire Pump Enclosure is shown in Figure 2.2-1.

The Diesel Driven Fire Pump Enclosure, although not specifically credited for fire protection, has been conservatively included in the scope of license renewal because it:

- Provides shelter to components relied on during certain postulated fire events

A complete list of Diesel Driven Fire Pump Enclosure structural components requiring an aging management review and the component intended functions are provided in Table 3.6-7. The aging management review for the Diesel Driven Fire Pump Enclosure is discussed in Subsection 3.6.2.

2.4.2.6 DISCHARGE STRUCTURE

Engineering features located along the west edge of the plant secured area are collectively referred to as the Discharge Structure. The primary purpose of the Discharge Structure is to provide for the emission of effluent from circulating water, intake cooling water, screen wash, and storm drains into the cooling water canals.

The Unit 3 Discharge Structure includes a concrete seal well, north concrete headwall, south concrete headwall, and associated steel framing and platforms. The seal well introduces flow from the buried circulating water piping into the cooling water canals. The concrete seal well also provides a base on which structural steel framing and platforms are supported. The north headwall introduces flow from the safety-related intake cooling water pipe and non-safety related screen refuse and

storm drain pipes. The south headwall introduces flow from the non-safety related intake cooling water pipe.

The Unit 4 Discharge Structure includes a concrete seal well, south concrete headwall, and associated steel framing and platforms. The seal well introduces flow from the buried circulating water piping into the cooling water canals. The concrete seal well also provides a base on which structural steel framing and platforms are supported. The south headwall introduces flow from the safety-related intake cooling water pipe and the non-safety-related intake cooling water and storm drainpipes. Unit 4 does not require a north headwall since the screen refuse pipe is common to both Units and is part of the Unit 3 north concrete headwall.

The primary function of the headwalls is to protect the embankment from currents introduced by the discharge water. While the Discharge Structure performs no nuclear safety-related function, the safety-related intake cooling water piping penetrates the concrete headwall. Failure of the intake cooling water pipe concrete headwall could jeopardize the safety function of the Intake Cooling Water System. The Units 3 and 4 Discharge Structure is shown in Figure 2.2-1.

The Discharge Structure is in the scope of license renewal because:

- It is a non-safety related structure whose failure could prevent satisfactory accomplishment of required safety-related functions

A complete list of Discharge Structure structural components requiring an aging management review and the component intended functions are provided in Table 3.6-8. The aging management review for the Discharge Structure is discussed in Subsection 3.6.2.

2.4.2.7 ELECTRICAL PENETRATION ROOMS

Each Unit has two Electrical Penetration Rooms housing safety-related structures and components. Unit 3 has a West Electrical Penetration Room and a South Electrical Penetration Room. Unit 4 has a West Electrical Penetration Room and a North Electrical Penetration Room. All four rooms are reinforced concrete enclosures that contain electrical containment penetrations and cables. The west rooms are independent structures located immediately west of each containment. The north and south rooms are integral with the auxiliary building and are located at the western most interface between the auxiliary building and the containment

buildings. The Electrical Penetration Rooms are shown in Figure 2.2-1 and are described in UFSAR Section 5E-2.2.

The Electrical Penetration Rooms are in the scope of license renewal because they:

- Provide structural support to safety-related components
- Provide shelter/protection to safety-related components (including radiation shielding)
- Provide fire barriers to retard spreading of a fire
- Provide missile barriers
- Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provide structural support and shelter to components relied on during certain postulated fire, anticipated transients without scram, and station blackout events

A complete list of Electrical Penetration Rooms structural components requiring an aging management review and the component intended functions are provided in Table 3.6-9. The aging management review for the Electrical Penetration Rooms is discussed in Subsection 3.6.2.

2.4.2.8 EMERGENCY DIESEL GENERATOR BUILDINGS

The original emergency on-site AC power source for Turkey Point Units 3 and 4 consisted of two emergency diesel generators. The two original emergency diesel generators are presently identified as 3A and 3B, and are housed in the Unit 3 Emergency Diesel Generator Building. In 1990 and 1991, two additional emergency diesel generator units, labeled 4A and 4B, were added to the emergency power system. The Unit 4 Emergency Diesel Generator Building was designed and constructed to house the additional units. The Emergency Diesel Generator Buildings are shown in Figure 2.2-1 and are described in UFSAR Sections 5.3.2 (Unit 3) and 5.3.4 (Unit 4).

Both the Unit 3 and Unit 4 Emergency Diesel Generator Buildings are reinforced concrete structures housing safety-related systems, structures, and components. The first floor of each building is divided into two bays, each bay containing one of the two engine-generator sets housed in the building. The Emergency Diesel

Generator Buildings also house components of the emergency diesel generator subsystems, such as the fuel oil, starting air, lubricating oil, combustion air, and exhaust air equipment.

The Emergency Diesel Generator Buildings are in the scope of license renewal because they:

- Provide structural support to safety-related components
- Provide shelter/protection to safety-related components (including radiation shielding)
- Provide fire barriers to retard spreading of a fire
- Provide missile barriers
- Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provide flood protection barriers
- Provide structural support and shelter to components relied on during certain postulated fire, anticipated transients without scram, and station blackout events

A complete list of Emergency Diesel Generator Buildings structural components requiring an aging management review and the component intended functions are provided in Table 3.6-10. The aging management review for the Emergency Diesel Generator Buildings is discussed in Subsection 3.6.2.

2.4.2.9 FIRE PROTECTION MONITORING STATION

The Fire Protection Monitoring Station is a reinforced concrete and concrete block structure located adjacent to the west wall of the control building. The Fire Protection Monitoring Station contains numerous video screens used to monitor various areas throughout the plant for fire detection as a compensatory measure pending resolution of corrective actions related to Thermolag. The Fire Protection Monitoring Station is shown in Figure 2.2-1.

The Fire Protection Monitoring Station, although not specifically credited for fire protection, has been conservatively included in the scope of license renewal because it:

- Provides structural support and shelter to components relied on during certain postulated fire events

A complete list of Fire Protection Monitoring Station structural components requiring an aging management review and the component intended functions are provided in Table 3.6-11. The aging management review for the Fire Protection Monitoring Station is discussed in Subsection 3.6.2.

2.4.2.10 FIRE RATED ASSEMBLIES

Fire Rated Assemblies include the following: fire barriers, fire doors, fire dampers, penetration seals, and electrical conduit seals. The Fire Rated Assemblies are described in UFSAR Appendix 9.6A, Section 3.11.

Fire barriers are provided to ensure that the function of one train of redundant equipment necessary to achieve and maintain hot standby and cold shutdown conditions remains free of fire damage. Fire barriers provide a means of limiting fire travel by compartmentalization and containment. Turkey Point fire barriers include walls, floors, and ceilings; raceway protection; structural steel fireproofing; manhole covers and hatches; and radiant energy shields. Thermolag barriers, raceway protection, and structural steel fireproofing were evaluated with Fire Rated Assemblies. Concrete walls, floors, and ceilings were evaluated with the specific structure in which they reside; manhole covers were evaluated with Yard Structures; and radiant energy shields (located inside containment) were evaluated with the Containments.

Fire door assemblies prevent the spread of fire through passageways and fire barriers. Fire door assemblies protect openings in walls and partitions against the spread of fire.

Fire dampers are provided to prevent the spread of fire through ventilation penetrations. Fire dampers were evaluated with the Fire Protection System in Subsection 2.3.3.14.

Penetration seals are provided to maintain the integrity of fire barriers at barrier penetrations. Those penetrations through fire barriers that are not restored by grout or concrete are sealed as follows. Mechanical, electrical, and structural steel penetrations are sealed with solid silicone elastomer, boot seals, high-density self-supporting gel seals, prefabricated fire seals, or hydrosil material seals.

Electrical conduit seals are used to protect open-ended conduit from fixed water suppression spray, to keep Halon from escaping the area protected by Halon suppression, and to limit flame propagation. A conduit seal is provided for an open-ended conduit if the configuration of the conduit is such that water can be conducted into equipment containing electrical terminations, and for open-ended conduits in fire areas protected by Halon suppression if the conduit penetrates a boundary of the fire area. Conduit seals are also provided for conduits penetrating fire barriers as appropriate.

The Fire Rated Assemblies are in the scope of license renewal because they:

- Provide shelter/protection to safety-related components (control room fire doors are a portion of the control room environmental envelope)
- Provide fire barriers to retard spreading of a fire

A complete list of Fire Rated Assemblies structural components requiring an aging management review and the component intended functions are provided in Table 3.6-12. The aging management review for the Fire Rated Assemblies is discussed in Subsection 3.6.2.

2.4.2.11 INTAKE STRUCTURE

The Intake Structure is a reinforced concrete and steel structure consisting of eight intake channels (bays). The Intake Structure supports the six safety-related intake cooling water pumps, the eight non-safety related circulating water pumps, and the three non-safety related screen wash pumps. These pumps take suction from the intake channels and supply water to Turkey Point Units 3 and 4.

At the inlet to each channel, a stationary (grizzly) screen collects large debris to prevent damage to the traveling screens. There are eight traveling screens, one for each intake channel, located just downstream of the grizzly screens. The traveling screens remove small debris from the intake water, thus preventing debris from reaching the suction of the pumps. The Intake Structure is shown in Figure 2.2-1 and is described in UFSAR Section 5.3.2.

The Intake Structure is in the scope of license renewal because it:

- Provides structural support to safety-related components
- Provides shelter/protection to safety-related components

- Provides a source of cooling water for plant shutdown
- Provides structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provides flood protection barriers
- Provides filtration of process fluid so that downstream equipment is protected
- Provides structural support to components relied on during certain postulated fire and station blackout events

A complete list of Intake Structure structural components requiring an aging management review and the component intended functions are provided in Table 3.6-13. The aging management review for the Intake Structure is discussed in Subsection 3.6.2.

2.4.2.12 MAIN STEAM AND FEEDWATER PLATFORMS

The Main Steam and Feedwater Platforms are steel and concrete structures, located just outside containment, that contain safety-related SCs from the main steam, feedwater, and auxiliary feedwater systems. The Main Steam Platforms are located directly west of the Unit 3 and 4 containment buildings. The Feedwater Platforms are located northwest of the Unit 3 containment and southwest of the Unit 4 containment. The Main Steam and Feedwater Platforms are shown in Figure 2.2-1.

The Main Steam and Feedwater Platforms are in the scope of license renewal because they:

- Provide structural support to safety-related components.
- Provide shelter/protection to safety-related components.
- Provide missile barriers.
- Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions.
- Provide structural support and shelter to components relied on during certain postulated fire, anticipated transients without scram, and station blackout events.
- Provide pipe whip restraint and jet impingement protection.

A complete list of Main Steam and Feedwater Platforms structural components requiring an aging management review and the component intended functions are provided in Table 3.6-14. The aging management review for the Main Steam and Feedwater Platforms is discussed in Subsection 3.6.2.

2.4.2.13 PLANT VENT STACK

The Plant Vent Stack is a steel tubular structure used for releasing processed gases to the atmosphere. The Plant Vent Stack is supported at its base by the auxiliary building roof and laterally restrained near its top by the Unit 4 containment structure. Structural failure of the Plant Vent Stack could impact safety-related equipment. The Plant Vent Stack is shown in Figure 2.2-1.

The Plant Vent Stack is in the scope of license renewal because:

- It is a non-safety related structure whose failure could prevent satisfactory accomplishment of required safety-related functions

A complete list of Plant Vent Stack structural components requiring an aging management review and the component intended functions are provided in Table 3.6-15. The aging management review for the Plant Vent Stack is discussed in Subsection 3.6.2.

2.4.2.14 SPENT FUEL STORAGE AND HANDLING

Spent Fuel Handling includes all the equipment and tools necessary to remove spent fuel from the reactor vessels, transport spent fuel to the spent fuel pools, place spent fuel in the appropriate storage rack cell, and remove spent fuel from the spent fuel storage pools for alternative storage. The major equipment required for Spent Fuel Handling includes: the reactor cavity seal rings, the manipulator cranes, the fuel transfer system (located in the refueling canal inside containment and in the fuel transfer canal in the spent fuel building), the fuel transfer tubes, penetration sleeves, and gate valves, the spent fuel bridge cranes, the fuel handling tools, and the spent fuel cask crane.

Spent Fuel Storage includes all the structural components necessary to store spent fuel in the spent fuel storage pools, excluding the concrete structure. The major structural items required for Spent Fuel Storage are the spent fuel pit liners, the keyway gates, and the spent fuel storage racks. The concrete fuel handling building (including the spent fuel pool and the concrete sliding doors) is part of the auxiliary

building and is screened with the rest of the auxiliary building structure in Subsection 2.4.2.1.

The spent fuel storage pools are designed for the underwater storage of spent fuel assemblies and control rods after removal from the reactor. The spent fuel pits are lined on their interior surfaces with a stainless steel liner. Stainless steel storage racks rest on the pool floor and hold the spent fuel assemblies. Fuel assemblies are placed and held in vertical cells of a rectangular high-density array. The racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby ensuring the necessary spacing between assemblies. The high-density stainless steel storage racks have Boraflex inserts, a proprietary neutron absorbing material. Note: Turkey Point has determined that Boraflex is not a Time-Limited Aging Analysis, however, Boraflex is addressed in Subsection 3.6.2 and the Boraflex Surveillance Program, Appendix B Subsection 3.2.2.

Spent Fuel Storage and Handling System is described in UFSAR Sections 5.2.4 and 9.5.

Spent Fuel Storage and Handling is in the scope of license renewal because it:

- Provides pressure boundary
- Provides structural support to safety-related components
- Provides shelter/protection to safety-related components (including radiation shielding)
- Provides rated fire barriers to retard spreading of a fire
- Provides missile barriers

A complete list of Spent Fuel Storage and Handling structural components requiring an aging management review and the component intended functions are provided in Table 3.6-16. The aging management review for Spent Fuel Storage and Handling is discussed in Subsection 3.6.2.

2.4.2.15 TURBINE BUILDING

The Turbine Building is a reinforced concrete and steel structure. It is primarily an open steel frame built on reinforced concrete mat foundations. The reinforced concrete turbine pedestals are the dominant structural features of the Turbine Building. The building is essentially rectangular in shape with the long north/south axis sharing the Unit 3 and 4 turbine centerline orientation. The Turbine Building is

located just west of the Unit 3 and Unit 4 Containments. The ground floor is surrounded by a floodwall to protect turbine building equipment. The Turbine Building is shown in Figure 2.2-1.

The Turbine Building houses the following Unit 3 and 4 safety-related equipment and structures: 4160V switchgear, 480V load centers, and associated concrete enclosures; the steam generator feedwater pump discharge valves and associated blockwall enclosures; and the 3A and 4A motor control centers and associated steel enclosures. In addition, the following miscellaneous safety-related equipment is included in the Turbine Building: the auxiliary feedwater supply lines from the condensate storage tanks and numerous conduits and cable trays.

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

- Provides structural support to safety-related components
- Provides shelter/protection to safety-related components
- Provides rated fire barriers to retard spreading of a fire
- Provides missile barriers
- Provides structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provides flood protection barriers
- Provides structural support and shelter to components relied on during certain postulated fire, anticipated transients without scram, and station blackout events

A complete list of Turbine Building structural components requiring an aging management review and the component intended functions are provided in Table 3.6-17. The aging management review for the Turbine Building is discussed in Subsection 3.6.2.

2.4.2.16 TURBINE GANTRY CRANES

The Turkey Point Fossil Units 1 and 2 Turbine Gantry Crane has a rated capacity of 70/15 tons. The Turkey Point Nuclear Units 3 and 4 Turbine Gantry Crane has a rated capacity of 145/35 tons. The two Turbine Gantry Cranes share rails common to all four units.

The Units 1 and 2 Turbine Gantry Crane is used almost exclusively on Units 1 and 2, and the Units 3 and 4 Turbine Gantry Crane is used almost exclusively on Units 3 and 4. Although infrequent, when the Units 1 and 2 Turbine Gantry Crane is used on Units 3 and 4, an evaluation is performed to ensure conformance with NUREG-0612. The Turbine Gantry Cranes are described in UFSAR Section 5I.3.

The Turbine Gantry Cranes are in the scope of license renewal because:

- They are non-safety related structures whose failure could prevent satisfactory accomplishment of required safety-related functions

A complete list of Turbine Gantry Cranes structural components requiring an aging management review and the component intended functions are provided in Table 3.6-18. The aging management review for the Turbine Gantry Cranes is discussed in Subsection 3.6.2.

2.4.2.17 TURKEY POINT UNITS 1 AND 2 CHIMNEYS

The Turkey Point Units 1 and 2 Chimneys, located directly north of Unit 3, do not perform any safety-related functions or directly protect safety-related equipment. However, these structures have been designed to not fail and cause an adverse interaction with any safety-related systems when subjected to the Class I seismic and wind loads. The Turkey Point Units 1 and 2 Chimneys are described in UFSAR Section 5A-1.4.2.

The Turkey Point Units 1 and 2 Chimneys are in the scope of license renewal because:

- They are non-safety related structures whose failure could prevent satisfactory accomplishment of required safety-related functions

A complete list of Turkey Point Units 1 and 2 Chimneys structural components requiring an aging management review and the component intended functions are provided in Table 3.6-19. The aging management review for the Turkey Point Units 1 and 2 Chimneys is discussed in Subsection 3.6.2.

2.4.2.18 YARD STRUCTURES

Yard Structures include concrete foundations for miscellaneous in-scope equipment and structures, concrete trenches for in-scope piping and utilities, and concrete duct banks and manholes for in-scope electrical systems that are not included within an

existing in-scope structure. Steel support structures (e.g., yard pipe supports) associated with the above described concrete structures were evaluated with the associated system. Yard Structures are shown in Figure 2.2-2.

The Yard Structures are in the scope of license renewal because they:

- Provide structural support to safety-related components
- Provide shelter/protection to safety-related components
- Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of required safety-related functions
- Provide structural support to components relied on during certain postulated fire, anticipated transients without scram, and station blackout events
- Provide pipe whip restraint and/or jet impingement protection

A complete list of Yard Structures structural components requiring an aging management review and the component intended functions are provided in Table 3.6-20. The aging management review for Yard Structures is discussed in Subsection 3.6.2.

2.4.3 REFERENCES

2.4-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 1, Nuclear Energy Institute, January 2000.

2.4-2 R. J. Hovey (FPL) letter to U. S. Nuclear Regulatory Commission, “Response to Generic Letter 98-04, Potential for Degradation of Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," L-98-272, November 9, 1998.

2.4-3 WCAP-14422, "License Renewal Evaluation: Aging Management for Reactor Coolant System Supports," Revision 2, March 1997.

2.4-4 C. I. Grimes (NRC) letter to R. A. Newton (WOG), "Draft Safety Evaluation Concerning the Westinghouse Owners Group License Renewal Evaluation: Aging Management for Reactor Coolant System Supports, WCAP-14422, Revision 2," February 25, 2000.

TABLE 2.4-1
REACTOR COOLANT SYSTEM SUPPORTS APPLICANT ACTION ITEMS FROM SECTION 4.1 OF WCAP-14422 DRAFT SAFETY EVALUATION

Renewal Applicant Action Item	Turkey Point-Specific Response
(1) The Westinghouse Owners Group did not clearly define the term "local" in its report. However, the aging management programs should be the same for all concrete structures and structural components, therefore, the license renewal applicants must describe the aging management program for adjacent concrete structures and any differences from the aging management program for the local concrete structures.	All concrete located inside Containment is addressed in the Containment aging management review with no distinction made between the local and adjacent concrete.
(2) A license renewal applicant will have to justify any differences between its Reactor Coolant System support system and the figures and descriptions of the supports systems contained in the Westinghouse Owners Group report.	The Turkey Point Reactor Coolant System support configurations are consistent with the configurations described in the generic technical report.
(3) A license renewal applicant will have to justify any differences between the materials used for its Reactor Coolant System supports and the values listed in Table 2-4 of the Westinghouse Owners Group report.	The materials used for the Turkey Point Reactor Coolant System supports are bounded by the materials specified in Table 2.4 of the generic technical report. Specifically, the materials of concern in the draft NRC Safety Evaluation, A7 and A36, are not used in any of the Turkey Point Reactor Coolant System supports.
(4) Recommendation from Section 5 of the Westinghouse Owners Group report:	
- Identification and evaluation of any plant-specific Time-Limited Aging Analyses applicable to their Reactor Coolant System supports.	- Plant-specific Time-Limited Aging Analyses are addressed in Chapter 4.
- Identification and evaluation of current-term programs implemented within the current licensing term to address technical issues from industry practices and United States Nuclear Regulatory Commission (NRC) directives [that] should be continued into the license renewal term. Modifications to or elimination of these programs have to be justified.	- Current term programs do not deviate from the recommended aging management programs. None of the current term programs will be modified or eliminated for license renewal.

TABLE 2.4-1 (continued)
REACTOR COOLANT SYSTEM SUPPORTS
APPLICANT ACTION ITEMS FROM SECTION 4.1 OF WCAP-14422 DRAFT SAFETY EVALUATION

Renewal Applicant Action Item	Turkey Point-Specific Response
Identification and justification of plant-specific programs that deviate from the recommended aging management programs.	- The only deviation from the recommended aging management programs identified in the generic technical report is in regard to AMP-1.3, " Stress Corrosion Cracking for Bolting." Based on the identified support bolting materials and environment, stress corrosion cracking of support bolting was eliminated as an aging effect requiring management at Turkey Point, as discussed in Section 3.6.
- Identification of any specific program necessary to ensure that proper preload is retained for the component supports within the scope of this report.	- Based on the Reactor Coolant System support materials and environment, creep and stress relaxation of bolting were eliminated as aging effects requiring management at Turkey Point, as discussed in Section 3.6.
- Identification of any evidence of aging degradation in inaccessible areas during the current licensing term that is considered to potentially affect system intended functions. A plan of action to address any identified potential degradation should be provided.	- The nozzle supports for the reactor vessel and the upper steam generator seismic ring are not currently inspected under Turkey Point's ASME Section XI inservice inspection (ISI) program. Based on results of inspections performed in 1987, reactor vessel supports showed no sign of aging. Therefore, no inspections beyond those Reactor Coolant System supports currently inspected under the ISI Program are warranted.
- Verification that the plant is bounded by this generic technical report. The actions applicants must take to verify that their plant is bounded will be described in an implementation procedure.	- Based on review of the Reactor Coolant System supports' configuration, operating environment, and materials, Turkey Point is bounded by the Reactor Coolant System Supports Generic Technical Report as described in Subsection 2.4.1.2. The actions taken to verify that the plant is bounded are described in Subsection 2.3.1.1.1. These actions are included in the aging management review implementing procedure.
Plant-specific evaluation of potential degradation due to irradiation of the components within the scope of this report.	- The potential for irradiation embrittlement of vessel supports external to the vessel was eliminated as an aging effect requiring management at Turkey Point, as discussed in Section 3.6.

TABLE 2.4-1 (continued) REACTOR COOLANT SYSTEM SUPPORTS APPLICANT ACTION ITEMS FROM SECTION 4.1 OF WCAP-14422 DRAFT SAFETY EVALUATION

Renewal Applicant Action Item	Turkey Point-Specific Response
(5) The Westinghouse Owners Group report states that concrete degradation from irradiation will be addressed by plant-specific evaluation. The staff agrees with this suggestion and the license renewal applicant must develop plant-specific program(s) to evaluate this concern.	The Turkey Point neutron fluence levels and maximum integrated gamma doses were evaluated. Based on this evaluation, concrete degradation due to irradiation was eliminated as an aging effect requiring management at Turkey Point, as discussed in Section 3.6.
(6) The attributes of the aging management programs provided in the Westinghouse Owners Group report do not address all elements as listed in Section 3.0.I.C of the standard review plan for license renewal. The applicants should address the missing review elements and describe the plant-specific experience, if any, related to aging degradation of the Reactor Coolant System supports in their applications.	All elements listed in Subsection 3.0.I.C of the standard review plan for license renewal are addressed in the Turkey Point aging management programs, as described in Appendix B.
(7) A license renewal applicant must provide the necessary details to perform leakage identification walkdowns and the details of the leakage monitoring program(s), especially the frequencies, for Aging Management Program 1-1 and Aging Management Program 1-2.	Leakage identification walkdowns and leakage monitoring for Aging Management Programs 1-1, "Aggressive Chemical Attack and Corrosion for Steel," and 1-2, "Aggressive Chemical Attack and Corrosion for Concrete Embedments," are addressed by the Turkey Point Boric Acid Wastage Surveillance Program. The details of the Boric Acid Wastage Surveillance Program are provided in Appendix B.
(8) All structures and structural components need a baseline inspection to document the condition of the structures and structural components. Therefore, the renewal applicants will have to have plant-specific baseline inspection results for all structures and structural components, or a planned inspection to obtain such results and validate the aging management programs prior to entering the period of extended operation.	Baseline inspections are used as the baseline condition against which all future inspections are compared. Baseline inspections are generally performed when there are no existing, documented inspection results. Although not characterized as “baseline inspections” at the time they were performed, inspections that serve as baseline inspections have been performed and documented for the Reactor Coolant System supports under the In-service Inspection Program.
(9) In accordance to the Westinghouse Owners Group report, leakage walkdowns and monitoring are plant-specific. Therefore, a license renewal applicant will have to provide the necessary qualitative or quantitative acceptance criteria for leakage walkdowns and monitoring.	Leakage walkdowns and monitoring are included in the Turkey Point Boric Acid Wastage Surveillance Program. The details of the Boric Acid Wastage Surveillance Program are provided in Appendix B.

TABLE 2.4-2
REACTOR COOLANT SYSTEM SUPPORTS OPEN ITEMS FROM SECTION 4.2 OF WCAP-14422 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(1) The Westinghouse Owners Group report contains many discrepancies and omissions:	
- Wear plates and bearing pads are included as support components and are within the scope of this Westinghouse Owners Group report but are not identified in Table 2-1 as parts and sub-components requiring an aging management review.	- These items should be included in the generic technical report table. The aging effect evaluation for these items is included in Section 3.2.5 of the generic technical report and is consistent with the Turkey Point aging management review discussed in Subsection 3.6.1.5.
- Sketches of Reactor Coolant Pump support configuration 4 and Pressurizer support configuration 2 are not provided in the Westinghouse Owners Group report.	- These sketches could be added to the generic technical report for clarity but have no bearing on the Turkey Point aging management review since the Turkey Point reactor coolant pump supports are consistent with configuration 5 of the generic technical report.
- Section 3.2.9 of the Westinghouse Owners Group report indicates that ASTM A36 steel is used in Steam Generator and Reactor Coolant Pump supports, however, ASTM A36 steel is not included in the list of material for the primary component supports (Table 2-4).	- ASTM A36 steel is not used in steam generator and reactor coolant pump supports at Turkey Point. The only place where A36 steel is used in Reactor Coolant System supports at Turkey Point is in the Pressurizer Surge Line support.
- The 1963 AISC manual (Ref. 3) states that the following steel materials are commonly used for steel construction but they are not listed in Table 2-4 of the Westinghouse Owners Group report. They are ASTM A7, A36, A242, A373, A440, and A441 structural steel and ASTM A325 bolts.	- The Turkey Point aging management review identifies which materials are used in Reactor Coolant System supports. ASTM A36, A440, A441, and A325 materials are used in Reactor Coolant System supports at Turkey Point and these materials were evaluated in the Turkey Point aging management review described in Subsection 3.6.1.5. Based on Turkey Point's review, consideration of these materials in the aging management review presented in the generic technical report will not affect the conclusions reached.
- There are no specific descriptions and sketches for the pressurizer surge line supports.	- The surge line in each unit has a typical deadweight spring can trapeze assembly to counter the deadweight loads. In addition, there are three (3) whip restraints per line.

TABLE 2.4-2 (continued)
REACTOR COOLANT SYSTEM SUPPORTS
OPEN ITEMS FROM SECTION 4.2 OF
WCAP-14422 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(2) Temper embrittlement and strain aging embrittlement are the most common forms of thermal embrittlement that are seen in ferritic materials as stated in Section 3.2.4 of the Westinghouse Owners Group report. The Westinghouse Owners Group report has determined that temper embrittlement is not a concern for the ferritic materials of Reactor Coolant System supports. However, the Westinghouse Owners Group report does not address the aging effects from strain aging embrittlement but states that thermal embrittlement is not applicable. Westinghouse Owners Group should discuss the applicability of the aging effects caused by strain aging embrittlement to the Reactor Coolant System support components.	The generic technical report treats 'temper embrittlement' and 'strain aging embrittlement' as one mechanism called 'thermal aging embrittlement.' The generic technical report conclusion that thermal aging embrittlement is not applicable is meant to include temper embrittlement and strain aging embrittlement. The Turkey Point aging management review is consistent with that conclusion.

TABLE 2.4-2 (continued)
REACTOR COOLANT SYSTEM SUPPORTS
OPEN ITEMS FROM SECTION 4.2 OF
WCAP-14422 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(3) Appendix C of NUREG-0577 addresses this item and groups many Westinghouse Owners Group member plants as Group I "plants requiring further evaluation." Although Table 3.9-3 of SRP-LR and Table B9 of NUREG-1557 indicated that "low fracture toughness is not significant for containment internal structures," in general, these two documents only addressed the containment internal structures as a whole and did not specifically address the Reactor Coolant System support components. Westinghouse Owners Group recognizes this concern and states in Section 3.2.9 of its report that "Utilities with potential problems were required to demonstrate that the suspect structures have adequate fracture toughness to comply with the criteria defined in NUREG-0577." However, it further states that "low fracture toughness does not cause detrimental aging effects that must be addressed by maintenance programs." The staff does not believe that the Westinghouse Owners Group report provides sufficient information to support this conclusion. Westinghouse Owners Group should confirm that its member plants listed as Group I in Appendix C of NUREG-0577 have performed the recommended evaluations in accordance with NUREG-0577 to demonstrate that the steel components of their Reactor Coolant System supports have sufficient fracture toughness to perform their intended functions.	Turkey Point is not identified as a Group 1 plant in Appendix C of NUREG-0577, Revision 1. In addition, the NUREG states that, "…a risk evaluation was performed and the results incorporated in a value-impact analysis…" and concludes, "…requirements to certify the acceptability of material or design also should not be imposed. Such action would provide no safety benefit…." Therefore, plants were not required to demonstrate that steel components of the Reactor Coolant System supports have sufficient fracture toughness to perform their intended functions.

TABLE 2.4-2 (continued)
REACTOR COOLANT SYSTEM SUPPORTS
OPEN ITEMS FROM SECTION 4.2 OF
WCAP-14422 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(4) The Westinghouse Owners Group report states that concrete operating temperature should not exceed 150°F and local area temperature should be kept under 200°F. The Westinghouse Owners Group report further states that reactor pressure vessel supports could be subjected to high temperatures that could potentially result in a local temperature above 200°F if supplemental cooling is not provided. For those support configurations where the local temperature at concrete surfaces could exceed 200°F, special design features are incorporated based on air or water cooling to keep local temperature below 200°F. These temperatures are specified in the ASME Code. Therefore, elevated temperature is not a concern for concrete. Because the operating temperature of concrete components are kept below the limits specified by the code by means of supplemented cooling, the staff considers that the aging effects of elevated temperature are applicable to the Reactor Coolant System supports and are being managed by supplemented cooling features. The Westinghouse Owners Group report should indicate that the aging effects associated with elevated temperatures are applicable and requires that applicants for license renewal demonstrate that existing design features are capable of preventing any unacceptable degradation during the extended period of operation.	Cracking due to elevated temperature is not an aging effect requiring management since concrete temperatures are below American Concrete Institute (ACI) code thresholds due to normal containment and cooling. The Normal Containment and Control Rod Drive Mechanism Cooling System is in the scope of license renewal and will be operated consistent with current operations during the period of extended operation. Failure of the Normal Containment and Control Rod Drive Mechanism Cooling System would be event related and would require plant actions to restore for long-term continued power operations.

TABLE 2.4-2 (continued)
REACTOR COOLANT SYSTEM SUPPORTS
OPEN ITEMS FROM SECTION 4.2 OF
WCAP-14422 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(5) AMP-1.2 specifies inspection frequency in accordance with the requirements of Subsection IWF-2410 (Inspection Program) and Table IWB-2412-1, each 10-year interval following the first interval, 10-year inspection program, with IWB-2412. The staff considers the frequency proposed by Westinghouse Owners Group not to be adequate. The proposed frequency is in accordance with ASME standards, but the inspections are to the requirements of ACI Standards, therefore, the frequency of inspection should also follow the recommendations of the ACI standards. Inspection frequencies recommended by ACI 349.3R-96 are every 10 years for below grade structures and controlled interiors and every 5 years for all other structures. Section 4.2.4.1 of NUREG/CR-6424 has the same recommendation for inspection frequencies. The Westinghouse Owners Group should revise the inspection frequency of AMP-1.2 to that recommended by ACI 349.3R-96.	The Turkey Point aging management review identified two aging management programs for concrete embedments. These are the Boric Acid Wastage Surveillance Program for boric acid leaks and the Systems and Structures Monitoring Program for managing general loss of material and change in material properties. Both aging management programs, described in Appendix B, meet or exceed the frequencies recommended in ACI 349.3R.

TABLE 2.4-2 (continued)
REACTOR COOLANT SYSTEM SUPPORTS
OPEN ITEMS FROM SECTION 4.2 OF
WCAP-14422 DRAFT SAFETY EVALUATION

Open Item	Turkey Point-Specific Response
(6) AMP-1.2 specifies acceptance criteria in accordance with several ACI standards. These ACI standards are ACI 201.2R-77, ACI224.1R-89, and ACI 224R-89. The staff has reviewed these ACI standards and concluded that, except for ACI 224.1R, they are mainly for design and construction rather than aging effects management because those concrete properties are built-in by design and construction. However, they do contain attributes that can be used to develop inspection acceptance criteria for AMP-1.2. For leakage walkdowns and leakage monitoring, the acceptance criteria are the same as that listed for AMP-1.1. The staff has also reviewed ACI 349.3R-96, which is referenced in the Westinghouse Owners Group report for surveillance technique, and concluded it has acceptance criteria that can be modified and used as the inspection acceptance criteria for AMP-1.2. These criteria include acceptance without further evaluation, acceptance after review, and conditions requiring further evaluation. Therefore, the staff considers that Westinghouse Owners Group, as a minimum, should provide a description of the inspection acceptance criteria similar to that of ACI 349.3R-96.	The Systems and Structures Monitoring Program acceptance criteria, described in Appendix B, incorporate the requirements of AMP-1.2.

2.5 SCOPING AND SCREENING RESULTS - ELECTRICAL AND INSTRUMENTATION AND CONTROLS (I&C)

The methodology used in identifying electrical/I&C components requiring an aging management review is discussed in Subsection 2.1.2.3. The screening for electrical/ I&C components was performed on a generic component commodity group basis for the in-scope electrical/I&C systems listed in Table 2.2-3, as well as the electrical/I&C component commodity groups associated with in-scope mechanical systems and civil structures listed in Tables 2.2-1 and 2.2-2. The methodology employed is consistent with the guidance in NEI 95-10 [Reference 2.5-1].

The interface of electrical/I&C components with other types of components and the assessments of these interfacing components are provided in the appropriate mechanical or civil/structural sections. For example, the assessment of electrical racks, panels, frames, cabinets, cable trays, conduit, and their supports is provided in the civil/structural assessment documented in Sections 2.4 and 3.6.

The electrical/I&C components included in the screening were the separate electrical/I&C components that were not parts of larger components. For example, the wiring, terminal blocks, and connections located internal to a breaker cubicle were considered to be parts of the breaker. Accordingly, the breaker was screened, but not the internal parts.

2.5.1 ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

The electrical/I&C component commodity groups were identified from a review of controlled drawings, the plant equipment database, and interface with the parallel mechanical and civil/structural screening efforts. The in-scope electrical/I&C component commodity groups identified at Turkey Point Units 3 and 4 are listed in Table 2.5-1. This list includes all electrical/I&C component commodity groups listed in Appendix B of NEI 95-10 [Reference 2.5-1], with the exception of the following component commodity groups that were eliminated from consideration based on plant level scoping:

- **Electrical Bus** - The isolated-phase buses and switchyard buses are not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).
- **Transmission Conductors** - Transmission conductors are not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).
- **High Voltage Insulators** - High voltage insulators are not relied on to meet the license renewal scoping requirements of 10 CFR 54.4(a).

No additional component commodity groups, beyond those listed in Appendix B of NEI 95-10, were identified.

2.5.2 APPLICATION OF SCREENING CRITERION 10 CFR 54.21(a)(1)(i) TO ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

Following the identification of the electrical/I&C component commodity groups, the criterion of 10 CFR 54.21(a)(1)(i) was applied to identify component commodity groups that perform their intended function passively. This evaluation was performed utilizing the guidance of 10 CFR 54.21(a)(1)(i) and NEI 95-10 [Reference 2.5-1].

The following electrical/I&C component commodity groups were determined to meet the screening criterion of 10 CFR 54.21(a)(1)(i) and were further evaluated against the criterion of 10 CFR 54.21(a)(1)(ii):

- Insulated Cables and Connections (including splices, connectors, and terminal blocks)
- Uninsulated Ground Conductors
- Electrical/I&C Penetration Assemblies

2.5.3 APPLICATION OF SCREENING CRITERION 10 CFR 54.21(a)(1)(ii) TO SPECIFIC ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

10 CFR 54.21(a)(1)(ii) allows the exclusion of those component commodity groups that are subject to replacement based on a qualified life or specified time period. The 10 CFR 54.21(a)(1)(ii) screening criterion was applied to the specific component commodity groups that were included by application of the 10 CFR 54.21(a)(1)(i) criterion. The results of this review are discussed below.

2.5.3.1 INSULATED CABLES AND CONNECTIONS

The function of insulated cables and connections is to electrically connect specified sections of an electrical circuit to deliver voltage, current, or signals. Electrical cables and their required terminations (i.e., connections) are reviewed as a single component commodity group. The types of connections included in this review are splices, connectors, and terminal blocks. Numerous insulated cables and connections are included in the Environmental Qualification Program. The insulated cables and connections that are included in this program have a qualified life that is documented in the Environmental Qualification Program. Components in the Environmental Qualification Program are replaced by the end of the qualified life. Accordingly, all insulated cables and connections within the Environmental Qualification Program are replacement items under 10 CFR 54.21(a)(1)(ii) and are not subject to an aging management review. Note that Time-Limited Aging Analyses associated with electrical/I&C components within the Environmental Qualification Program are discussed in Subsection 4.4.1.

Insulated cables and connections that perform an intended function within the scope of license renewal, but are not included in the Environmental Qualification Program, meet the criterion of 10 CFR 54.21(a)(1)(ii) and are subject to an aging management review.

2.5.3.2 UNINSULATED GROUND CONDUCTORS

Uninsulated ground conductors are electrical/I&C conductors that are uninsulated (bare) and are used to make ground connections for electrical/I&C equipment. Uninsulated ground conductors are connected to electrical/I&C equipment housings and electrical/I&C enclosures as well as metal structural features, such as the cable

tray system and building structural steel. Uninsulated ground conductors are isolated or insulated from the electrical/I&C operating circuits.

Uninsulated ground conductors are relied upon in safety analyses and plant evaluations at Turkey Point to perform a function that demonstrates compliance with the Commission's regulations for fire protection. Uninsulated ground conductors meet the criterion of 10 CFR 54.21(a)(1)(ii) and are subject to an aging management review.

2.5.3.3 ELECTRICAL/I&C PENETRATION ASSEMBLIES

Electrical/I&C penetration assemblies included in the Environmental Qualification Program have a qualified life that is documented. Therefore, electrical/I&C penetration assemblies in the Environmental Qualification Program do not meet the criterion of 10 CFR 54.21(a)(1)(ii) and are not subject to an aging management review.

A review of the electrical/I&C penetrations determined that in addition to the electrical/I&C penetration assemblies included in the Environmental Qualification Program, an additional eleven (2 power / 9 instrumentation & control) electrical/I&C penetration assemblies on each unit were determined to support SCs inside containment that are in the scope of license renewal. The twenty-two electrical/I&C penetration assemblies that are in the scope of license renewal, but not included in the Environmental Qualification Program, meet the criterion of 10 CFR 54.21(a)(1)(ii) and are subject to an aging management review.

2.5.4 ELECTRICAL/I&C COMPONENTS REQUIRING AN AGING MANAGEMENT REVIEW

The electrical and I&C component commodity groups subject to an aging management review include:

- Insulated Cables and Connections (including splices, connectors, and terminal blocks and splices) not included in the Environmental Qualification Program
- Uninsulated Ground Conductors
- Twenty-two electrical/I&C penetration assemblies that are in the scope of license renewal but not included in the Environmental Qualification Program

The intended function for the electrical and I&C component commodity groups subject to an aging management review is to electrically connect specified sections of an electrical circuit to deliver voltage, current, or signals. A complete list of electrical and I&C component commodity groups requiring an aging management review and the component commodity group intended functions are provided in Table 3.7-5. The aging management review for electrical and I&C component commodity groups is discussed in Section 3.7.

2.5.5 REFERENCES

2.5-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 1, Nuclear Energy Institute, January 2000.

TABLE 2.5-1
ELECTRICAL/I&C COMPONENT COMMODITY GROUPS

ELECTRICAL/I&C COMPONENT COMMODITY GROUPS INSTALLED AT TURKEY POINT FOR IN-SCOPE SYSTEMS AND STRUCTURES			
Alarm units	Circuit breakers	Fuses	Signal conditioners
Analyzers	Communication equipment	Generators/motors	Solenoid operators
Annunciators		Heat tracing	Solid-state devices
Batteries	Electrical/I&C controls and panel internal component assemblies	Heaters	Surge arresters
Cables and connections (terminal blocks, connectors, and splices) Bus - insulated cables and connectors		Indicators	Switches
		Isolators	Switchgear Motor control centers Power distribution panels
		Light bulbs	
		Loop controllers	
Cables and connections (terminal blocks, connectors, and splices) Bus - uninsulated ground cables	Electrical/I&C penetration assemblies	Meters	Transformers
		Power supplies	
		Radiation monitors	Transmitters
	Elements Resistance temperature detectors (RTDs) Sensors Thermocouples Transducers	Recorders	
Chargers Converters Inverters		Regulators	
		Relays	

3.0 AGING MANAGEMENT REVIEW RESULTS

For those structures and components that are identified as being subject to an aging management review, 10 CFR 54.21(a)(3) requires demonstration that the effects of aging will be adequately managed so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. The information provided in this chapter provides essential input to the required aging management review as it identifies and discusses the aging effects requiring management.

This chapter describes the results of the aging management reviews of the components and structures, identified in Chapter 2, "Structures and Components Subject to Aging Management Review." This chapter:

- provides references to the descriptions of common aging management programs
- identifies the components and structural components subject to aging management review, and their intended functions
- discusses the materials and internal and external environments
- describes or references the processes used to identify aging effects
- describes industry and plant-specific operating experiences with respect to the aging effects
- identifies the aging effects requiring management
- lists the aging management programs for aging effects requiring management.

Common aging management programs are contained in Section 3.1. For those structures and components identified as being subject to an aging management review, the results are contained in Section 3.2 for Reactor Coolant Systems, Section 3.3 for Engineered Safety Features Systems, Section 3.4 for Auxiliary Systems, Section 3.5 for Steam And Power Conversion Systems, Section 3.6 for structures and structural components (Subsection 3.6.1 for Containment and Subsection 3.6.2 for other structures), and Section 3.7 for electrical and instrumentation and controls. Aging management program descriptions are contained in Appendix B.

Tables 3.0-1 and 3.0-2 contain descriptions of the internal and external service environments at Turkey Point, which will be used in subsequent sections of this chapter. The environments used in the aging management reviews are listed in the "Environment" column in Tables 3.0-1 and 3.0-2. Within this Application, some of the internal environments have been subdivided into groups based on the fluid chemistry. The subgroups are identified in the "Description" column in Table 3.0-1.

TABLE 3.0-1
INTERNAL SERVICE ENVIRONMENTS

Environment	Description
Air/Gas	Includes atmospheric air, dry/filtered instrument air, nitrogen, hydrogen, carbon dioxide, and Halon
Treated water	Base water for all clean systems. Demineralized water that can be deaerated, or include corrosion inhibitors, biocides, and boric acid, or any combination of these treatments Within this Application, treated water has been subdivided into groups based on the chemistry of the water: Treated water – primary – Normal operating Reactor Coolant System chemistry Treated water – secondary – Normal operating secondary chemistry, including Main Steam, Feedwater, and Blowdown Systems Treated water – borated – Systems that contain borated water except those included in treated water – primary, including Chemical and Volume Control, Spent Fuel Cooling, and Emergency Core Cooling Systems Treated water – All other treated water systems, including Component Cooling Water, Emergency Diesel Generator Cooling, and Chilled Water Systems
Raw water	Water that enters the plant from the cooling water canals, ocean, bay, or city water source that has not been demineralized. In general, the water has been rough filtered to remove large particles and may contain a biocide for control of micro-organisms and macro-organisms. Although city water is purified for drinking purposes, it is conservatively classified as raw water for the purposes of aging management review. Within this Application, raw water has been subdivided into groups based on the chemistry of the water: Raw water – cooling canals – Salt water used as the ultimate heat sink Raw water – city water – Potable water supplied to the water treatment plant and the Fire Protection System Raw water – floor drainage – Fluids collected in building drains. The fluids can be treated water (primary, secondary, borated, or other), raw water (cooling water canals or city water), fuel oil, or lubricating oil
Fuel oil	Emergency diesel generator, diesel fire pump, and standby steam generator feedwater pump fuel oil
Lubricating oil	Lubricating oil for emergency diesel generators, pumps, and other components
Ohmic heating	Thermal stress on power cable materials can be due to ohmic heating resulting from electrical current

TABLE 3.0-2
EXTERNAL SERVICE ENVIRONMENTS

Environment [1]	Description
Outdoor [2]	Moist, salt-laden atmospheric air, temperature 30ºF-95ºF, humidity 5%-95%, exposed to weather, including precipitation and wind
Indoor – not air conditioned[2]	Atmospheric air, temperature 104ºF (40ºC) maximum, humidity 5%-95ºF%, not exposed to weather
Indoor – air conditioned[2]	Atmospheric air, specific temperature/humidity range dependent on building/room. Typically, temperature 70ºF-79ºF, humidity 60%-80%, not exposed to weather
Containment air [2]	Atmospheric air, temperature 120ºF maximum, humidity 5%-95%, total integrated dose rate – 1 rad/hour (excluding equipment located inside the reactor cavity), not exposed to weather Note: Safety-related equipment in the containment has been analyzed to 122ºF (50ºC) continuous and 125ºF for 2 weeks/year
Borated water leaks	Exposed to leakage from borated water systems
Buried	Above groundwater elevation, exposed to soil/fill. Below groundwater elevation, exposed to soil/fill and groundwater. Groundwater contains aggressive chemicals that can attack susceptible materials
Embedded/Encased	Reinforcing or embedded steel or piping in concrete

NOTES: 1. For certain components and structural components that are submerged, the applicable environment in Table 3.0-1 is specified.
2. Where wetted conditions exist (e.g., condensation), the item is annotated with the applicable external environment in Chapter 3 system and structures tables.

3.1 COMMON AGING MANAGEMENT PROGRAMS

The Turkey Point programs described in Appendix B are credited for managing the effects of aging. This section does not describe the aging management programs nor discuss how the programs will manage the identified aging effects. Appendix B describes these programs and provides the information necessary to demonstrate that the identified aging effects will be adequately managed, to provide reasonable assurance that the systems, structures, and components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.1.1 CHEMISTRY CONTROL PROGRAM

This information is contained in Appendix B, Subsection 3.2.4.

3.1.2 FPL QUALITY ASSURANCE PROGRAM

This information is contained in Appendix B, Section 2.0.

3.1.3 SYSTEMS AND STRUCTURES MONITORING PROGRAM

This information is contained in Appendix B, Subsection 3.2.15.

3.2 REACTOR COOLANT SYSTEMS

Reactor Coolant Systems components within the scope of license renewal that require aging management review are identified in Subsection 2.3.1. The following reactor coolant mechanical and structural components are included in this section:

- Reactor coolant piping
- Regenerative and excess letdown heat exchangers
- Pressurizers
- Reactor vessels
- Reactor vessel internals
- Reactor coolant pumps
- Steam generators

Determination of the aging effects related to Reactor Coolant Systems components begins with identification of aging effects defined in industry literature. From this set of aging effects, the materials, operating environments, and operating stresses define the aging effects requiring management for each component that is subject to an aging management review. Aging effects requiring management are then validated by a review of industry and Turkey Point Units 3 and 4 operating experience to provide assurance that all aging effects requiring management are identified.

The determination of aging effects requiring management considers the materials, environments, and stresses of Turkey Point Units 3 and 4 components. The aging effects requiring management for the Reactor Coolant Systems components are discussed in the following subsections.

The only areas inaccessible for inspection for the Reactor Coolant Systems are a limited number of locations internal to certain components (i.e., pressurizers, reactor vessel internals, and steam generators). Aging effects associated with these areas are addressed in the aging management review discussed in Subsections 3.2.3, 3.2.5, and 3.2.7 respectively.

3.2.1 REACTOR COOLANT PIPING

Reactor coolant piping consists of Class 1 and non-Class 1 components. The aging management review results for reactor coolant Class 1 piping components are discussed in Subsection 3.2.1.1, and reactor coolant non-Class 1 piping components are discussed in Subsection 3.2.1.2.

3.2.1.1 CLASS 1 PIPING

Reactor coolant Class 1 piping components are within the scope of license renewal as discussed in Subsection 2.3.1.2. Reactor coolant Class 1 piping components that are subject to an aging management review are listed in Table 3.2-1.

As stated previously in Subsection 2.3.1.2, WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components" [References 3.2-1 through 3.2-3], is not incorporated by reference in this Application. However, the Turkey Point aging management review was compared to WCAP-14575 with the results presented below.

The design and operation of the reactor coolant piping were reviewed using the process described in Subsection 2.3.1.1.1. This review confirmed that the Turkey Point Units 3 and 4 reactor coolant Class 1 piping is bounded by the description of Class 1 piping contained in WCAP-14575 with regard to design criteria and features, materials of construction, fabrication techniques, installed configuration, modes of operation, and environments/exposures. The component intended functions for reactor coolant Class 1 piping are inclusive of the intended functions identified in WCAP-14575. In addition to the functions identified in WCAP-14575, Turkey Point has identified an additional function for the flow restricting orifices and reducers. These orifices and reducers provide throttling to limit the maximum flow through a postulated line break in an attached non-Class 1 line to a value within the makeup capability of the Chemical and Volume Control System. These orifices and reducers provide the code class break.

Several open items and applicant action items were identified and documented in the NRC draft safety evaluation of WCAP-14575 [Reference 3.2-4]. The Turkey Point-specific responses to those open items and applicant action items relevant to the identification of reactor coolant piping components subject to aging management review are provided in Tables 2.3-2 and 2.3-3.

WCAP-14575 identified loss of material due to wear of reactor coolant pump and Class 1 valve mechanical closures. Turkey Point has concluded that loss of material due to wear does not require aging management for reactor coolant Class 1 piping components. Turkey Point has determined that cracking due to stress corrosion and loss of mechanical closure integrity due to aggressive chemical attack and stress relaxation are additional aging effects, not included in WCAP-14575, that require management in the license renewal term.

The aging management programs described in WCAP-14575 include six attributes and are established on an aging mechanism basis. The Turkey Point aging management programs referred to in this aging management review and described in Appendix B contain ten attributes, and are established on a program basis.

3.2.1.1.1 MATERIALS AND ENVIRONMENTS

Reactor coolant Class 1 piping components are exposed to an internal environment of treated water – primary, and external environments of containment air and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

Reactor coolant Class 1 piping components are constructed of stainless steel and low alloy steel. Note there are no Alloy 600 penetrations associated with reactor coolant Class 1 piping components. The piping components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.1.1.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Reduction in fracture toughness
- Loss of mechanical closure integrity

CRACKING

Cracking due to flaw growth and stress corrosion is an aging effect requiring management for the period of extended operation. At Turkey Point, cracking due to fatigue is identified as a Time-Limited Aging Analysis (TLAA) and is addressed in Subsections 4.3.1 and 4.3.4.

Growth of original manufacturing flaws over time by service loading can cause cracking. Detection and evaluation of flaws is important in maintaining the structural integrity of the reactor coolant Class 1 pressure boundary. ASME Section XI inservice examinations of components are intended to detect significant flaw growth and development. These examinations provide assurance that significant flaws do not exist, or a large flaw subject to crack growth would be detected so that it could be characterized, evaluated, and repaired, if necessary. In order to confirm that cracking is not occurring in small bore (<4 inches in diameter) reactor coolant Class 1 piping, a one-time inspection to complement present ASME Section XI examinations will be performed. Continued performance of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, supplemented by the one-time Small Bore Class 1 Piping Inspection, provides assurance that flaw growth is managed and that the intended function of reactor coolant Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

Stress corrosion cracking is a localized, nonductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate susceptible material from reactor coolant Class 1 piping components. In addition, to reduce the susceptibility of reactor coolant Class 1 piping component materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides assurance that stress corrosion cracking is managed and that the intended function of reactor coolant Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

REDUCTION IN FRACTURE TOUGHNESS

Reduction in fracture toughness due to thermal embrittlement is an aging effect requiring management for the period of extended operation. Thermal embrittlement refers to gradual and progressive changes in the microstructure and properties of a

material due to exposure to elevated temperatures for an extended period. The only reactor coolant Class 1 piping components subject to reduction in fracture toughness due to thermal embrittlement are austenitic stainless steel castings. Cast austenitic stainless steel (CASS) reactor coolant Class 1 piping components at Turkey Point consist of the primary loop elbows, reactor coolant pump casings and closure flanges, and selected valves exceeding a temperature threshold criterion of 482ºF. Reduction in fracture toughness of the reactor coolant pump casings and closures is discussed in Subsection 3.2.6.

Since the primary loop elbows are cast austenitic stainless steel, the reactor coolant primary loop Leak-Before-Break (LBB) analysis considers the effects of thermal embrittlement. The LBB analysis has been identified as a TLAA and has been reevaluated for the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii) (see Subsection 4.7.3). Consistent with the conclusions drawn in the NRC draft safety evaluation for WCAP-14575, Turkey Point has chosen the evaluation method to disposition reduction in fracture toughness due to thermal embrittlement of the primary loop elbows. Therefore, an aging management program to manage this effect for the reactor coolant primary loop CASS elbows is not required.

Consistent with the conclusions drawn in the NRC draft safety evaluation for WCAP-14575, screening of Class 1 CASS valves for susceptibility to thermal embrittlement is not required during the period of extended operation because the reduction in fracture toughness of these components should not have a significant impact on critical flaw size. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that reduction in fracture toughness due to thermal aging is managed and that the intended function of the reactor coolant Class 1 CASS valves is maintained consistent with the current licensing basis for the period of extended operation.

LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity can result from stress relaxation and/or aggressive chemical attack.

Loss of mechanical closure integrity due to stress relaxation is a relevant aging effect that requires management. This aging effect can be managed by periodic in-service inspections and leakage testing. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that loss of

mechanical closure integrity due to stress relaxation is managed and that the intended function of reactor coolant Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

Loss of mechanical closure integrity due to aggressive chemical attack has been observed in the industry and is the most common aging mechanism of concern for ferritic fasteners of stainless steel components. Mechanical closure bolting associated with reactor coolant Class 1 piping components is made of low alloy steel bolting material and is subject to aggressive chemical attack from potential borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of mechanical closure integrity due to aggressive chemical attack is managed and that the intended function of reactor coolant Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

3.2.1.1.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to reactor coolant Class 1 and non-Class 1 piping components includes the following:

- NRC Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants"
- NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems"
- NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification"
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Information Notice 79-19, " Pipe Cracks in Stagnant Borated Water Systems at PWR Plants"
- NRC Information Notice 82-14, "TMI-1 Steam Generator/Reactor Coolant System Chemistry/Corrosion Problem"
- NRC Information Notice 82-30, "Loss of Thermal Sleeves in Reactor Coolant System Piping at Certain Westinghouse PWR Power Plants"

- NRC Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting From Boric Acid Corrosion"
- NRC Information Notice 87-46, "Undetected Loss of Reactor Coolant"
- NRC Information Notice 88-30, "Target Rock Two-Stage Safety-Relief Valve Setpoint Drift Update"
- NRC Information Notice 88-80, "Unexpected Piping Movement Attributed to Thermal Stratification"
- NRC Information Notice 89-07, "Failures of Small Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Render Emergency Diesels Inoperable"
- NRC Information Notice 91-74, "Changes in Pressurizer Safety Valve Setpoints Before Installation"
- NRC Information Notice 91-87, "Hydrogen Embrittlement of Raychem Craft Couplings"
- NRC Information Notice 92-15, "Failure of Primary System Compression Fittings"
- NRC Information Notice 92-36, "Intersystem Loss-of-Coolant Accident Outside Containment"
- NRC Information Notice 92-86, "Unexpected Restriction to Thermal Growth of Reactor Coolant Piping"
- NRC Information Notice 93-02, "Malfunction of a Pressurizer Code Safety Valve"
- NRC Information Notice 93-66, "Switchover to Hot-Leg Injection Following a Loss-of-Coolant Accident in Pressurized Water Reactors"
- NRC Information Notice 93-90, "Unisolatable Reactor Coolant System Leak Following Repeated Application of Leak Sealant"
- NRC Information Notice 94-55, "Problems with Copes-Vulcan Pressurizer Power-Operated Relief Valves"
- SOER 25-87, "Pressurizer Surge Line Thermal Stratification"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.2.1.1.2. Note that a summary of industry experience associated with reactor coolant Class 1 piping components is also provided in WCAP-14575.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of reactor coolant Class 1 piping component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.2.1.1.2.

3.2.1.1.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.1.1.2. Table 3.2-1 contains the results of the aging management review for reactor coolant Class 1 piping components and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Small Bore Class 1 Piping Inspection

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of reactor coolant Class 1 piping components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.1.2 NON-CLASS 1 PIPING

Reactor coolant non-Class 1 piping components are within the scope of license renewal, as discussed in Subsection 2.3.1.2. Reactor coolant non-Class 1 piping components that are subject to an aging management review are listed in Table 3.2-1.

3.2.1.2.1 MATERIALS AND ENVIRONMENTS

Reactor coolant non-Class 1 piping components are exposed to internal environments of air/gas, treated water, treated water – primary, and lubricating oil; and external environments of containment air and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

Reactor coolant non-Class 1 piping components are constructed of stainless steel, low alloy steel, carbon steel, admiralty brass, and 90/10 copper nickel. The reactor coolant non-Class 1 piping components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.1.2.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Loss of material
- Loss of mechanical closure integrity

<u>CRACKING</u>

Cracking due to stress corrosion is an aging effect requiring management for the period of extended operation. At Turkey Point, cracking due to fatigue is identified as a Time-Limited Aging Analysis and is addressed in Subsection 4.3.4.

Stress corrosion cracking is a localized, nonductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate susceptible material from reactor coolant non-Class 1 piping components. In addition, to reduce the susceptibility of reactor coolant non-Class 1 piping component materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides assurance that stress corrosion cracking is managed and that the intended function of the reactor coolant non-Class 1 piping

components is maintained consistent with the current licensing basis for the period of extended operation.

LOSS OF MATERIAL

Loss of material is an aging effect requiring management for the period of extended operation. The aging mechanisms that can cause loss of material for reactor coolant non-Class 1 piping components are general corrosion, crevice corrosion, pitting corrosion, microbiologically influenced corrosion, selective leaching, galvanic corrosion, and aggressive chemical attack.

General corrosion, crevice corrosion, pitting corrosion, microbiologically influenced corrosion, and selective leaching have been identified as aging mechanisms for internal surfaces of reactor coolant non-Class 1 piping components. The Chemistry Control Program provides an effective means of controlling these corrosion mechanisms and provides assurance that the intended function of reactor coolant non-Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

Galvanic corrosion has been identified as an aging mechanism between the reactor coolant pump lower bearing heat exchanger tube coil (copper alloy) and the component cooling water supply piping (carbon steel); and between the reactor coolant pump upper bearing heat exchanger tubes (brass) and the carbon steel heat exchanger tube sheet. Although galvanic action is considered as a corrosion mechanism, no adverse effect of galvanic corrosion has been identified for these material combinations and environments at Turkey Point. The Galvanic Corrosion Susceptibility Inspection Program provides assurance that galvanic corrosion is managed and that the intended function of reactor coolant non-Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

General corrosion and pitting corrosion have been identified as aging mechanisms for external surfaces of carbon steel components. Although existing protective coatings applied to these surfaces have effectively protected them from corrosion effects, the Systems and Structures Monitoring Program provides assurance that general corrosion and pitting corrosion are managed and that the intended function of reactor coolant non-Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

Aggressive chemical attack was identified as an aging mechanism for external surfaces of carbon steel components exposed to potential borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of material due to aggressive chemical attack is managed and that the intended function of reactor coolant non-Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity can result from aggressive chemical attack.

Loss of mechanical closure integrity due to aggressive chemical attack has been observed in the industry and is the most common aging mechanism of concern for ferritic fasteners of stainless steel components. Mechanical closure bolting associated with reactor coolant non-Class 1 piping components is made of low alloy steel bolting material and is subject to aggressive chemical attack from potential borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of mechanical closure integrity due to aggressive chemical attack is managed and that the intended function of reactor coolant non-Class 1 piping components is maintained consistent with the current licensing basis for the period of extended operation.

3.2.1.2.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to reactor coolant non-Class 1 piping components is included in the listing provided in Subsection 3.2.1.1.3.

No aging effects requiring management were identified from the documents listed in Subsection 3.2.1.1.3 beyond those already identified in Subsection 3.2.1.2.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of

Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of reactor coolant non-Class 1 piping component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.2.1.2.2.

3.2.1.2.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.1.2.2. Table 3.2-1 contains the results of the aging management review for reactor coolant non-Class 1 piping components and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Galvanic Corrosion Susceptibility Inspection Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of reactor coolant non-Class 1 piping components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.2 REGENERATIVE AND EXCESS LETDOWN HEAT EXCHANGERS

The regenerative and excess letdown heat exchangers are within the scope of license renewal, as discussed in Subsection 2.3.1.3. Regenerative and excess letdown heat exchanger components that are subject to an aging management review are listed in Table 3.2-1.

3.2.2.1 MATERIALS AND ENVIRONMENTS

The regenerative and excess letdown heat exchangers are exposed to internal environments of treated water and treated water - primary, and external environments of containment air and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

The regenerative and excess letdown heat exchangers are constructed of stainless steel, low alloy steel, and carbon steel. The heat exchanger components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.2.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Loss of material
- Loss of mechanical closure integrity
- Fouling

3.2.2.2.1 CRACKING

Cracking due to stress corrosion is an aging effect requiring management for the period of extended operation.

Stress corrosion cracking is a localized, non-ductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate

susceptible material from the regenerative and excess letdown heat exchangers. In addition, to reduce the susceptibility of regenerative and excess letdown heat exchangers materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides assurance that stress corrosion cracking is managed and that the intended function of regenerative and excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

3.2.2.2.2 LOSS OF MATERIAL

Loss of material is an aging effect requiring management for the period of extended operation. The aging mechanisms that can cause loss of material for the excess letdown heat exchangers are general corrosion, crevice corrosion, pitting corrosion, galvanic corrosion, microbiologically influenced corrosion, and aggressive chemical attack. Note that the regenerative heat exchangers are of all welded, stainless steel construction and not subject to loss of material.

General corrosion has been identified as an aging mechanism for internal carbon steel surfaces of the excess letdown heat exchangers. Microbiologically influenced corrosion has been identified as an aging mechanism for the stainless steel tube sheets and the outside diameter of the stainless steel tubing of the excess letdown heat exchangers. These parts are exposed to component cooling water that contains dissolved oxygen. The addition of corrosion inhibitors to the component cooling water, in accordance with the Chemistry Control Program, provides an effective means of controlling general and microbiologically influenced corrosion in the shell sides of the excess letdown heat exchangers and provides assurance that the intended function of the excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

Galvanic corrosion has been identified as an aging mechanism for the internal surfaces of the carbon steel shells of the excess letdown heat exchangers at the vicinity of their contact point with the stainless steel tube sheets. Although galvanic action is considered as a corrosion mechanism, no adverse effect of galvanic corrosion has been identified for these material combinations and environments at Turkey Point. The Galvanic Corrosion Susceptibility Inspection Program provides assurance that galvanic corrosion is managed and that the intended function of the excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

The external carbon steel surfaces of the excess letdown heat exchanger shells are exposed to the containment air environment and are typically wetted with condensation when operating. General corrosion, crevice corrosion, pitting corrosion, and microbiologically influenced corrosion have been identified as aging mechanisms for external carbon steel surfaces of the excess letdown heat exchangers. Although existing protective coatings applied to these surfaces have effectively protected them from corrosion effects, the Systems and Structures Monitoring Program provides assurance that general corrosion, crevice corrosion, pitting corrosion, and microbiologically influenced corrosion are managed and that the intended function of the excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

Aggressive chemical attack was identified as an aging mechanism for the excess letdown heat exchanger external surfaces exposed to potential borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of material due to aggressive chemical attack is managed and that the intended function of the excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

3.2.2.2.3 LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity can result from aggressive chemical attack.

Loss of mechanical closure integrity due to aggressive chemical attack has been observed in the industry and is the most common aging mechanism of concern for ferritic fasteners of stainless steel components. Mechanical closure bolting associated with the excess letdown heat exchangers is made of low alloy steel bolting material and is subject to aggressive chemical attack from potential borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of mechanical closure integrity due to aggressive chemical attack is managed and that the intended function of the excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation. Note that there are no bolted mechanical closures associated with the regenerative heat exchangers.

3.2.2.2.4 FOULING

Aging mechanisms that can result in fouling of the heat exchanger heat transfer surfaces include biological fouling and particulate fouling.

Biological fouling has been identified as an aging mechanism affecting the excess letdown heat exchanger tubing exposed to component cooling water. The addition of biocides, in accordance with the Chemistry Control Program, provides assurance that the intended function of the excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

Particulate fouling has been identified as an aging mechanism for the regenerative and excess letdown heat exchanger tubing. The Chemistry Control Program provides assurance that particulate fouling is managed and that the intended function of the regenerative and excess letdown heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

3.2.2.3 OPERATING EXPERIENCE

3.2.2.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to the regenerative and excess letdown heat exchangers includes the following:

- NRC Bulletin 79-17, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”
- NRC Circular 76-06, “Stress Corrosion Cracks in Stagnant, Low Pressure Stainless Piping Containing Boric Acid Solution at PWRs"
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Information Notice 79-19, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”
- SAND 93-7070, "Aging Management Guideline for Commercial Nuclear Power Plants - Heat Exchangers"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.2.2.2.

3.2.2.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of

Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of regenerative and excess letdown heat exchanger component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.2.2.2.

3.2.2.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.2.2. Table 3.2-1 contains the results of the aging management review for the regenerative and excess letdown heat exchangers and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Galvanic Corrosion Susceptibility Inspection Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the regenerative and excess letdown heat exchanger components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.3 PRESSURIZERS

The pressurizers are within the scope of license renewal, as discussed in Subsection 2.3.1.4. Pressurizer components that are subject to an aging management review are listed in Table 3.2-1.

As stated previously in Subsection 2.3.1.4, WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers" [References 3.2-3, 3.2-5, and 3.2-6], is not incorporated by reference in this Application. However, the Turkey Point aging management review was compared to WCAP-14574 with the results presented below.

The design and operation of the Turkey Point Units 3 and 4 pressurizers were reviewed using the process described in Subsection 2.3.1.1.1. This review confirmed that the Turkey Point pressurizers are bounded by the description of the pressurizer in WCAP-14574 with regard to design criteria and features, modes of operation, intended functions, and environments/exposures. The materials of the Turkey Point pressurizers correspond to those in WCAP-14574 with the exception of the shells. The Turkey Point pressurizer shells are made of ASTM A-302 Grade B rather than SA 533 Grade A Class 2, as specified in WCAP-14574. This does not constitute a significant deviation because these materials are essentially the same. The fabrication techniques and installed configuration are the same as those specified in WCAP-14574 with the exception of the earthquake lugs and valve support brackets. The Turkey Point pressurizers do not include these items.

WCAP-14574 identifies ASME Section XI inspections as the program to manage stress corrosion cracking of the pressurizer nozzle safe ends. Stress corrosion cracking of stainless steel materials in the primary coolant environment is effectively managed by the Turkey Point Chemistry Control Program. Turkey Point has determined that cracking due to flaw growth is an additional aging effect that requires management at Turkey Point. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program is credited to manage the aging effects of cracking due to flaw growth.

The aging management programs described in WCAP-14574 include six attributes and are established on an aging mechanism basis. The Turkey Point aging management programs referred to in this aging management review and described in Appendix B contain ten attributes, and are established on a program basis.

3.2.3.1 MATERIALS AND ENVIRONMENTS

The pressurizers are exposed to an internal environment of treated water - primary and external environments of containment air and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

Pressurizer components are constructed of stainless steel, alloy steel, and carbon steel. Note there are no Alloy 600 penetrations associated with the pressurizers. The pressurizer components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.3.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Loss of material
- Loss of mechanical closure integrity

3.2.3.2.1 CRACKING

Cracking due to flaw growth and stress corrosion is an aging effect requiring management for the period of extended operation. At Turkey Point, cracking due to fatigue is identified as a Time-Limited Aging Analysis and is analytically addressed in Subsection 4.3.1.

Growth of original manufacturing flaws over time by service loading can cause cracking. Detection and evaluation of flaws is important in maintaining the structural integrity of the pressurizer pressure boundary. ASME Section XI inservice examinations of components are intended to detect significant flaw growth and development. These examinations provide assurance that significant flaws do not exist, or a large flaw subject to crack growth would be detected so that it could be characterized, evaluated, and repaired, if necessary. Continued performance of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that flaw growth is managed and that the intended function of

the pressurizers is maintained consistent with the current licensing basis for the period of extended operation.

Stress corrosion cracking is a localized, nonductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate susceptible material from the pressurizers. In addition, to reduce the susceptibility of pressurizer materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides assurance that stress corrosion cracking is managed and that the intended function of the pressurizers is maintained consistent with the current licensing basis for the period of extended operation.

3.2.3.2.2 LOSS OF MATERIAL

Loss of material due to aggressive chemical attack was identified as an aging mechanism requiring management for external surfaces of carbon and low alloy steel pressurizer components exposed to borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of material due to aggressive chemical attack is managed and that the intended function of the pressurizer is maintained consistent with the current licensing basis for the period of extended operation.

3.2.3.2.3 LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity can result from stress relaxation and/or aggressive chemical attack.

Loss of mechanical closure integrity due to stress relaxation is a relevant aging effect that requires management. This aging effect can be managed by periodic in-service inspections and leakage testing. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that loss of mechanical closure integrity due to stress relaxation is managed and that the intended function of the pressurizers is maintained consistent with the current licensing basis for the period of extended operation.

Loss of mechanical closure integrity due to aggressive chemical attack has been observed in the industry and is the most common aging mechanism of concern for ferritic fasteners of stainless steel components. Mechanical closure bolting associated with the pressurizers is made of low alloy steel bolting material and is

subject to aggressive chemical attack from potential borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of mechanical closure integrity due to aggressive chemical attack is managed and that the intended function of the pressurizers is maintained consistent with the current licensing basis for the period of extended operation.

3.2.3.3 OPERATING EXPERIENCE

3.2.3.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for pressurizer operating experience includes the following:

- NRC Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants”
- NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems"
- NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification"
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Information Notice 82-30, "Loss of Thermal Sleeves in Reactor Coolant System Piping at Certain Westinghouse PWR Power Plants"
- NRC Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting From Boric Acid Corrosion"
- NRC Information Notice 88-30, "Target Rock Two-Stage Safety-Relief Valve Setpoint Drift Update"
- NRC Information Notice 88-80, "Unexpected Piping Movement Attributed to Thermal Stratification"
- NRC Information Notice 90-10, “Primary Water Stress Corrosion Cracking of Inconel 600”
- NRC Information Notice 91-74, "Changes in Pressurizer Safety Valve Setpoints Before Installation"

- NRC Information Notice 93-02, "Malfunction of a Pressurizer Code Safety Valve"
- NRC Information Notice 94-55, "Problems with Copes-Vulcan Pressurizer Power-Operated Relief Valves"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.2.3.2. Note that a summary of industry experience associated with pressurizers is also provided in WCAP-14574.

3.2.3.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of pressurizer component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.2.3.2.

3.2.3.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.3.2. Table 3.2-1 contains the results of the aging management review for the pressurizers and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the pressurizer components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.4 REACTOR VESSELS

The reactor vessels are within the scope of license renewal, as discussed in Subsection 2.3.1.5. Reactor vessel components that are subject to an aging management review are listed in Table 3.2-1. Note that Turkey Point has included bottom mounted instrumentation tubing, thimble tubes, and the seal tables within the scope of this subsection.

3.2.4.1 MATERIALS AND ENVIRONMENTS

Reactor vessel components are exposed to internal environments of treated water - primary and air/gas, and external environments of containment air, treated water – primary (per note 1 on Table 3.0-2 for thimble tubes), and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

Reactor vessel components are constructed of stainless steel, low alloy steel, carbon steel, and Alloy 600. Reactor vessel components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.4.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Reduction in fracture toughness
- Loss of material
- Loss of mechanical closure integrity

3.2.4.2.1 CRACKING

Cracking due to flaw growth and stress corrosion is an aging effect requiring management for the period of extended operation. At Turkey Point, cracking due to fatigue (including reactor vessel underclad cracking) is identified as a Time-Limited Aging Analysis (TLAA) and is analytically addressed in Subsections 4.3.1 and 4.3.2.

Growth of original manufacturing flaws over time by service loading can cause cracking. Detection and evaluation of flaws is important in maintaining the structural integrity of the reactor vessel pressure boundary. ASME Section XI inservice examinations of components are intended to detect significant flaw growth and development. These examinations provide assurance that significant flaws do not exist, or a large flaw subject to crack growth would be detected so that it could be characterized, evaluated, and repaired, if necessary. Continued performance of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that flaw growth is managed and that the intended function of the reactor vessels is maintained consistent with the current licensing basis for the period of extended operation.

Stress corrosion cracking is a localized, nonductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate susceptible material from the reactor vessels. In addition, to reduce the susceptibility of reactor vessel materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides assurance that stress corrosion cracking is managed and that the intended function of the reactor vessels is maintained consistent with the current licensing basis for the period of extended operation.

Primary water stress corrosion cracking of the control rod drive mechanism housing tubes is a recognized industry issue. The Reactor Vessel Head Alloy 600 Penetration Inspection Program has been specifically designed to address primary water stress corrosion cracking of control rod drive mechanism housing tubes. The Reactor Vessel Head Alloy 600 Penetration Inspection Program, in conjunction with the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Chemistry Control Program, provide assurance that the intended function of the control rod drive mechanism housing tubes is maintained consistent with the current licensing basis for the period of extended operation. Note that the reactor vessels are the only Reactor Coolant System components with Alloy 600 penetrations at Turkey Point.

Stress corrosion cracking is an aging mechanism for reactor vessel closure studs and nuts. Visual, surface, and volumetric inspections performed as part of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program have been proven to be effective for managing the aging effects of stress corrosion

cracking and provide assurance that the intended function of the reactor vessel closure studs and nuts is maintained consistent with the current licensing basis for the period of extended operation.

Stress corrosion cracking of the external surfaces of the bottom mounted instrumentation guide tubes has been previously experienced at Turkey Point. Although corrective actions have been implemented which have virtually eliminated future stress corrosion cracking of these components, this aging effect requires management based on Turkey Point plant-specific experience (see Subsection 3.2.4.3.2 below). The Boric Acid Wastage Surveillance Program provides assurance that the intended function of the bottom mounted instrumentation guide tubes is maintained consistent with the current licensing basis for the period of extended operation.

3.2.4.2.2 REDUCTION IN FRACTURE TOUGHNESS

Fracture toughness is defined as the capability of a material to resist sudden failure caused by crack propagation. Fracture toughness of reactor vessel materials is reduced primarily by irradiation in the beltline region of the reactor vessel. Reduction in fracture toughness of reactor vessel beltline materials is an aging effect that requires management in the license renewal period. Several TLAAs associated with reduction in fracture toughness are addressed in Section 4.2. These TLAAs include pressurized thermal shock, upper-shelf energy, and pressure-temperature limit curves for heatup and cooldown. The Reactor Vessel Integrity Program ensures that the time-dependent parameters used in the TLAA evaluations remain valid for the license renewal period.

3.2.4.2.3 LOSS OF MATERIAL

Loss of material is an aging effect requiring management for the period of extended operation. The aging mechanisms that can cause loss of material for reactor vessels are general corrosion, mechanical wear, fretting wear, and aggressive chemical attack.

General corrosion has caused leakage of control rod drive mechanism canopy seal welds. Turkey Point Units 3 and 4 reactor vessels have experienced canopy seal weld leaks in the past. Canopy seal weld leaks are generally small (pin hole) in nature and are effectively managed through a combination of system pressure tests, performed in accordance with the requirements of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, and the Boric Acid

Wastage Surveillance Program. These programs provide assurance that the intended function of the canopy seal welds is maintained consistent with the current licensing basis for the period of extended operation.

Loss of material due to wear is an aging effect requiring management for the reactor closure studs, stud holes, nuts and washers, and core support lugs. Examinations performed as part of the existing ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provide assurance that the intended function of these reactor vessel components is maintained consistent with the current licensing basis for the period of extended operation.

Fretting wear is an aging mechanism that affects the bottom mounted instrumentation thimble tubes. The evaluation performed for thimble tube thinning has been identified as a TLAA and is discussed further in Subsection 4.7.1. Based on that evaluation, thimble tube N-05 requires aging management in accordance with 10 CFR 54.21(c)(1)(iii). The Thimble Tubes Inspection Program provides assurance that the intended function of the reactor vessel bottom mounted instrumentation thimble tubes is maintained consistent with the current licensing basis for the period of extended operation.

Loss of material due to aggressive chemical attack was identified as an aging mechanism requiring management for external surfaces of carbon and low alloy steel reactor vessel components exposed to borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of material due to aggressive chemical attack is managed and that the intended function of the reactor vessels is maintained consistent with the current licensing basis for the period of extended operation.

3.2.4.2.4 LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity can result from stress relaxation and/or aggressive chemical attack.

Loss of mechanical closure integrity due to stress relaxation is a relevant aging effect that requires management. This aging effect can be managed by periodic inservice inspections and leakage testing. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that loss of mechanical closure integrity due to stress relaxation is managed and that the intended function of the reactor vessels is maintained consistent with the current licensing basis for the period of extended operation.

Loss of mechanical closure integrity due to aggressive chemical attack has been observed in the industry and is the most common aging mechanism of concern for ferritic fasteners of stainless steel components. Mechanical closure bolting associated with the reactor vessels is made of low alloy steel bolting material and is subject to aggressive chemical attack from potential borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of mechanical closure integrity due to aggressive chemical attack is managed and that the intended function of the reactor vessels is maintained consistent with the current licensing basis for the period of extended operation.

3.2.4.3 OPERATING EXPERIENCE

3.2.4.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for reactor vessel operating experience includes the following:

- NRC Bulletin No. 88-09, “Thimble Tube Thinning in Westinghouse Reactors”
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Generic Letter 92-01, " Reactor Vessel Structural Integrity"
- NRC Generic Letter 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations”
- NRC Information Notice 87-44, "Thimble Tube Thinning in Westinghouse Reactors"
- NRC Information Notice 96-32, "Implementation of 10 CFR 50.55a(g)(6)(ii)(A), ‘Augmented Examination of Reactor Vessel’"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.2.4.2.

3.2.4.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports

for any documented instances of reactor vessel component aging, in addition to interviews with responsible engineering personnel. Turkey Point has experienced outside diameter initiated stress corrosion cracking of bottom mounted instrumentation guide tubes and loss of material due to general corrosion of canopy seal welds. Accordingly, aging management programs are identified to manage these effects. These aging effects are included with those identified in Subsection 3.2.4.2.

3.2.4.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.4.2. Table 3.2-1 contains the results of the aging management review for reactor vessel components and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Reactor Vessel Head Alloy 600 Penetration Inspection Program
- Reactor Vessel Integrity Program
- Thimble Tube Inspection Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the reactor vessel components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.5 REACTOR VESSEL INTERNALS

The reactor vessel internals are within the scope of license renewal, as discussed in Subsection 2.3.1.6. Reactor vessel internals components that are subject to an aging management review are listed in Table 3.2-1.

As stated previously in Subsection 2.3.1.6, WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals" [References 3.2-7 and 3.2-8], is not incorporated by reference in this Application. However, the Turkey Point aging management review was compared to WCAP-14577 with the results presented below.

The design and operation of the reactor vessel internals were reviewed using the process described in Subsection 2.3.1.1.1. This review confirmed that the Turkey Point Units 3 and 4 reactor vessel internals are bounded by the description of the reactor vessel internals in WCAP-14577 with regard to design criteria and features, modes of operation, intended functions, and environments/exposures. The materials of the Turkey Point Units 3 and 4 reactor vessel internals correspond to those in WCAP-14577. The fabrication techniques and installed configuration are the same as those specified in WCAP-14577.

WCAP-14577 was intended to bound all Westinghouse plants. However older plants, such as Turkey Point, have shorter fuel elements and the internals components with fluences greater than 10^{21} n/cm^2 do not include the lower support castings and the clevis bolts.

The discussion and evaluation of Time-Limited Aging Analyses in the reactor vessel internals in WCAP-14577 is general in nature and is not intended to satisfy plant-specific identification and disposition of reactor vessel internals TLAAs. WCAP-14577 identifies fatigue as the only TLAA generically applicable to Westinghouse reactor vessel internals, and then provides a discussion on options for dispositioning and management of fatigue. Turkey Point's TLAA identification effort also identified fatigue as the only TLAA applicable to the reactor vessel internals. Fatigue of the reactor vessel internals is addressed in Subsection 4.3.1.

The aging management programs described in WCAP-14577 include six attributes and are established on an aging mechanism basis. The Turkey Point aging management programs referred to in this aging management review and described in Appendix B contain ten attributes, and are established on a program basis.

3.2.5.1 MATERIALS AND ENVIRONMENTS

The reactor vessel internals are exposed to an environment of treated water - primary (see Table 3.0-1).

The reactor vessel internals components are constructed of stainless steel, Alloy 600, and Alloy X-750. The reactor vessel internals components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.5.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Reduction in fracture toughness
- Loss of material
- Loss of mechanical closure integrity
- Loss of preload
- Dimensional change

3.2.5.2.1 CRACKING

Cracking due to stress corrosion and irradiation assisted stress corrosion is an aging effect requiring management for the period of extended operation. At Turkey Point, cracking due to fatigue is identified as a Time-Limited Aging Analysis and is addressed in Subsection 4.3.1.

Stress corrosion cracking (SCC) is a localized, non-ductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate susceptible material from reactor vessel internals components. In addition, to reduce the susceptibility of reactor vessel internals materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides

assurance that stress corrosion cracking is managed and that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

Premature failure by intergranular environmental cracking of materials exposed to ionizing radiation has been termed irradiation assisted stress corrosion cracking (IASCC). Experience in pressurized water reactors (PWRs) in the United States, France, and Belgium indicates that IASCC is a plausible aging mechanism for PWR internals components. As with SCC, IASCC requires stress, environment, and a susceptible material. However, in the case of IASCC, a normally nonsusceptible material is rendered susceptible by exposure to neutron irradiation. Susceptibility has been observed at fluences as low as 1×10^{21} n/cm^2 (E > 0.1 MeV) in laboratory studies on Type 304 stainless steel in PWR environments. Type 316 stainless steel is less susceptible and field information suggests that greater exposures are required for the development of susceptibility. Therefore, reactor vessel internals components exposed to fluences greater than 1×10^{21} n/cm^2 (see Subsection 3.2.5.2.2) are potentially susceptible to irradiated assisted stress corrosion cracking. The Reactor Vessel Internals Inspection Program in conjunction with the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provide assurance that IASCC is managed and that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

3.2.5.2.2 REDUCTION IN FRACTURE TOUGHNESS

Reduction in fracture toughness due to thermal embrittlement and irradiation embrittlement is an aging effect requiring management for the period of extended operation. Thermal embrittlement refers to gradual and progressive changes in the microstructure and properties of a material due to exposure to elevated temperatures for an extended period. Reactor vessel internals components made from cast austenitic stainless steel (CASS) are potentially subject to reduction in fracture toughness due to thermal embrittlement. The only CASS components in the reactor vessel internals are the lower support castings. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Reactor Vessel Internals Inspection Program provide assurance that any reduction in fracture toughness due to thermal aging is managed and that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

Reduction in fracture toughness due to irradiation embrittlement is an aging effect requiring management for the period of extended operation. Exposure to high-energy neutrons can cause changes in the properties of stainless steel and nickel-based alloys used in reactor internals. Neutron irradiation can produce changes in mechanical properties by increasing yield and ultimate strength, and correspondingly decreasing ductility and fracture toughness of internals component materials. Studies show that embrittlement of stainless steel can occur at fluences as low as $1x10^{21}$ n/cm^2 (E> 0.1 MeV).

For Turkey Point Units 3 and 4, the following reactor vessel internals parts will be exposed to fluences greater than $1x10^{21}$ n/cm^2 (E > 0.1 MeV) during the period of extended operation and are potentially susceptible to irradiation embrittlement:

- lower core barrels
- baffle/former assemblies
- baffle/former bolts
- barrel/former bolts
- lower core plates, fuel pins
- lower support columns, bolts
- thermal shields

Therefore, reduction in fracture toughness due to irradiation embrittlement is an aging effect requiring management for the listed reactor vessel internals components. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program in conjunction with the Reactor Vessel Internals Inspection Program provide assurance that any reduction in fracture toughness due to irradiation embrittlement is managed and that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

3.2.5.2.3 LOSS OF MATERIAL

Loss of material due to mechanical wear is an aging effect requiring management for the period of extended operation. Loss of material due to wear can occur on the lower core plate fuel pins, core barrel flanges, guide tubes and guide pins, upper core plate alignment pins, and radial keys and clevis inserts. Inspections performed as part of the existing ASME Section XI, Subsections IWB, IWC, and IWD Inservice

Inspection Program provides assurance that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

3.2.5.2.4 LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity of upper support columns, guide tubes, and clevis insert bolts can occur due to cracking (SCC) and stress relaxation. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program in conjunction with the Chemistry Control Program provide assurance that loss of mechanical closure integrity is managed and that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

Loss of mechanical closure integrity associated with the lower support column, baffle/former bolts, and barrel/former bolts can occur due to cracking (IASCC), reduction in fracture toughness (irradiation embrittlement), irradiation creep, and stress relaxation. Significant data, information, and industry experience relative to the aging of this bolting is provided in WCAP-14577 [Reference 3.2-7] and, as such, not duplicated here. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program in conjunction with the Reactor Vessel Internals Inspection Program and Chemistry Control Program provide assurance that loss of mechanical closure integrity is managed and that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

3.2.5.2.5 LOSS OF PRELOAD

Loss of preload due to stress relaxation of the reactor vessel internals holddown springs is an aging effect requiring management for the period of extended operation. Inspections performed as part of the existing ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provide assurance that the loss of preload of the holddown springs is managed such that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

3.2.5.2.6 DIMENSIONAL CHANGE

Dimensional changes due to void swelling are a potential aging effect requiring management. Swelling, frequently referred to as cavity swelling or void swelling, is

defined as a gradual increase in size (dimensions) of a given reactor vessel internals part due to irradiation conditions. Void swelling has been postulated from laboratory testing for liquid metal fast breeder reactors (LMFBRs).

During the past 30 years, swelling of PWR internals components was not considered a significant age-related degradation mechanism. However, Garner, et al. [Reference 3.2-9], concluded that, based on LMFBR data, end-of-life exposures of some PWR internals will lead to significant levels (≥ 10%) of swelling. Foster, et al. [Reference 3.2-10], concluded that at the approximate reactor internals end-of-life dose of 100 displacements per atom, swelling would be less than 2% at irradiation temperatures between 572°F and 752°F. To date, field service experience in PWR plants has not shown any evidence of swelling.

With respect to swelling, industry data are currently being evaluated as part of Westinghouse Owners Group and Electric Power Research Institute (EPRI) Material Reliability Project programs. At present there have been no indications from the different reactor vessel internals bolt removal programs, or from any of the other inspection and functional evaluations (e.g., refueling), that there are any discernible effects attributable to swelling. An industry initiative to consider the accumulated data, engineering evaluations of the ramifications of swelling, and the field observations is presently scheduled to be complete in 2001.

In summary, it is known that void swelling can occur under certain conditions. The extent to which it degrades intended functions cannot be quantified but, at this point, some minor degradation in the form of dimensional changes is assumed. The Reactor Vessel Internals Inspection Program includes an evaluation of dimensional changes due to void swelling. If determined to be significant, program inspections will be performed. Therefore, the Reactor Vessel Internals Inspection Program provides assurance that the aging effect of dimensional change due to swelling is managed and that the intended function of the reactor vessel internals is maintained consistent with the current licensing basis for the period of extended operation.

3.2.5.3 OPERATING EXPERIENCE

3.2.5.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The

industry correspondence that was reviewed for reactor vessel internals operating experience includes the following:

- NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors”
- NRC Information Notice 87-19, “Perforation and Cracking of Rod Control Cluster Assemblies”
- NRC Information Notice 98-11, “Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants”

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.2.5.2. Note that a summary of industry experience associated with reactor vessel internals is also provided in WCAP-14577.

3.2.5.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of reactor vessel internals component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.2.5.2.

3.2.5.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.5.2. Table 3.2-1 contains the results of the aging management review for the reactor vessel internals components and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
- Chemistry Control Program
- Reactor Vessel Internals Inspection Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the reactor vessel internals components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.6 REACTOR COOLANT PUMPS

The reactor coolant pumps and integral thermal barrier heat exchangers are within the scope of license renewal, as discussed in Subsection 2.3.1.7. Reactor coolant pump components that are subject to an aging management review are listed in Table 3.2-1.

The reactor coolant pump is included in WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components" [References 3.2-1 through 3.2-3]. As stated previously in Subsection 2.3.1.7, WCAP-14575 is not incorporated by reference in this Application. However, the Turkey Point aging management review was compared to WCAP-14575 with the results presented below.

The design and operation of the reactor coolant pumps were reviewed using the process described in Subsection 2.3.1.1.1. This review confirmed that the Turkey Point Units 3 and 4 reactor coolant pumps are bounded by the description contained in WCAP-14575 with regard to design criteria and features, materials of construction, fabrication techniques, installed configuration, modes of operation, and environments/ exposures. The component intended functions for reactor coolant pumps are consistent with the intended functions identified in WCAP-14575.

Several open items and applicant action items were identified and documented in the NRC draft safety evaluation of WCAP-14575 [Reference 3.2-4]. The Turkey Point specific responses to those open items and applicant action items relevant to the identification of reactor coolant pump components subject to aging management review are provided in Tables 2.3-2 and 2.3-3.

WCAP-14575 identified loss of material due to wear of reactor coolant pump and Class 1 valve mechanical closures. Turkey Point has concluded that loss of material due to wear of mechanical closures does not require aging management for the reactor coolant pump. Turkey Point has determined that cracking due to stress corrosion and loss of mechanical closure integrity due to aggressive chemical attack are additional aging effects, not included in WCAP-14575, that require management in the license renewal term.

The aging management programs described in WCAP-14575 include six attributes and are established on an aging mechanism basis. The Turkey Point aging

management programs referred to in this aging management review and described in Appendix B contain ten attributes, and are established on a program basis.

3.2.6.1 MATERIALS AND ENVIRONMENTS

Reactor coolant pumps are exposed to an internal environment of treated water – primary, and external environments of containment air and potential borated water leaks. The integral thermal barrier heat exchangers are exposed to an internal environment of treated water and treated water – primary, and an external environment of containment air and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

The reactor coolant pump and integral thermal barrier heat exchanger components are constructed of stainless steel and low alloy steel. The reactor coolant pump and integral thermal barrier heat exchanger components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.6.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Reduction in fracture toughness
- Loss of material
- Loss of mechanical closure integrity
- Fouling

3.2.6.2.1 CRACKING

Cracking due to flaw growth and stress corrosion is an aging effect requiring management for the period of extended operation. At Turkey Point, cracking due to fatigue is identified as a Time-Limited Aging Analysis and is analytically addressed in Subsections 4.3.1 and 4.3.4.

Growth of original manufacturing flaws over time by service loading can cause cracking. Detection and evaluation of flaws is important in maintaining the structural integrity of the reactor coolant pump pressure boundary. ASME Section XI inservice examinations of components are intended to detect significant flaw growth and development. These examinations provide assurance that significant flaws do not exist, or a large flaw subject to crack growth would be detected so that it could be characterized, evaluated, and repaired, if necessary. Continued performance of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that flaw growth is managed and that the intended function of the reactor coolant pumps is maintained consistent with the current licensing basis for the period of extended operation.

Stress corrosion cracking is a localized, nonductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate susceptible material from the reactor coolant pumps. In addition, to reduce the susceptibility of reactor coolant pump materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides assurance that stress corrosion cracking is managed and that the intended function of the reactor coolant pumps is maintained consistent with the current licensing basis for the period of extended operation.

3.2.6.2.2 REDUCTION IN FRACTURE TOUGHNESS

Reduction in fracture toughness due to thermal embrittlement is an aging effect requiring management for the period of extended operation. Thermal embrittlement refers to gradual and progressive changes in the microstructure and properties of a material due to exposure to elevated temperatures for an extended period. The only reactor coolant pump components subject to reduction in fracture toughness due to thermal embrittlement are austenitic stainless steel castings. Cast austenitic stainless steel (CASS) Class 1 components at Turkey Point consist of the reactor coolant primary loop elbows, reactor coolant pump casings and closure flanges, and selected valves exceeding a temperature threshold criterion of 482ºF. Reduction in fracture toughness of the reactor coolant CASS primary loop elbows and valves is discussed in Subsection 3.2.1.

Consistent with the conclusions drawn in the NRC draft safety evaluation for WCAP-14575, the CASS reactor coolant pump casings and closure flanges do not require

an aging management program to manage thermal embrittlement beyond the examinations programmatically required by ASME Section XI as modified by Code Case N-481. Accordingly, the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that any reduction in fracture toughness due to thermal aging is managed and that the intended function of the reactor coolant pumps is maintained consistent with the current licensing basis for the period of extended operation.

3.2.6.2.3 LOSS OF MATERIAL

Loss of material is an aging effect requiring management for the period of extended operation. The aging mechanism that can cause loss of material for the reactor coolant pump integral thermal barrier heat exchanger is microbiologically influenced corrosion.

The internal surface of the reactor coolant pump integral thermal barrier heat exchanger tubing is exposed to a treated water environment. Loss of material due to microbiologically influenced corrosion has been identified as an aging effect for the internal surface of the reactor coolant pump integral thermal barrier heat exchanger tubing. The Chemistry Control Program provides assurance that the aging effect of loss of material due to microbiologically influenced corrosion is managed and that the intended function of the reactor coolant pump integral thermal barrier heat exchanger is maintained consistent with the current licensing basis for the period of extended operation.

3.2.6.2.4 LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity can result from stress relaxation and/or aggressive chemical attack.

Loss of mechanical closure integrity due to stress relaxation is a relevant aging effect that requires management. This aging effect can be managed by periodic inservice inspections and leakage testing. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that loss of mechanical closure integrity due to stress relaxation is managed and that the intended function of the reactor coolant pumps is maintained consistent with the current licensing basis for the period of extended operation.

Loss of mechanical closure integrity due to aggressive chemical attack has been observed in the industry and is the most common aging mechanism of concern for

ferritic fasteners of stainless steel components. Mechanical closure bolting associated with the reactor coolant pump components is made of low alloy steel bolting material and is subject to aggressive chemical attack. The Boric Acid Wastage Surveillance Program provides assurance that the aging effect of loss of mechanical closure integrity due to aggressive chemical attack is managed and that the intended function of the reactor coolant pumps is maintained consistent with the current licensing basis for the period of extended operation.

3.2.6.2.5 FOULING

Aging mechanisms that can result in fouling of the reactor coolant pump integral thermal barrier heat exchanger tubing include biological fouling and particulate fouling.

Biological fouling has been identified as an aging effect for tubes exposed to component cooling water. The addition of biocides, in accordance with the Chemistry Control Program, manages this aging effect and provides assurance that the intended function of reactor coolant pump integral thermal barrier heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

Particulate fouling has been identified as an aging effect for heat transfer surfaces of the reactor coolant pump integral thermal barrier heat exchangers. The Chemistry Control Program provides assurance that particulate fouling is managed and that the intended function of the reactor coolant pump integral thermal barrier heat exchangers is maintained consistent with the current licensing basis for the period of extended operation.

3.2.6.3 OPERATING EXPERIENCE

3.2.6.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for reactor coolant pumps operating experience includes the following:

- NRC Bulletin 79-17, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”

- NRC Circular 76-06, “Stress Corrosion Cracks in Stagnant, Low Pressure Stainless Piping Containing Boric Acid Solution at PWRs"
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Information Notice 79-19, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”
- NRC Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting From Boric Acid Corrosion"
- NRC Information Notice 92-86, "Unexpected Restriction to Thermal Growth of Reactor Coolant Piping"
- NRC Information Notice 93-61, "Excessive Reactor Coolant Leakage Following a Seal Failure in a Reactor Coolant Pump or Reactor Recirculation Pump"
- NRC Information Notice 93-84, "Determination of Westinghouse Reactor Coolant Pump Seal Failure"
- NRC Information Notice 93-90, "Unisolatable Reactor Coolant System Leak Following Repeated Application of Leak Sealant"
- NRC Information Notice 97-31, “Failures of Reactor Coolant Pump Thermal Barriers and Check Valves in Foreign Plants”

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.2.6.2. Note that a summary of industry experience associated with reactor coolant pumps is provided in WCAP-14575.

3.2.6.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of reactor coolant pump component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.2.6.2.

3.2.6.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.6.2. Table 3.2-1 contains the results of the aging management review for the reactor coolant pumps and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the reactor coolant pump components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.7 STEAM GENERATORS

The steam generators are within the scope of license renewal, as discussed in Subsection 2.3.1.8. Steam generator components that are subject to an aging management review are listed in Table 3.2-1.

3.2.7.1 MATERIALS AND ENVIRONMENTS

The steam generators are exposed to internal environments of treated water - primary and treated water – secondary, and external environments of containment air and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

The steam generator components are constructed of stainless steel, carbon steel, alloy steel, Alloy 600, and Alloy 690. The steam generator components, their intended functions, the materials, and environments are summarized in Table 3.2-1.

3.2.7.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Table 3.2-1 and are summarized in the following paragraphs.

The following internal and external aging effects require management during the period of extended operation:

- Cracking
- Loss of material
- Loss of mechanical closure integrity

3.2.7.2.1 CRACKING

Cracking due to flaw growth and stress corrosion is an aging effect requiring management for the period of extended operation. At Turkey Point, cracking due to fatigue is identified as a Time-Limited Aging Analysis and is analytically addressed in Subsection 4.3.1. Based on industry and Turkey Point experience with fatigue cracking of transition cone girth welds and feedwater nozzles, additional discussion is provided below.

Growth of original manufacturing flaws over time by service loading can cause cracking. Detection and evaluation of flaws is important in maintaining the structural

integrity of the steam generator pressure boundary. ASME Section XI inservice examinations of components are intended to detect significant flaw growth and development. These examinations provide assurance that significant flaws do not exist, or a large flaw subject to crack growth would be detected so that it could be characterized, evaluated, and repaired, if necessary. Continued performance of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that flaw growth is managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Stress corrosion cracking is a localized, nonductile failure caused by a combination of stress, susceptible material, and an aggressive environment. Specific design, fabrication, and construction measures were taken to minimize or eliminate susceptible material from steam generator components. In addition, to reduce the susceptibility of steam generator materials to stress corrosion cracking, Turkey Point prevents sensitized stainless steels from coming in contact with an aggressive environment. The Chemistry Control Program provides assurance that stress corrosion cracking is managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Industry operating experience has shown transition cone girth welds to be susceptible to cracking due to fatigue. Plant-specific and industry operating experiences have shown the steam generator feedwater nozzles to be susceptible to cracking due to fatigue. Since these particular failure mechanisms have been experienced, aging management of fatigue cracking of the steam generator transition cone girth weld and feedwater nozzle is required for the period of extended operation. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that cracking due to fatigue is managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Industry experience has shown that steam generator tubing is susceptible to primary water stress corrosion cracking (PWSCC) and secondary side intergranular attack (IGA) and intergranular stress corrosion cracking (IGSCC). The existing steam generator tube material, Alloy 600 TT (thermally treated), has an increased resistance to all forms of corrosion as compared to the tubing supplied with the original Turkey Point Units 3 and 4 steam generators. However, Alloy 600 TT tubing in the steam generators remains susceptible to IGA/IGSCC and PWSCC. The aging

effects of IGA/IGSCC and PWSCC can be managed by the continuation of the current steam generator tube inservice inspection program. The Chemistry Control Program and steam generator tube inspections performed in accordance with the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Steam Generator Integrity Program provide assurance that IGA/IGSCC and PWSCC are managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Industry experience has also shown steam generator tube plugs to be susceptible to PWSCC. The root cause of the PWSCC has been attributed to tube plugs fabricated from improperly heat-treated Alloy 600 material. Alloy 600 mechanical tube plugs installed on Turkey Point Units 3 and 4 have been removed and replaced with properly heat-treated Alloy 690 TT tube plugs. These replacement tube plugs have a high resistance to PWSCC. However, since the industry has experienced PWSCC of steam generator tubes, PWSCC has been determined to be an aging effect requiring management. The Steam Generator Integrity Program and the Chemistry Control Program provide assurance that PWSCC is managed and that the intended function of the steam generator tube plugs is maintained consistent with the current licensing basis for the period of extended operation.

3.2.7.2.2 LOSS OF MATERIAL

Loss of material is an aging effect requiring management for the period of extended operation. The aging mechanisms that can cause loss of material for the steam generators are general corrosion, crevice corrosion, pitting corrosion, mechanical wear, and aggressive chemical attack.

General corrosion, pitting corrosion, and crevice corrosion have been identified as aging mechanisms for internal surfaces of carbon steel and low alloy steel components on the steam generator secondary side. General corrosion, pitting corrosion, and crevice corrosion of the secondary side steam generator internal surfaces is mitigated by maintaining adequate secondary side chemistry controls. The Chemistry Control Program provides assurance that general corrosion, pitting corrosion, and crevice corrosion are managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Pitting of the secondary side of the steam generator tubing has occurred at a number of older plants. The location of the pitting is generally in the sludge pile region on the secondary face of the tube sheet. Pitting is not expected to be a significant aging mechanism for the Turkey Point Units 3 and 4 steam generators due to the low amount of copper in the secondary system, careful control of the oxidizing in the secondary water, and the routine removal of tube sheet sludge via lancing. The Chemistry Control Program and Steam Generator Integrity Program provide assurance that pitting corrosion is managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Steam generator tubes in the U-bend region of the tube bundles have shown minor wear at the intersection with the anti-vibration bars. Wear of the steam generator U-tubes is an aging mechanism that requires management. Eddy current examinations, performed in accordance with the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Steam Generator Integrity Program, provide assurance that tube wear is managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Loss of material due to aggressive chemical attack has been identified as an aging mechanism requiring management for external surfaces of carbon steel components exposed to borated water leaks. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of material due to aggressive chemical attack is managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

3.2.7.2.3 LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity can result from stress relaxation and/or aggressive chemical attack.

Loss of mechanical closure integrity due to stress relaxation is a relevant aging effect that requires management. This aging effect can be managed by following the current periodic inservice inspection and leakage testing program. The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides assurance that loss of mechanical closure integrity due to stress relaxation is

managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

Loss of mechanical closure integrity due to aggressive chemical attack has been observed in the industry and is the most common aging mechanism of concern for ferritic fasteners of stainless steel components. Low alloy steel mechanical closure bolting associated with the steam generators and exposed to potential borated water leaks is subject to aggressive chemical attack. The Boric Acid Wastage Surveillance Program provides assurance that the aging mechanism of loss of mechanical closure integrity due to aggressive chemical attack is managed and that the intended function of the steam generators is maintained consistent with the current licensing basis for the period of extended operation.

3.2.7.3 OPERATING EXPERIENCE

3.2.7.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for steam generator operating experience includes the following:

- NRC Bulletin 79-13, “Cracking in Feedwater System Piping”
- NRC Bulletin 88-02, “Rapidly Propagating Cracks in Steam Generator Tubes"
- NRC Bulletin 89-01, “Failure of Westinghouse Steam Generator Tube Mechanical Plugs"
- NRC Generic Letter 82-32, “Potential Steam Generator Related Generic Requirements”
- NRC Generic Letter 85-02, “Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity"
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Generic Letter 95-03, “Circumferential Cracking of Steam Generator Tubes"

- NRC Generic Letter 95-05, “Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"
- NRC Generic Letter 97-06, “Degradation of Steam Generator Internals"
- NRC Information Notice 79-27, “Steam Generator Tube Ruptures at Two Plants"
- NRC Information Notice 82-06, “Failure of Steam Generator Primary Side Manway Closure Studs”
- NRC Information Notice 82-14, “TMI-1 Steam Generator/Reactor Coolant System Chemistry/Corrosion Problem”
- NRC Information Notice 82-37, “Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating Pressurized Water Reactor"
- NRC Information Notice 83-24, “Loose Parts in the Secondary Side of Steam Generators at Pressurized Water Reactors”
- NRC Information Notice 84-18, “Stress Corrosion Cracking in Pressurized Water Reactor Systems"
- NRC Information Notice 84-49, “Intergranular Stress Corrosion Cracking Leading to Steam Generator Tube Failure"
- NRC Information Notice 85-37, “Chemical Cleaning of Steam Generators at Millstone 2"
- NRC Information Notice 85-65, “Crack Growth in Steam Generator Girth Welds"
- NRC Information Notice 88-06, “Foreign Objects in Steam Generators"
- NRC Information Notice 88-31, “Steam Generator Tube Rupture Analysis Deficiency"
- NRC Information Notice 88-99, “Detection and Monitoring of Sudden and/or Rapidly Increasing Primary-to-Secondary Leakage"
- NRC Information Notice 89-33, “Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs"
- NRC Information Notice 89-65, “Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox"

- NRC Information Notice 90-04, “Cracking of the Upper Shell-to-Transition Cone Welds in Steam Generators"
- NRC Information Notice 90-49, “Stress Corrosion Cracking in PWR Steam Generator Tubes"
- NRC Information Notice 91-43, “Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate"
- NRC Information Notice 91-67, “Problems With the Reliable Detection of Intergranular Attack (IGA) of Steam Generator Tubing"
- NRC Information Notice 92-80, “Operation With Steam Generator Tubes Seriously Degraded"
- NRC Information Notice 93-20, “Thermal Fatigue Cracking of Feedwater Piping to Steam Generators"
- NRC Information Notice 93-52, Draft NUREG-1477, “Voltage-Based Interim Plugging Criteria for Steam Generator Tubes"
- NRC Information Notice 93-56, “Weaknesses in Emergency Operating Procedures Found as a Result of Steam Generator Tube Rupture“
- NRC Information Notice 94-05, “Potential Failure of Steam Generator Tubes Sleeved with Kinetically Welded Sleeves"
- NRC Information Notice 94-43, “Determination of Primary-to-Secondary Steam Generator Leak Rate"
- NRC Information Notice 94-62, “Operational Experience on Steam Generator Tube Leaks and Tube Ruptures"
- NRC Information Notice 94-88, “Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes"
- NRC Information Notice 95-40, “Supplemental Information to Generic Letter 95-03, Circumferential Cracking of Steam Generator Tubes"
- NRC Information Notice 96-09, “Damage in Foreign Steam Generator Internals"
- NRC Information Notice 96-09, Supplement 1, “Damage in Foreign Steam Generator Internals"
- NRC Information Notice 96-38, “Results of Steam Generator Tube Examinations"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.2.7.2.

3.2.7.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of steam generator component aging, in addition to interviews with responsible engineering personnel. The Turkey Point Units 3 and 4 steam generators (with the exception of the channel heads and steam domes) were replaced in 1982 and 1983. This replacement was due to significant degradation of the original mill annealed Alloy 600 tubing and deterioration of the carbon steel support plates. As previously discussed, cracking of feedwater nozzles due to fatigue has been experienced at Turkey Point. No additional aging effects requiring management were identified from this review beyond those identified in Subsection 3.2.7.2.

3.2.7.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.2.7.2. Table 3.2-1 contains the results of the aging management review for the steam generators and summarizes the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Steam Generator Integrity Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the steam generator components listed in Table 3.2-1 are maintained consistent with the current licensing basis for the period of extended operation.

3.2.8 REFERENCES

3.2-1 WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," Revision 1, August 1996.

3.2-2 R. A. Newton (WOG) letter to U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated April 18, 1997) for Additional Information on WOG Generic Technical Report WCAP-14575, License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," June 13, 1997.

3.2-3 R. A. Newton (WOG) letter to U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated June 4, 1999) for Additional Information on WOG Generic Technical Reports WCAP-14574, Aging Management Evaluation for Pressurizers, and WCAP-14575, Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," July 19, 1999.

3.2-4 C. I. Grimes (NRC) letter to R. A. Newton (WOG), "Draft Safety Evaluation Concerning Westinghouse Owners Group License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components, WCAP-14575, Revision 1, August 1996," February 10, 2000.

3.2-5 WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," Revision 0, July 1996.

3.2-6 R. A. Newton (WOG) letter to U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated May 6, 1997) for Additional Information on WOG Generic Technical Report WCAP-14574 - License Renewal Evaluation: Aging Management for Pressurizers," May 30, 1997.

3.2-7 WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," Revision 0, June 1997.

3.2-8 R. A. Newton (WOG) to letter U. S. Nuclear Regulatory Commission, "Response to NRC Request (dated June 14, 1999) for Additional Information on WOG Generic Technical Report: WCAP-14577, License Renewal Evaluation: Aging Management for Reactor Vessel Internals," November 24, 1999.

3.2-9 Garner, F.A., L.R. Greenwood, and D. L. Harrod, “Potential High Fluence Response of Pressure Vessel Internals Constructed from Austenitic Stainless Steels,” Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems, August 1993.

3.2-10 Foster, J. P. and A. Boltax, “Correlation of Irradiation Creep Data Obtained in Fast and Thermal Neutron Spectra with Displacement Cross Sections," Journal of Nuclear Materials, No. 89 (1980).

TABLE 3.2-1
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Coolant Piping – Class 1 Components					
Internal Environment					
Valves ≥ 4 inches	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca • Examination Ca
Valves < 4 inches	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca
Piping/fittings ≥ 4 inches	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Reactor Coolant Piping – Class 1 Components (continued)					
Internal Environment (continued)					
Piping/fittings < 4 inches	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C Small Bore Class
Orifices Reducers	Pressure boundary Throttling	Stainless steel	Treated water - primary	Cracking	Chemistry Control
Primary loop elbows	Pressure boundary	Stainless steel - cast	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Reactor Coolant Piping – Class 1 Components (continued)					
Internal Environment (continued)					
Valves ≥ 4 inches	Pressure boundary	Stainless steel - cast	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca • Examination Ca
				Reduction in fracture toughness	ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca • Examination Ca
Valves < 4 inches	Pressure boundary	Stainless steel - cast	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca
				Reduction in fracture toughness	ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Coolant Piping – Class 1 Components (continued)					
External Environment					
Valves Piping/fittings	Pressure boundary	Stainless steel	Containment air	None	None required
Orifices Reducers	Pressure boundary Throttling	Stainless steel	Containment air	None	None required
Primary loop elbows Valves	Pressure boundary	Stainless steel - cast	Containment air	None	None required
Bolting (mechanical closures)	Pressure boundary	Low alloy steel	Borated water leaks	Loss of mechanical closure integrity	Boric Acid Wastage ASME Section XI, S and IWD Inservice I • Examination Ca • Examination Ca • Examination Ca

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Coolant Piping – Non-Class 1 Components					
Internal Environment					
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Treated water – primary	Cracking	Chemistry Control P
Valves Piping/fittings Tubing/fittings Flexible hoses Filters Vessels	Pressure boundary	Stainless steel	Air/Gas	None	None required
Reactor coolant pump motor upper bearing oil heat exchanger channels and covers	Pressure boundary	Carbon steel	Treated water	Loss of material	Chemistry Control P Galvanic Corrosion Program
Reactor coolant pump motor upper bearing oil heat exchanger tube sheets	Pressure boundary	Carbon steel	Treated water	Loss of material	Chemistry Control P
			Lubricating oil	None	None required

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Coolant Piping – Non-Class 1 Components (continued)					
Internal Environment (continued)					
Reactor coolant pump motor upper bearing oil heat exchanger tubing and lower bearing oil cooling coils	Pressure boundary [1]	Admiralty brass 90/10 CuNi	Treated water (inside diameter)	Loss of material	Chemistry Control P
			Lubricating oil (outside diameter)	None	None required
Piping/fittings associated with reactor coolant pump oil coolers	Pressure boundary	Carbon steel	Treated water	Loss of material	Chemistry Control P Galvanic Corrosion Program

NOTE: 1. Heat transfer is not a license renewal intended function for these components. These items are within the scope ensure the integrity of the Component Cooling Water System pressure boundary.

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Coolant Piping – Non-Class 1 Components (continued)					
External Environment					
Valves Piping/fittings Tubing/fittings Flexible hoses Filters Vessels	Pressure boundary	Stainless steel	Containment air	None	None required
Reactor coolant pump motor upper bearing oil heat exchanger channels and covers	Pressure boundary	Carbon steel	Containment air	Loss of material	Systems and Struct
			Borated water leaks	Loss of material	Boric Acid Wastage
Piping/fittings associated with reactor coolant pump oil coolers	Pressure boundary	Carbon steel	Containment air	Loss of material	Systems and Struct
			Borated water leaks	Loss of material	Boric Acid Wastage
Bolting (mechanical closures)	Pressure boundary	Low alloy steel	Borated water leaks	Loss of mechanical closure integrity	Boric Acid Wastage

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	
Regenerative and Excess Letdown Heat Exchangers					
Internal Environment					
Excess letdown heat exchanger shells, shell heads, shell nozzles, and shell flanges	Pressure boundary	Carbon steel	Treated water	Loss of material	Chemistry Co Galvanic Cor Program
Excess letdown heat exchanger channels and heads, nozzles, divider plates	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Co
Excess letdown heat exchanger tubes	Pressure boundary Heat transfer	Stainless steel	Treated water – primary (inside diameter)	Cracking Fouling	Chemistry Co
			Treated water (outside diameter)	Loss of material Fouling	Chemistry Co
Excess letdown heat exchanger tube sheets	Pressure boundary	Stainless steel	Treated water – primary (tube side)	Cracking	Chemistry Co
			Treated water (shell side)	Loss of material	Chemistry Co
Regenerative heat exchangers	Pressure boundary Heat transfer	Stainless steel	Treated water – primary (shell and tube sides)	Fouling Cracking	Chemistry Co

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Regenerative and Excess Letdown Heat Exchangers (continued)					
External Environment					
Excess letdown heat exchanger shells, shell heads, shell nozzles, and shell flanges	Pressure boundary	Carbon steel	Containment air (wetted with condensation)	Loss of material	Systems and Stru
			Borated water leaks	Loss of material	Boric Acid Wastag
Excess letdown heat exchanger channels, channel heads, and nozzles	Pressure boundary	Stainless steel	Containment air	None	None required
Regenerative heat exchangers	Pressure boundary	Stainless steel	Containment air	None	None required
Bolting (mechanical closures)	Pressure boundary	Low alloy steel	Borated water leaks	Loss of mechanical closure integrity	Boric Acid Wastag

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Pressurizers					
Internal Environment					
Upper heads Lower heads	Pressure boundary	Carbon steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Surge nozzles Spray nozzles Safety nozzles Relief nozzles	Pressure boundary	Carbon steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Surge nozzle safe ends Spray nozzle safe ends Safety nozzle safe ends Relief nozzle safe ends	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Surge nozzle thermal sleeves Spray nozzle thermal sleeves	Pressure boundary[1]	Stainless steel	Treated water - primary	Cracking	Chemistry Control

NOTE: 1. The thermal sleeves are not part of the pressure boundary, but do provide thermal shielding to minimize nozzle l fatigue.

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Pressurizers (continued)					
Internal Environment (continued)					
Heater wells Heater sheaths	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Shells	Pressure boundary	Alloy steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Instrument nozzles Thermowells	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C
Manway covers	Pressure boundary	Carbon steel w/ stainless steel disc insert	Treated water - primary	Cracking	Chemistry Control

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	F
Pressurizers (continued)					
External Environment					
Upper heads Lower heads	Pressure boundary	Carbon steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wa
Surge nozzles Spray nozzles Safety nozzles Relief nozzles	Pressure boundary	Carbon steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wa
Surge nozzle safe ends Spray nozzle safe ends Safety nozzle safe ends Relief nozzle safe ends	Pressure boundary	Stainless steel	Containment air	None	None required
Heater wells Heater sheaths	Pressure boundary	Stainless steel	Containment air	None	None required
Support skirts and flanges	Structural support	Carbon steel	Containment air	Cracking	ASME Section and IWD Inser • Examinati
			Borated water leaks	Loss of material	Boric Acid Wa

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Pressurizers (continued)					
External Environment (continued)					
Shells	Pressure boundary	Alloy steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wastag
Instrument nozzles Thermowells	Pressure boundary	Stainless steel	Containment air	None	None required
Manway covers	Pressure boundary	Carbon steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wastag
Manway cover bolts	Pressure boundary	Low alloy steel	Borated water leaks	Loss of mechanical closure integrity	Boric Acid Wastag ASME Section XI, and IWD Inservice • Examination C • Examination C

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Reactor Vessels					
Internal Environment					
Closure head domes Closure head flanges	Pressure boundary	Low alloy steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Upper shell flanges	Pressure boundary Support reactor vessel internals	Low alloy steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Upper shells	Pressure boundary	Low alloy steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Reactor Vessels (continued)					
Internal Environment (continued)					
Primary inlet nozzles Primary outlet nozzles	Pressure boundary	Low alloy steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca
Primary nozzle safe ends	Pressure boundary	Stainless steel weld butter	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca
Intermediate shells Lower shells Circumferential welds	Pressure boundary	Low alloy steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control I ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca
				Reduction in fracture toughness	Reactor Vessel Inte

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Reactor Vessels (continued)					
Internal Environment (continued)					
Bottom head toruses Bottom head domes	Pressure boundary Support reactor vessel internals	Low alloy steel w/ stainless steel cladding	Treated water - primary	Cracking	Chemistry Control ASME Section XI, IWD Inservice Insp • Examination C • Examination C
Control rod drive mechanism rod travel housings Control rod drive mechanism latch housings Control rod drive mechanism housing flanges	Pressure boundary	Stainless steel	Treated water - primary	Cracking	Chemistry Control
				Loss of material	ASME Section XI, IWD Inservice Insp • Examination C Boric Acid Wastag
Control rod drive mechanism housing tubes (head adapters)	Pressure boundary	Alloy 600	Treated water - primary	Cracking	Chemistry Control Reactor Vessel He Inspection Progra ASME Section XI, IWD Inservice Insp • Examination C • Examination C • Examination C

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Reactor Vessels (continued)					
Internal Environment (continued)					
Head vent pipes	Pressure boundary	Alloy 600 Stainless steel	Treated water - primary	Cracking	Chemistry Control ASME Section XI, IWD Inservice Insp • Examination C • Examination C
O-ring leak monitor tubes	Pressure boundary	Stainless steel	Air/Gas	None	None required
Core support lugs	Support reactor vessel internals	Alloy 600	Treated water - primary	Cracking	Chemistry Control ASME Section XI, IWD Inservice Insp • Examination C
				Loss of material	ASME Section XI, IWD Inservice Insp • Examination C
Instrumentation tubes Instrumentation tube safe ends	Pressure boundary Support thimble tubes	Alloy 600 Stainless steel	Treated water - primary	Cracking	Chemistry Control ASME Section XI, IWD Inservice Insp • Examination C • Examination C

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Reactor Vessels (continued)					
Internal Environment (continued)					
Bottom mounted instrumentation guide tubes	Pressure boundary Support thimble tubes	Stainless steel	Treated water - primary	Cracking	Chemistry Control ASME Section XI, IWD Inservice Insp • Examination C
Seal table fittings	Pressure boundary Support thimble tubes	Stainless steel	Treated water - primary	Cracking	Chemistry Control

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Vessels (continued)					
External Environment					
Closure head domes (includes lifting lugs) Closure head flanges Upper shells Primary inlet nozzles Primary outlet nozzles Intermediate shells Lower shells	Pressure boundary	Low alloy steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wastage
Upper shell flanges	Pressure boundary Support reactor vessel internals	Low alloy steel	Containment air	Loss of material	ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca
			Borated water leaks	Loss of material	Boric Acid Wastage
Refueling seal ledges	Structural support [1]	Carbon steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wastage
Primary nozzle safe ends	Pressure boundary	Stainless steel weld butter	Containment air	None	None required
Nozzle support pads	Structural support	Low alloy steel weld build-up	Containment air	None	None required

NOTE: 1. Provides structural support for refueling cavity seal ring.

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Vessels (continued)					
External Environment (continued)					
Bottom head toruses Bottom head domes	Pressure boundary Support reactor vessel internals	Low alloy steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wastage
Control rod drive mechanism rod travel housings Control rod drive mechanism latch housings Control rod drive mechanism housing flanges	Pressure boundary	Stainless steel	Containment air	None	None required
Control rod drive mechanism housing tubes (head adapters)	Pressure boundary	Alloy 600	Containment air	None	None required
Control rod drive mechanism ventilation shroud support rings	Structural support [1]	Carbon steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wastage
Head vent pipes	Pressure boundary	Alloy 600 Stainless steel	Containment air	None	None required

NOTE: 1. Provides structural support to control rod drive mechanism cooling credited for station blackout

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Reactor Vessels (continued)					
External Environment (continued)					
O-ring leak monitor tubes	Pressure boundary	Stainless steel	Containment air	None	None required
Instrumentation tubes Instrumentation tube safe ends	Pressure boundary Support thimble tubes	Alloy 600 Stainless steel	Containment air	None	None required
Bottom mounted instrumentation guide tubes	Pressure boundary Support thimble tubes	Stainless steel	Containment air	Cracking	Boric Acid Wastag
Bottom mounted instrumentation flux thimble tubes[1]	Pressure boundary Support thimble tubes	Stainless steel	Treated water - primary	Cracking	Chemistry Control
				Loss of material	Thimble Tube Insp
Seal tables	Support thimble tubes	Stainless steel	Containment air	None	None required
Seal table fittings	Pressure boundary Support thimble tubes	Stainless steel	Containment air	None	None required

NOTE: 1. Fretting wear of thimble tubes was identified as a TLAA (Subsection 4.7.1). Option iii of 10 CFR 54.21(c)(1) was
aging effect.

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Reactor Vessels (continued)					
External Environment (continued)					
Closure studs, nuts, and washers	Pressure boundary	Low alloy steel	Containment air	Loss of material Loss of mechanical closure integrity	ASME Section XI and IWD Inservic • Examination • Examination
			Borated water leaks	Loss of mechanical closure integrity Cracking	Boric Acid Wasta ASME Section XI and IWD Inservic • Examination • Examination

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management[1]	Prog
Reactor Vessel Internals					
Lower core plates and fuel pins Core barrels and flanges	Core support Flow distribution	Stainless steel	Treated water - primary	Cracking	Chemistry Control
				Cracking Reduction in fracture toughness	ASME Section XI, and IWD Inservice • Examination C Reactor Vessel Int
				Loss of material (fuel pins and flanges only)	ASME Section XI, and IWD Inservice • Examination C
Lower support columns	Core support	Stainless steel	Treated water - primary	Cracking	Chemistry Control
				Cracking Reduction in fracture toughness	ASME Section XI, and IWD Inservice • Examination C Reactor Vessel Int
Radial keys and clevis inserts	Core support	Stainless steel (radial keys) Alloy 600 (clevis inserts)	Treated water - primary	Cracking	Chemistry Control
				Loss of material	ASME Section XI, and IWD Inservice • Examination C

NOTE: 1. The reactor vessel internals parts requiring management for dimensional changes due to void swelling (if any) h determined.

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Reactor Vessel Internals (continued)					
Upper core plate alignment pins	Guide and support rod control cluster assemblies (RCCAs)	Stainless steel	Treated water – primary	Cracking	Chemistry Control
				Loss of material	ASME Section XI, IWD Inservice Insp • Examination C
Baffle and former assemblies	Core support Flow distribution	Stainless steel	Treated water – primary	Cracking	Chemistry Control
				Cracking Reduction in fracture toughness	ASME Section XI, IWD Inservice Insp • Examination C Reactor Vessel Int
Core barrel outlet nozzles Diffusers	Flow distribution	Stainless steel	Treated water – primary	Cracking	Chemistry Control
Upper support plates Upper support columns	Guide and support RCCAs	Stainless steel	Treated water – primary	Cracking	Chemistry Control
Secondary core support Upper core plate	Core support Flow distribution	Stainless steel	Treated water – primary	Cracking	Chemistry Control
Head/vessel alignment pins	Core support	Stainless steel	Treated water – primary	Cracking	Chemistry Control
Guide tubes and guide pins	Guide and support RCCAs	Stainless steel Alloy X-750 (pins)	Treated water – primary	Loss of material	ASME Section XI, IWD Inservice Insp • Examination C
				Cracking	Chemistry Control

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Reactor Vessel Internals (continued)					
Internals holddown springs	Core support	Stainless steel	Treated water – primary	Loss of preload	ASME Section XI, IWD Inservice Insp • Examination C
				Cracking	Chemistry Control
Thermal shields	Shield vessel	Stainless steel	Treated water – primary	Cracking	Chemistry Control
				Cracking Reduction in fracture toughness	ASME Section XI, IWD Inservice Insp • Examination C Reactor Vessel Int
Bottom mounted instrumentation columns Upper instrumentation columns	Guide and support instrumentation	Stainless steel	Treated water – primary	Cracking	Chemistry Control
Head cooling spray nozzles	Flow distribution	Stainless steel	Treated water – primary	Cracking	Chemistry Control
Lower support castings	Core support Flow distribution	Stainless steel – cast	Treated water – primary	Cracking	Chemistry Control
				Reduction in fracture toughness	ASME Section XI, IWD Inservice Insp • Examination C Reactor Vessel Int

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Pro
Reactor Vessel Internals (continued)					
Bolting (upper support column, guide tube, clevis insert)	Core support	Stainless steel Alloy X-750 (clevis insert bolts)	Treated water – primary	Loss of mechanical closure integrity	Chemistry Control ASME Section XI, IWD Inservice Insp • Examination C
Bolting (lower support column, baffle/former, barrel/former)	Core support	Stainless steel	Treated water – primary	Loss of mechanical closure integrity	Chemistry Control ASME Section XI, IWD Inservice Insp • Examination C Reactor Vessel Int

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Program
Reactor Coolant Pumps					
Internal Environment					
Reactor coolant pumps	Pressure boundary (casing and main flange)	Stainless steel – cast	Treated water – primary	Cracking	Chemistry Control Pro ASME Section XI, Sub and IWD Inservice Ins • Examination Cate by Code Case N-4 • Examination Categ • Examination Categ
				Reduction in fracture toughness	ASME Section XI, Sub and IWD Inservice Ins • Examination Cate by Code Case N-4 • Examination Categ
	Pressure boundary (thermal barrier flanges)	Stainless steel	Treated water – primary	Cracking	Chemistry Control Pro ASME Section XI, Sub and IWD Inservice Ins • Examination Categ

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Program
Reactor Coolant Pumps (continued)					
Internal Environment (continued)					
Thermal barrier heat exchanger tubes	Pressure boundary Heat transfer	Stainless steel	Treated water – primary (outside diameter)	Fouling Cracking	Chemistry Control Pro
			Treated water (inside diameter)	Fouling Loss of material	
Thermal barrier heat exchanger nozzles	Pressure boundary	Stainless steel	Treated water	Loss of material	Chemistry Control Pro

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Reactor Coolant Pumps (continued)					
External Environment					
Reactor coolant pumps Thermal barrier heat exchanger nozzles	Pressure boundary	Stainless steel – cast Stainless steel	Containment air	None	None required
Reactor coolant pump lugs	Structural support	Stainless steel	Containment air	Cracking	ASME Section XI, S and IWD Inservice • Examination C Case N-509
Bolting (mechanical closures)	Pressure boundary	Low alloy steel	Borated water leaks	Loss of mechanical closure integrity	Boric Acid Wastage ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca • Examination Ca

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Steam Generators					
Internal Environment					
Channel heads	Pressure boundary	Carbon steel w/ stainless steel cladding	Treated water – primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Primary inlet and outlet nozzles	Pressure boundary	Carbon steel w/ stainless steel cladding	Treated water – primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C
Primary inlet and outlet nozzle safe ends	Pressure boundary	Stainless steel weld butter	Treated water – primary	Cracking	Chemistry Control ASME Section XI, and IWD Inservice • Examination C • Examination C

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Steam Generators (continued)					
Internal Environment (continued)					
Tube sheets	Pressure boundary	Alloy steel w/ Alloy 600 cladding (primary side)	Treated water – primary	Cracking	Chemistry Control F ASME Section XI, S and IWD Inservice I • Examination Ca • Examination Ca
		Alloy steel (secondary side)	Treated water – secondary	Cracking	ASME Section XI, S and IWD Inservice I • Examination Ca • Examination Ca
				Loss of material	Chemistry Control F

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Steam Generators (continued)					
Internal Environment (continued)					
U-tubes	Pressure boundary Heat transfer	Alloy 600 TT	Treated water – primary (inside diameter)	Cracking	Chemistry Control I Steam Generator In ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca
			Treated water – secondary (outside diameter)	Cracking Loss of material	Chemistry Control I Steam Generator In ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca
Divider plates	Flow distribution	Alloy 600	Treated water - primary	Cracking	Chemistry Control I
Steam generator tube plugs	Pressure boundary	Alloy 690 TT	Treated water - primary	Cracking	Chemistry Control I Steam Generator In
Primary manways	Pressure boundary	Carbon steel with stainless steel disc inserts	Treated water - primary	Cracking	Chemistry Control I

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Steam Generators (continued)					
Internal Environment (continued)					
Upper and lower shells Elliptical heads Transition cones	Pressure boundary	Alloy steel	Treated water - secondary	Cracking	ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca
				Loss of material	Chemistry Control I
Feedwater nozzles Steam outlet nozzles	Pressure boundary	Alloy steel	Treated water - secondary	Cracking	ASME Section XI, S and IWD Inservice • Examination Ca • Examination Ca
				Loss of material	Chemistry Control I
Steam flow limiters	Throttling	Alloy steel	Treated water - secondary	Loss of material	Chemistry Control I
Blowdown nozzles and secondary side shell penetrations	Pressure boundary	Alloy steel	Treated water - secondary	Cracking	ASME Section XI, S and IWD Inservice • Examination Ca
				Loss of material	Chemistry Control I
Secondary closure covers	Pressure boundary	Carbon steel	Treated water - secondary	Loss of material	Chemistry Control I
Tube bundle wrappers Wrapper support systems	Structural support	Carbon steel	Treated water - secondary	Loss of material	Chemistry Control I

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Progr
Steam Generators (continued)					
Internal Environment (continued)					
Tube support plates	Structural support	Stainless steel	Treated water - secondary	Cracking	Chemistry Control I
Anti-vibration bars	Structural support	Chrome-plated Alloy 600	Treated water - secondary	Cracking	Chemistry Control I

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Steam Generators (continued)					
External Environment					
Channel heads Primary manways Primary inlet and outlet nozzles	Pressure boundary	Carbon steel	Containment air	None	None required
			Borated water leaks	Loss of material	Boric Acid Wastag
Primary inlet and outlet nozzle safe ends	Pressure boundary	Stainless steel weld butter	Containment air	None	None required
Upper and lower shells Elliptical heads Transition cones Feedwater nozzles Steam outlet nozzles	Pressure boundary	Alloy steel	Containment air	None	None required
Secondary closure covers	Pressure boundary	Carbon steel	Containment air	None	None required
Blowdown piping nozzles and secondary side shell penetrations	Pressure boundary	Alloy steel	Containment air	None	None required

TABLE 3.2-1 (continued)
REACTOR COOLANT SYSTEMS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Prog
Steam Generators (continued)					
External Environment (continued)					
Support pads	Structural support	Carbon steel	Containment air	Cracking	ASME Section XI, S IWD Inservice Insp • Examination Ca
			Borated water leaks	Loss of material	Boric Acid Wastage
Seismic lugs	Structural support	Alloy steel	Containment air	Cracking	ASME Section XI, S IWD Inservice Insp • Examination Ca
Bolting (mechanical closures – primary)	Pressure boundary	Low alloy steel	Borated water leaks	Loss of mechanical closure integrity	Boric Acid Wastage ASME Section XI, S IWD Inservice Insp • Examination Ca • Examination Ca
Bolting (mechanical closures – secondary)	Pressure boundary	Low alloy steel	Containment air	Loss of mechanical closure integrity	ASME Section XI, S IWD Inservice Insp • Examination Ca

3.3 ENGINEERED SAFETY FEATURES SYSTEMS

The following systems are included in this section:

- Emergency Containment Cooling
- Containment Spray
- Containment Isolation
- Safety Injection
- Residual Heat Removal
- Emergency Containment Filtration
- Containment Post Accident Monitoring and Control

Subsection 2.3.2 provides a description of these systems and identifies the components requiring an aging management review for license renewal. For the Engineered Safety Features Systems, the specific materials and environments, the resulting aging effects, and the specific programs to manage these aging effects are listed in Tables 3.3-1 through 3.3-7. Appendix C contains the process that identified the aging effects requiring management for non-Class 1 components.

3.3.1 MATERIALS AND ENVIRONMENTS

The Engineered Safety Features Systems are exposed to internal environments of treated water - borated, treated water, lubricating oil, and air/gas; and external environments of outdoor, indoor - not air-conditioned, containment air, embedded/encased, and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

The only parts of systems or components considered to be inaccessible for inspection are those that are buried or encased in concrete. These environments are addressed as part of the aging management review process; see Table 3.0-2, “External Service Environments." Potential aging effects associated with these environments are reviewed and those aging effects requiring management are identified along with the credited aging management program(s). All other parts of systems and components can be accessed, if required. The only Engineered Safety Features System containing inaccessible parts is Residual Heat Removal. This system contains some stainless steel piping that is embedded/encased in concrete.

The tanks, pumps, heat exchangers, piping, tubing, and associated components and commodity groups for these systems are constructed of stainless and carbon steels, cast iron, Inconel, brass, bronze, copper, and aluminum. The components and commodity groups, their intended functions, the materials, and environments for the Engineered Safety Features Systems are summarized in Tables 3.3-1 through 3.3-7.

3.3.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Tables 3.3-1 through 3.3-7. The aging effects requiring management for each system are summarized in the following paragraphs.

Emergency Containment Cooling - The aging effects requiring management are loss of material for carbon steel components and loss of material and fouling for admiralty brass heat exchanger tubing. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Containment Spray - The aging effects requiring management are loss of material for carbon steel, stainless steel, cast iron, and brass components; and loss of material and fouling for stainless steel heat exchanger tubing. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Containment Isolation - The aging effect requiring management is loss of material for carbon steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Safety Injection - The aging effects requiring management are loss of material for carbon steel, stainless steel, brass, and cast iron components; cracking for certain stainless steel components; and loss of material and fouling for Inconel heat exchanger tubing and gray cast iron thrust bearing coolers. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Residual Heat Removal - The aging effects requiring management are loss of material for carbon steel and stainless steel components; cracking for certain stainless steel components; and loss of material, cracking, and fouling for stainless steel heat exchanger tubing. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Emergency Containment Filtration - The aging effect requiring management is loss of material for carbon steel and stainless steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Containment Post Accident Monitoring and Control - The aging effect requiring management is loss of material for carbon steel and stainless steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

3.3.3 OPERATING EXPERIENCE

3.3.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to Engineered Safety Features Systems includes the following:

- NRC Bulletin 79-03, "Longitudinal Weld Defects in ASME SA-312 Type 304 Stainless Steel Pipe"
- NRC Bulletin 79-17, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”
- NRC Bulletin 80-24, “Prevention of Damage Due to Water Leakage Inside Containment"
- NRC Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants"
- NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems”
- NRC Bulletin 89-02, “Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design”
- NRC Bulletin 93-02, “Debris Plugging of Emergency Core Cooling Suction Strainers"
- NRC IE Circular 76-06, “Stress Corrosion Cracks in Stagnant, Low Pressure Stainless Piping Containing Boric Acid Solution at PWRs"
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Generic Letter 91-17, "Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants"
- NRC Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"

- NRC Information Notice 79-19, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”
- NRC Information Notice 80-05, "Chloride Contamination of Safety Related Piping and Components"
- NRC Information Notice 80-15, “Axial (Longitudinal) Oriented Cracking in Piping”
- NRC Information Notice 80-40, “Excessive Nitrogen Supply Pressure Actuates Safety-Relief Valve Operation to Cause Reactor Depressurization"
- NRC Information Notice 83-46, "Common-Mode Valve Failures Degrade Surry's Recirculation Spray Subsystem"
- NRC Information Notice 84-18, “Stress Corrosion Cracking in Pressurized Water Reactor Systems”
- NRC Information Notice 84-89, “Stress Corrosion Cracking in Non-Sensitized 316 Stainless Steel”
- NRC Information Notice 85-34, "Heat Tracing Contributes to Corrosion Failure of Stainless Steel Piping"
- NRC Information Notice 85-59, “Valve Stem Corrosion Failures”
- NRC Information Notice 88-37, Flow Blockage of Cooling Water to Safety System Components"
- NRC Information Notice 89-30, "High Temperature Environments at Nuclear Power Plants"
- NRC Information Notice 89-77, "Debris in Containment Emergency Sumps and Incorrect Screen Configuration"
- NRC Information Notice 89-80, "Potential for Water Hammer, Thermal Stratification, and Steam Binding in High Pressure Coolant Injection Piping"
- NRC Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600"
- NRC Information Notice 90-65, "Recent Orifice Plate Problems"
- NRC Information Notice 91-05, “Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles”

- NRC Information Notice 94-63, " Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks"
- NRC Information Notice 95-26, “Defect in Safety-Related Pump Parts due to Inadequate Heat Treatment”
- NRC Information Notice 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillance"
- NRC Information Notice 96-27, “Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation”
- NRC Information Notice 97-13, "Deficient Conditions Associated with Protective Coatings at Nuclear Power Plants"
- NRC Information Notice 97-46, “Unisolable Crack in High Pressure Injection Piping (Oconee)”

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.3.2.

3.3.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of Engineered Safety Features Systems component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.3.2.

3.3.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.3.2. Tables 3.3-1 through 3.3-7 contain the results of the aging management review for the Engineered Safety Features Systems and summarize the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Containment Spray System Piping Inspection Program
- Emergency Containment Cooler Inspection
- Field Erected Tanks Internal Inspection
- Galvanic Corrosion Susceptibility Inspection Program
- Periodic Surveillance and Preventive Maintenance Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the Engineered Safety Features Systems components listed in Tables 3.3-1 through 3.3-7 are maintained consistent with the current licensing basis for the period of extended operation.

TABLE 3.3-1
EMERGENCY CONTAINMENT COOLING

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment				
Emergency containment cooler headers	Pressure boundary	Carbon steel	Treated water	Loss of material
Emergency containment cooler tubes (inside diameter)	Pressure boundary Heat transfer	Admiralty brass	Treated water	Loss of material[1]
				Fouling
Emergency containment cooler housings	Pressure boundary	Carbon steel	Air/Gas	Loss of material

NOTE: 1. Emergency containment cooler tube wear was identified as a TLAA (see Subsection 4.7.2). Option (iii) of 10 CF
selected to address this aging effect.

TABLE 3.3-1 (continued)
EMERGENCY CONTAINMENT COOLING

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment				
Emergency containment cooler headers	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Emergency containment cooler tubes (outside diameter)	Pressure boundary	Admiralty brass	Containment air	None
Emergency containment cooler housings	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.3-2
CONTAINMENT SPRAY

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment				
Containment spray pumps	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
Containment spray pump seal water heat exchanger shells and covers	Pressure boundary	Cast iron	Treated water - borated	Loss of material
Containment spray pump seal water heat exchanger tube shields	Pump seal cooling	Brass	Treated water - borated	Loss of material
Containment spray pump seal water heat exchanger tubes (outside diameter), tube coil bands and clips.	Pressure boundary Heat transfer	Stainless steel	Treated water	Loss of material Fouling
Containment spray pump seal water heat exchanger tubes (inside diameter)	Pressure boundary Heat transfer	Stainless steel	Treated water - borated	Loss of material Fouling

TABLE 3.3-2 (continued)
CONTAINMENT SPRAY

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment (continued)				
Containment spray pump seal water cyclone separators	Pressure boundary Filtering	Stainless steel	Treated water - borated	Loss of material
Valves Piping/ fittings Tubing/ fittings (upstream of MOV-3/4-880A and -880B)	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
Valves Piping/ fittings Tubing/ fittings (downstream of MOV-3/4-880A and -880B to containment penetrations)	Pressure boundary	Stainless steel	Air/Gas Treated water - borated	Loss of material
Valves Piping/fittings (downstream of containment penetrations to Containment elevation 65')	Pressure boundary	Carbon steel	Air/Gas Treated water - borated	Loss of material

TABLE 3.3-2 (continued)
CONTAINMENT SPRAY

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment (continued)				
Valves Piping/fittings (above Containment elevation 65')	Pressure boundary	Carbon steel	Air/Gas	None
Spray nozzles	Pressure boundary Spray	Bronze	Air/Gas	None
Orifices	Pressure boundary Throttling	Stainless steel	Treated water - borated	Loss of material

TABLE 3.3-2 (continued)
CONTAINMENT SPRAY

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment				
Containment spray pumps	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Containment spray pump seal water heat exchanger shells and covers	Pressure boundary	Cast iron	Indoor – not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Containment spray pump seal water heat exchanger flex fittings	Pressure boundary	Brass	Indoor – not air-conditioned	None
Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings Tubing/fittings Filters	Pressure boundary	Stainless steel	Indoor – not air-conditioned	None
Spray nozzles	Pressure boundary Spray	Bronze	Containment air	None

TABLE 3.3-2 (continued)
CONTAINMENT SPRAY

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment (continued)				
Orifices	Pressure boundary Throttling	Stainless steel	Containment air	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.3-3
CONTAINMENT ISOLATION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management	
Containment Purge Systems					
Internal Environment					
Valves Piping/fittings	Pressure boundary	Carbon steel	Air/Gas	None	N
Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None	N
Debris screen gratings	Filtration	Carbon steel - galvanized	Air/Gas	None	N
Debris screen banding	Filtration	Stainless steel	Air/Gas	None	N
External Environment					
Valves Piping/fittings	Pressure boundary	Carbon steel	Outdoor	Loss of material	S S F
Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air	Loss of material	S S F
			Borated water leaks	Loss of material	E S
Tubing/fittings	Pressure boundary	Stainless steel	Outdoor Containment air	None	N
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity	E S

TABLE 3.3-3 (continued)
CONTAINMENT ISOLATION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Breathing Air Systems				
Internal Environment				
Valves Piping/fittings	Pressure boundary	Stainless steel	Air/Gas	None
External Environment				
Valves Piping/fittings	Pressure boundary	Stainless steel	Containment air Indoor - not air-conditioned,	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.3-3 (continued)
CONTAINMENT ISOLATION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Nitrogen and Hydrogen Systems				
Internal Environment				
Valves Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None
Valves Piping/fittings	Pressure boundary	Carbon steel	Air/Gas	None
External Environment				
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Containment air Indoor - not air-conditioned	None
Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air Indoor - not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.3-4
SAFETY INJECTION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment				
Refueling water storage tanks	Pressure boundary	Carbon steel	Air/Gas	Loss of material
			Treated water - borated	Loss of material
Accumulators	Pressure boundary	Stainless steel (cladding)	Treated water - borated	Loss of material
			Air/Gas	None
Safety injection pumps	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
Safety injection pump thrust bearing coolers	Pressure boundary Heat transfer	Gray cast iron	Treated water - borated	Loss of material Fouling
			Lubricating oil	None
Safety injection pump shaft seal heat exchanger tubes (inside diameter)	Pressure boundary Heat transfer	Inconel	Treated water - borated	Loss of material Fouling
Safety injection pump shaft seal heat exchanger tubes (outside diameter)	Pressure boundary Heat transfer	Inconel	Treated water	Loss of material Fouling

TABLE 3.3-4 (continued)
SAFETY INJECTION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment (continued)				
Safety injection pump shaft seal heat exchanger tube shields	Pressure boundary	Brass	Treated water - borated	Loss of material
Safety injection pump shaft seal heat exchanger shells and covers	Pressure boundary	Cast iron	Treated water	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
Flow elements Orifices	Pressure boundary Throttling	Stainless steel	Treated water - borated	Loss of material

TABLE 3.3-4 (continued)
SAFETY INJECTION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment				
Refueling water storage tanks	Pressure boundary	Carbon steel	Outdoor	Loss of material
			Borated water leaks	Loss of material
Accumulators	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Safety injection pumps	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Safety injection pump thrust bearing coolers	Pressure boundary	Gray cast iron	Indoor – not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Safety injection pump shaft seal heat exchanger shells and covers	Pressure boundary	Cast iron	Indoor – not air-conditioned	Loss of material
			Borated water leaks	Loss of material

TABLE 3.3-4 (continued)
SAFETY INJECTION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment (continued)				
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Indoor – not air-conditioned Containment air	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Outdoor	None
Piping/fittings (large bore, thin wall outdoors, and outdoors in trenches)	Pressure boundary	Stainless steel	Outdoor	Cracking [1]
Flow elements Orifices	Pressure boundary Throttling	Stainless steel	Indoor – not air-conditioned Containment air	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

NOTE: 1. Plant experience has identified the potential for cracking in non-stress relieved heat affected zones of weld joints

TABLE 3.3-5
RESIDUAL HEAT REMOVAL

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment				
Residual heat removal pumps	Pressure boundary	Stainless steel	Treated water - borated	Loss of material Cracking[1]
Residual heat removal heat exchanger shells and baffles	Pressure boundary	Carbon steel	Treated water	Loss of material
Residual heat removal heat exchanger tubes	Pressure boundary Heat transfer	Stainless steel	Treated water – borated (inside diameter)	Loss of material Cracking[1] Fouling
			Treated water (outside diameter)	Loss of material Fouling
Residual heat removal heat exchanger tube sheets	Pressure boundary	Stainless steel	Treated water – borated (tubes side)	Loss of material Cracking[1]
			Treated water (shell side)	Loss of material
Residual heat removal pump seal water heat exchanger shells, covers, and baffles	Pressure boundary	Carbon steel	Treated water	Loss of material

NOTE: 1. Stress corrosion cracking is an aging effect requiring management for stainless steel at temperatures >140°F.

TABLE 3.3-5 (continued)
RESIDUAL HEAT REMOVAL

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment (continued)				
Residual heat removal pump seal water heat exchanger tubes (outside diameter)	Pressure boundary Heat transfer	Stainless steel	Treated water	Loss of material Fouling
Residual heat removal pump seal water heat exchanger tubes (inside diameter)	Pressure boundary Heat transfer	Stainless steel	Treated water - borated	Loss of material Cracking[1] Fouling
Valves Piping/fittings Tubing/fittings Thermowells	Pressure boundary	Stainless steel	Treated water - borated	Loss of material Cracking[1]
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None
Check valve 3-753A	Pressure boundary	Stainless steel	Treated water - borated	Loss of material Cracking[1] Cracking[2]
Flow elements Orifices	Pressure boundary Throttling	Stainless steel	Treated water - borated	Loss of material Cracking[1]

NOTES: 1. Stress corrosion cracking is an aging effect requiring management for stainless steel at temperatures >140°F.
2. Monitoring of existing flaw is required.

TABLE 3.3-5 (continued)
RESIDUAL HEAT REMOVAL

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment				
Residual heat removal pumps	Pressure boundary	Stainless steel	Indoor – not air-conditioned	None
Residual heat removal heat exchanger shells	Pressure boundary	Carbon steel	Indoor – not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Residual heat removal pump seal water heat exchanger shells and covers	Pressure boundary	Carbon steel	Indoor – not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings Tubing/fittings Thermowells	Pressure boundary	Stainless steel	Indoor – not air-conditioned	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Containment air	None
Piping/fittings	Pressure boundary	Stainless steel	Embedded/Encased	None
Flow elements Orifices	Pressure boundary Throttling	Stainless steel	Indoor – not air-conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.3-6
EMERGENCY CONTAINMENT FILTRATION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment				
Emergency containment filter housings	Pressure boundary	Carbon steel	Air/Gas	Loss of material
Emergency containment filter floodjet spray nozzles	Pressure boundary Spray	Brass	Air/Gas	None
Piping/fittings	Pressure boundary	Copper	Air/Gas	None
Valves Piping/fittings	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None

TABLE 3.3-6 (continued)
EMERGENCY CONTAINMENT FILTRATION

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment				
Emergency containment filter housings	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Emergency containment filter floodjet spray nozzles	Pressure boundary Spray	Brass	Containment air	None
Piping/fittings	Pressure boundary	Copper	Containment air	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Containment air	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.3-7
CONTAINMENT POST ACCIDENT MONITORING AND CONTR

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment				
Hydrogen monitor pumps	Pressure boundary	Stainless steel	Air/Gas	None
Containment air radiation monitor sample pump casings	Pressure boundary	Aluminum	Air/Gas	None
Containment ventilation high-efficiency particulate air (HEPA) and charcoal filter housings	Pressure boundary	Stainless steel	Air/Gas	None
Post Accident Sampling System cooler shells, covers, and tube coils [1]	Pressure boundary	Stainless steel	Treated water	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Carbon steel	Air/Gas	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None
Valves Piping/fittings,	Pressure boundary	Brass	Air/Gas	None

NOTE: 1. Heat transfer is not a license renewal intended function for these components.

TABLE 3.3-7
CONTAINMENT POST ACCIDENT MONITORING AND CONTR

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
Internal Environment (continued)				
Tubing/fittings	Pressure boundary	Copper	Air/Gas	None
Orifices	Pressure boundary Throttling	Stainless steel	Air/Gas	None

TABLE 3.3-7 (continued)
CONTAINMENT POST ACCIDENT MONITORING AND CONTR

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment				
Hydrogen monitor pumps	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Containment air radiation monitor sample pump casings	Pressure boundary	Aluminum	Indoor - not air-conditioned	None
Containment ventilation HEPA and charcoal filter housings	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Post Accident Sampling System cooler shells, covers, and tube coils	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Carbon steel	Indoor - not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material

TABLE 3.3-7 (continued)
CONTAINMENT POST ACCIDENT MONITORING AND CONTR

Component / Commodity Group	Intended Function	Material	Environment	Aging Effect Requiring Management
External Environment (continued)				
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Valves Piping/fittings	Pressure boundary	Stainless steel	Containment air	None
Valves Piping/fittings Filters	Pressure boundary	Brass	Indoor - not air-conditioned	None
Orifices	Pressure boundary Throttling	Stainless steel	Indoor - not air-conditioned	None
Tubing/fittings	Pressure boundary	Copper	Indoor - not air-conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

3.4 AUXILIARY SYSTEMS

The following systems are included in this section:

- Intake Cooling Water
- Component Cooling Water
- Spent Fuel Pool Cooling
- Chemical and Volume Control
- Primary Water Makeup
- Sample Systems
- Waste Disposal
- Instrument Air
- Normal Containment and Control Rod Drive Mechanism Cooling
- Auxiliary Building and Electrical Equipment Room Ventilation
- Control Building Ventilation
- Emergency Diesel Generator Building Ventilation
- Turbine Building Ventilation
- Fire Protection
- Emergency Diesel Generators and Support Systems

Subsection 2.3.3 provides a description of these systems and identifies the components requiring an aging management review for license renewal. Appendix C contains the process that identified the aging effects requiring management for non-Class 1 components.

3.4.1 MATERIALS AND ENVIRONMENT

The Auxiliary Systems are exposed to internal environments of air/gas, raw water – cooling canals, raw water – city water, raw water – floor drainage, treated water – primary, treated water – borated, treated water – secondary, treated water, lubricating oil, and fuel oil; and external environments of outdoor, indoor – not air conditioned, indoor - air conditioned, containment air, embedded/encased, buried, and potential borated water leaks. (See Tables 3.0-1 and 3.0-2).

The only parts of systems or components considered to be inaccessible for inspection are those that are buried or embedded/encased in concrete. These environments are addressed as part of the aging management review process; see Table 3.0-2, “External Service Environments." Potential aging effects associated with these environments are reviewed and those aging effects requiring management are identified along with the credited aging management program(s). All other parts of systems and components can be accessed, if required. The only Auxiliary Systems containing inaccessible parts are Intake Cooling Water and Fire Protection, which contain buried cast iron and carbon steel piping and valves, and Waste Disposal and Spent Fuel Pool Cooling, which contain stainless steel piping embedded/encased in concrete.

The tanks, pumps, heat exchangers, piping, tubing, valves, and associated components and commodity groups for these systems are constructed of carbon steel, stainless steel, cast iron, monel, aluminum, aluminum-bronze, bronze, copper nickel, brass, aluminum brass, red brass, Worthite (nickel based alloy), copper, canvas, neoprene, and rubber. The components, their intended functions, materials, and environments for the Auxiliary Systems are summarized in Tables 3.4-1 through 3.4-15.

3.4.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Tables 3.4-1 through 3.4-15. The aging effects requiring management for each system are summarized in the following paragraphs.

Intake Cooling Water – The aging effects requiring management are loss of material for carbon steel, stainless steel, cast iron, monel, bronze, aluminum–bronze, and copper nickel components; and cracking for rubber expansion joints. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Component Cooling Water – The aging effects requiring management are loss of material for carbon steel, stainless steel, cast iron, brass, and copper nickel components; and loss of material and fouling for aluminum-brass heat exchanger tubing. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Spent Fuel Pool Cooling – The aging effects requiring management are loss of material for carbon steel, stainless steel, and worthite (nickel based alloy) components; and loss of material and fouling for stainless steel heat exchanger tubing. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Chemical and Volume Control – The aging effects requiring management are loss of material for carbon steel, stainless steel, and cast iron components; loss of material and fouling for copper components; and cracking for certain stainless steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Primary Water Makeup – The aging effect requiring management is loss of material for stainless steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Sample Systems – The aging effects requiring management are loss of material for carbon steel and stainless steel components, and cracking for certain stainless steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Waste Disposal – The aging effects requiring management are loss of material for carbon steel and stainless steel components and admiralty brass heat exchanger tubing, and fouling for stainless steel drain piping. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Instrument Air – The aging effects requiring management are loss of material for carbon steel, stainless steel, and copper alloy components; and fouling for aluminum heat exchanger fins. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Normal Containment and Control Rod Drive Mechanism Cooling – The aging effects requiring management are loss of material for carbon steel components, cracking for neoprene and coated canvas flexible connectors, and loss of material and fouling for admiralty brass, stainless steel, and aluminum heat exchanger tubing and fins. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Auxiliary Building and Electrical Equipment Room Ventilation – The aging effects requiring management are loss of material for carbon steel components and cracking for coated canvas flexible connectors. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Control Building Ventilation – The aging effects requiring management are loss of material for carbon steel, stainless steel, copper, and aluminum components; and cracking for coated canvas flexible connectors.

Emergency Diesel Generator Building Ventilation – No aging effects requiring management were identified.

Turbine Building Ventilation – The aging effects requiring management are loss of material for carbon steel, stainless steel, and aluminum components; loss of material and fouling for copper components; and cracking for coated canvas flexible connectors.

Fire Protection – The aging effects requiring management are loss of material for carbon steel, stainless steel, cast iron, and copper alloy components; loss of material and fouling for copper alloy heat exchanger tubing; and cracking for rubber

expansion joints and flexible connectors. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Emergency Diesel Generators and Support Systems – The aging effects requiring management are loss of material for carbon steel, cast iron, stainless steel, copper, and copper alloy components; cracking for stainless steel and rubber expansion joints and flexible couplings; and loss of material and fouling for red brass and copper alloy heat exchanger tubing.

3.4.3 OPERATING EXPERIENCE

3.4.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to Auxiliary Systems includes the following:

- NRC Bulletin 79-03, “Longitudinal Weld Defects in ASME SA-312 Type 304 Stainless Steel Pipe”
- NRC Bulletin 79-17, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”
- NRC Bulletin 80-24, “Prevention of Damage Due to Water Leakage Inside Containment”
- NRC Bulletin 81-03, “Flow Blockage of Cooling Water to Safety System Components by Corbicula sp. (Asiatic clam) and Mytilus sp. (mussel)”
- NRC Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants”
- NRC Bulletin 87-01, “Thinning of Pipe Walls in Nuclear Power Plants”
- NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems”
- NRC Bulletin 89-02, “Stress Corrosion Cracking Of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting In Anchor Darling Model S350W Swing Check Valves Or Valves Of Similar Design”
- NRC Circular 76-06, “Stress Corrosion Cracks in Stagnant, Low Pressure Stainless Piping Containing Boric Acid Solution at PWRs”
- NRC Circular 80-11, “Emergency Diesel Generator Lube Oil Cooler Failures”
- NRC Generic Letter 83-26, “Clarification of Surveillance Requirements for Diesel Fuel Oil Impurity Level Tests”
- NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants”

- NRC Generic Letter 88-14, “Instrument Air Supply System Problems Affecting Safety Related Equipment”
- NRC Generic Letter 89-08, “Erosion/Corrosion Induced Pipe Wall Thinning”
- NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety Related Equipment”
- NRC Generic Letter 90-05, “Guidance for Performing Temporary Non-Code Repair of ASME Class 1, 2 and 3 Piping”
- NRC Generic Letter 91-17, “Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants”
- NRC Information Notice 79-19, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants”
- NRC Information Notice 79-23, “Emergency Diesel Generator Lube Oil Coolers”
- NRC Information Notice 80-05, “Chloride Contamination of Safety Related Piping and Components”
- NRC Information Notice 80-15, “Axial (Longitudinal) Oriented Cracking in Piping”
- NRC Information Notice 80-40, “Excessive Nitrogen Supply Pressure Actuates Safety-Relief Valve Operation to Cause Reactor Depressurization”
- NRC Information Notice 81-38, “Potentially Significant Equipment Failures Resulting from Contamination of Air-Operated Systems”
- NRC Information Notice 83-46, “Common-Mode Valve Failures Degrade Surry’s Recirculation Spray Subsystem”
- NRC Information Notice 84-18, “Stress Corrosion Cracking in Pressurized Water Reactor Systems”
- NRC Information Notice 84-71, “Graphitic Corrosion of Cast Iron in Salt Water”
- NRC Information Notice 85-24, “Failures of Protective Coatings in Pipes and Heat Exchangers”
- NRC Information Notice 86-96, “Heat Exchanger Fouling Can Cause Inadequate Operability of Service Water Systems”

- NRC Information Notice 87-28, “Air Systems Problems at U.S. Light Water Reactors”
- NRC Information Notice 88-17, “Summary of Responses to NRC Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants”
- NRC Information Notice 88-37, “Flow Blockage of Cooling Water to Safety System Components”
- NRC Information Notice 89-01, “Valve Body Erosion”
- NRC Information Notice 89-07, “Failures of Small Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Render Emergency Diesel Generators Inoperable”
- NRC Information Notice 89-30, “High Temperature Environments at Nuclear Power Plants”
- NRC Information Notice 89-76, “Biofouling Agent: Zebra Mussel”
- NRC Information Notice 90-39, “Recent Problems with Service Water Systems”
- NRC Information Notice 90-65, “Recent Orifice Plate Problems”
- NRC Information Notice 91-05, “Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles”
- NRC Information Notice 91-46, “Degradation of Emergency Diesel Generator Fuel Oil Delivery Systems”
- NRC Information Notice 93-12, “Summary of Observations Compiled During Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs”
- NRC Information Notice 94-58, “Reactor Coolant Pump Lube Oil Fire”
- NRC Information Notice 94-59, “Accelerated Dealloying of Cast Aluminum-Bronze Valves Caused by Microbiologically Induced Corrosion”
- NRC Information Notice 94-79, “Microbiologically Influenced Corrosion of Emergency Diesel Generator Service Water Piping”
- NRC Information Notice 97-13, “Deficient Conditions Associated with Protective Coatings at Nuclear Power Plants”
- NRC Information Notice 98-43, “Leaks in Emergency Diesel Generator Lubricating Oil and Jacket Cooling Water Piping”

- NRC Information Notice 99-07, “Failed Fire Protection Deluge Valves and Potential Testing Deficiencies in Preaction Sprinkler Systems”

No aging effects requiring management were identified from the above documents beyond those identified in Subsection 3.4.2.

3.4.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. The review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of auxiliary systems component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.4.2.

3.4.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.4.2. Tables 3.4-1 through 3.4-15 contain the results of the aging management review for the Auxiliary Systems and summarize the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Fire Protection Program
- Galvanic Corrosion Susceptibility Inspection Program
- Intake Cooling Water System Inspection Program
- Periodic Surveillance and Preventive Maintenance Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the auxiliary systems components listed in Tables 3.4-1 through 3.4-15 are maintained consistent with the current licensing basis for the period of extended operation.

TABLE 3.4-1
INTAKE COOLING WATER[1]

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Intake cooling water pumps	Pressure boundary	Stainless steel Carbon steel Cast iron	Raw water – cooling canals	Loss of material
Intake cooling water pump expansion joints	Pressure boundary	Rubber	Raw water – cooling canals	Cracking
Basket strainers (shell)	Pressure boundary	Carbon steel	Raw water – cooling canals	Loss of material
Basket strainers (internal screens)	Filtration	Stainless steel	Raw water – cooling canals	Loss of material
Valves Piping/fittings (main lines upstream of basket strainers)	Pressure boundary	Cast iron	Raw water – cooling canals	Loss of material

NOTE: 1. The component cooling water heat exchangers are included with the Component Cooling Water System (see Ta

TABLE 3.4-1 (continued)
INTAKE COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Piping/fittings (main lines between basket strainers and component cooling water heat exchangers)	Pressure boundary	Cast iron	Raw water – cooling canals	Loss of material
Valves Piping/fittings (main lines downstream of component cooling water heat exchangers above ground)	Structural support [1]	Cast iron	Raw water – cooling canals	Loss of material
Piping/fittings (main lines downstream of component cooling water heat exchangers buried)	Structural integrity [2]	Cast iron	Raw water – cooling canals	None
Piping/fittings	Pressure boundary	Stainless steel	Raw water – cooling canals	Loss of material

NOTES: 1. Pressure boundary not required because heat transfer has been accomplished. Structural support required to
integrity function of the system.
2. Pressure boundary not required because heat transfer has been accomplished. Structural integrity required to

TABLE 3.4-1 (continued)
INTAKE COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Piping/fittings (vents on the outlet channels of the component cooling water heat exchangers)	Pressure boundary	Carbon steel	Raw water – cooling canals	Loss of material
Piping/fittings (instrumentation)	Pressure boundary	Bronze	Raw water – cooling canals	Loss of material
Piping/fittings (vents and drains welded to the component cooling water heat exchanger channels)	Pressure boundary	Copper nickel	Raw water – cooling canals	Loss of material
Valves Piping/fittings Chemical injection nozzles (Unit 3 only)	Pressure boundary	Monel	Raw water – cooling canals	Loss of material
Tubing/fittings	Pressure boundary	Stainless steel	Raw water – cooling canals	Loss of material
Valves (pump discharges)	Pressure boundary	Aluminum – bronze	Raw water – cooling canals	Loss of material

TABLE 3.4-1 (continued)
INTAKE COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves (instrumentation)	Pressure boundary	Stainless steel Bronze	Raw water – cooling canals	Loss of material
Valves (component cooling water heat exchanger vent and drain valves)	Pressure boundary	Bronze	Raw water – cooling canals	Loss of material
Orifices	Pressure boundary Throttling	Stainless steel	Raw water – cooling canals	Loss of material
Thermowells	Pressure boundary	Stainless steel	Raw water – cooling canals	Loss of material

TABLE 3.4-1 (continued)
INTAKE COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Intake cooling water pumps	Pressure boundary	Stainless steel Carbon steel Cast iron	Outdoor	Loss of material
Intake cooling water pump expansion joints	Pressure boundary	Rubber	Outdoor	Cracking
Basket strainers (shell)	Pressure boundary	Carbon steel	Outdoor	Loss of material
			Borated water leaks	Loss of material
Piping/fittings (upstream of the component cooling water heat exchangers buried)	Pressure boundary	Cast iron	Buried [1]	None
Valves Piping/fittings (intake structure and turbine building above ground)	Pressure boundary	Cast iron	Outdoor	Loss of material

NOTE: 1. Based on volumetric inspections and analysis, the corrosion rate due to loss of material aging mechanisms will n
intended functions.

TABLE 3.4-1 (continued)
INTAKE COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves Piping/fitting (component cooling water heat exchanger rooms upstream of the heat exchangers above ground)	Pressure boundary	Cast iron	Outdoor	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings (downstream of component cooling water heat exchangers above ground)	Structural support [1]	Cast iron	Outdoor	Loss of material
			Borated water leaks	Loss of material
Piping/fittings (downstream of component cooling water heat exchangers buried)	Structural integrity [2]	Cast iron	Buried	None

NOTES: 1. Pressure boundary not required because heat transfer has been accomplished. Structural support required to
integrity function of the system.
2. Pressure boundary not required because heat transfer has been accomplished. Structural integrity required to

TABLE 3.4-1 (continued)
INTAKE COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Piping/fittings	Pressure boundary	Stainless steel Carbon steel	Outdoor	Loss of material
		Carbon steel	Borated water leaks	Loss of material
		Bronze	Outdoor	None
Piping/fittings (welded to component cooling water heat exchanger channels)	Pressure boundary	Copper nickel	Outdoor	None
Piping/fittings Valves Chemical injection nozzles (Unit 3 only)	Pressure boundary	Monel	Outdoor	None
Tubing/fittings	Pressure boundary	Stainless steel	Outdoor	Loss of material

TABLE 3.4-1 (continued)
INTAKE COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves	Pressure boundary	Aluminum – bronze Bronze	Outdoor	None
		Stainless steel	Outdoor	Loss of material
		Carbon steel	Borated water leaks	Loss of material
			Outdoor	Loss of material
Orifices	Pressure boundary Throttling	Stainless steel	Outdoor	Loss of material
Thermowells	Pressure boundary	Stainless steel	Outdoor	Loss of material
Bolting (mechanical closures in component cooling water heat exchanger rooms)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-2
COMPONENT COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Component cooling water head tanks	Pressure boundary	Stainless steel	Air/Gas	None
			Treated water	Loss of material
Component cooling water surge tanks	Pressure boundary	Carbon steel	Treated water	Loss of material
Pressure vessels (air reservoirs)	Pressure boundary	Carbon steel	Air/Gas	None
Component cooling water pumps	Pressure boundary	Cast iron	Treated water	Loss of material
Component cooling water heat exchanger shells, baffle plates, connecting rods, spacers, and nuts	Pressure boundary	Carbon steel	Treated water	Loss of material
Component cooling water heat exchanger tube sheets	Pressure boundary	Copper nickel	Raw water – cooling canals (tube side)	Loss of material
			Treated water (shell side)	Loss of material

TABLE 3.4-2 (continued)
COMPONENT COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Component cooling water heat exchanger tubes	Pressure boundary Heat transfer	Aluminum-brass	Raw water – cooling canals (inside diameter)	Loss of material Fouling
			Treated water (outside diameter)	Loss of material
Component cooling water heat exchanger channels, and channel door overlay	Pressure boundary	Copper nickel	Raw water – cooling canals	Loss of material
Valves Piping/fittings	Pressure boundary	Carbon steel	Treated water	Loss of material
Valves Piping/fittings Tubing/fittings Thermowells	Pressure boundary	Stainless steel	Treated water	Loss of material
Valves Piping/fittings Tubing/fittings Filters	Pressure boundary	Stainless steel Carbon steel Brass	Air/Gas	None

TABLE 3.4-2 (continued)
COMPONENT COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves	Pressure boundary	Brass	Treated water	Loss of material
Rotometers	Pressure boundary Throttling	Stainless steel	Treated water	Loss of material
		Carbon steel	Treated water	Loss of material
Orifices	Pressure boundary Throttling	Stainless steel	Treated water	Loss of material
		Stainless steel	Air/Gas	None

TABLE 3.4-2 (continued)
COMPONENT COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Component cooling water head tanks	Pressure boundary	Stainless steel	Outdoor	None
Component cooling water surge tanks [1]	Pressure boundary	Carbon steel	Indoor - not air-conditioned	Loss of material
Pressure vessels (air reservoirs)	Pressure boundary	Carbon steel	Indoor - not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Component cooling water pumps	Pressure boundary	Cast iron	Outdoor	Loss of material
			Borated water leaks	Loss of material
Component cooling water heat exchanger shells, flanges, and doors	Pressure boundary	Carbon steel	Outdoor	Loss of material
			Borated water leaks	Loss of material
Component cooling water heat exchanger channels	Pressure boundary	Copper nickel	Outdoor	None

NOTE: 1. The component cooling water surge tanks are not located near borated water sources.

TABLE 3.4-2 (continued)
COMPONENT COOLING WATER

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air Indoor - not air-conditioned Outdoor	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings Tubing/fittings Filters Thermowells	Pressure boundary	Stainless steel	Containment air Indoor - not air-conditioned Outdoor	None
Orifices Rotometers	Pressure boundary Throttling	Stainless steel	Indoor – not air conditioned Containment air Outdoor	None
Rotometers	Pressure boundary Throttling	Carbon steel	Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Valves	Pressure boundary	Brass	Outdoor Indoor – not air conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-3
SPENT FUEL POOL COOLING

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Spent fuel pool cooling pumps Refueling water purification pumps	Pressure boundary	Stainless steel	Treated water – borated	Loss of material
Emergency spent fuel pool cooling pumps	Pressure boundary	Worthite (nickel based alloy)	Treated water – borated	Loss of material
Spent fuel pool cooling heat exchanger tubes	Pressure boundary Heat transfer	Stainless steel	Treated water – borated (inside diameter)	Loss of material Fouling
			Treated water (outside diameter)	Loss of material Fouling
Spent fuel pool cooling heat exchanger tube sheets	Pressure boundary	Stainless steel	Treated water – borated (tube side)	Loss of material
			Treated water (shell side)	Loss of material
Spent fuel pool cooling heat exchanger channels	Pressure boundary	Stainless steel	Treated water - borated	Loss of material

TABLE 3.4-3 (continued)
SPENT FUEL POOL COOLING

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Spent fuel pool cooling heat exchanger shells, covers and baffles	Pressure boundary	Carbon steel	Treated water	Loss of material
Valves Piping/fittings Tubing/fittings Filters Demineralizers	Pressure boundary	Stainless steel	Treated water – borated	Loss of material
Flow elements Orifices	Pressure boundary Throttling	Stainless steel	Treated water – borated	Loss of material

TABLE 3.4-3 (continued)
SPENT FUEL POOL COOLING

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Spent fuel pool cooling pumps	Pressure boundary	Stainless steel	Indoor – not air conditioned	None
Refueling water purification pumps	Pressure boundary	Stainless steel	Outdoor	None
Emergency spent fuel pool cooling pumps	Pressure boundary	Worthite (nickel based alloy)	Indoor – not air conditioned	None
Spent fuel pool cooling heat exchanger shells and covers	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings Tubing/fittings Filters Demineralizers	Pressure boundary	Stainless steel	Indoor – not air conditioned Outdoor	None
Piping/fittings	Pressure boundary	Stainless steel	Encased/Embedded (concrete)	None
			Treated water – borated (submerged)	Loss of material

TABLE 3.4-3 (continued)
SPENT FUEL POOL COOLING

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Flow elements Orifices	Pressure boundary Throttling	Stainless steel	Indoor – not air conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-4
CHEMICAL AND VOLUME CONTROL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Boric acid storage tanks Boric acid batching tank	Pressure boundary	Stainless steel	Treated water - borated	Loss of material Cracking [1]
			Air/Gas	None
Volume control tanks Holdup tanks	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
			Air/Gas	None
Boric acid storage tank pumps	Pressure boundary	Stainless steel	Treated water - borated	Loss of material Cracking [1]
Charging pumps	Pressure boundary	Stainless steel	Treated water - borated	Cracking
				Loss of material
Charging pump suction stabilizers and pulsation dampeners	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
Charging pump oil cooler bonnets	Pressure boundary	Cast iron	Treated water	Loss of material

NOTE: 1. Plant experience has identified the potential for cracking of previously heat traced components.

TABLE 3.4-4 (continued)
CHEMICAL AND VOLUME CONTROL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Charging pump oil cooler tubes	Pressure boundary Heat transfer	Copper	Treated water (inside diameter)	Loss of material Fouling
			Lubricating oil (outside diameter)	None
Non-regenerative heat exchanger shells and tube supports	Pressure boundary	Carbon steel	Treated water	Loss of material
Non-regenerative heat exchanger tubes	Pressure boundary Heat transfer	Stainless steel	Treated water – borated (inside diameter)	Loss of material Cracking [1]
			Treated water (outside diameter)	Loss of material
Non-regenerative heat exchanger tube sheets	Pressure boundary	Stainless steel	Treated water – borated (tube side)	Loss of material Cracking [1]
			Treated water (shell side)	Loss of material

NOTE: 1. Portions of the system >140ºF are potentially susceptible to stress corrosion cracking.

TABLE 3.4-4 (continued)
CHEMICAL AND VOLUME CONTROL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Non-regenerative heat exchanger channels	Pressure boundary	Stainless steel	Treated water – borated	Loss of material Cracking [1]
Valves Piping/fittings Thermowells (upstream of non-regenerative heat exchanger and downstream of regenerative heat exchanger)	Pressure boundary	Stainless steel	Treated water – borated	Loss of material Cracking [1]
Valves Piping/fittings Thermowells Tubing/fittings Demineralizers Filters (main charging and letdown loops)	Pressure boundary	Stainless steel	Treated water - borated	Loss of material

NOTE: 1. Portions of the system >140ºF are potentially susceptible to stress corrosion cracking.

TABLE 3.4-4 (continued)
CHEMICAL AND VOLUME CONTROL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Piping/fittings Thermowells Tubing/fittings Filters (boric acid supply to charging pump suction headers)	Pressure boundary	Stainless steel	Treated water - borated	Loss of material Cracking [1]
Valves Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None
Valves	Pressure boundary	Brass	Air/Gas	None
Orifices (main charging and letdown loops)	Pressure boundary Throttling	Stainless steel	Treated water – borated	Loss of material
Orifices Flow meters (boric acid supply to charging pump suction headers)	Pressure boundary Throttling	Stainless steel	Treated water – borated	Loss of material Cracking [1]

NOTE: 1. Plant experience has identified the potential for cracking of previously heat-traced components.

TABLE 3.4-4 (continued)
CHEMICAL AND VOLUME CONTROL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Boric acid storage tanks Boric acid batching tank	Pressure boundary	Stainless steel	Indoor - not air-conditioned	Cracking [1]
Volume control tanks Holdup tanks	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Boric acid storage tank pumps	Pressure boundary	Stainless steel	Indoor - not air-conditioned	Cracking [1]
Charging pumps	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Charging pump suction stabilizers and discharge dampeners	Pressure boundary	Stainless steel	Indoor - not air-conditioned	None
Charging pump oil cooler shells	Pressure boundary	Brass	Indoor - not air-conditioned	None
Charging pump oil cooler bonnets	Pressure boundary	Cast iron	Indoor - not air-conditioned	Loss of material
			Borated water leaks	Loss of material

NOTE: 1. Plant experience has identified the potential for cracking of previously heat-traced components.

TABLE 3.4-4 (continued)
CHEMICAL AND VOLUME CONTROL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Non-regenerative heat exchanger shells	Pressure boundary	Carbon steel	Indoor - not air-conditioned	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings Thermowells Tubing/fittings Demineralizers Filters (main charging and letdown loops)	Pressure boundary	Stainless steel	Containment air Indoor – not air conditioned	None
Valves Piping/fittings Thermowells Tubing/fittings Filters (boric acid supply to charging pump suction headers)	Pressure boundary	Stainless steel	Indoor – not air conditioned	Cracking[1]
Instrument solenoid valves	Pressure boundary	Brass	Containment air Indoor – not air conditioned	None

NOTE: 1. Plant experience has identified the potential for cracking of previously heat-traced components.

TABLE 3.4-4 (continued)
CHEMICAL AND VOLUME CONTROL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Orifices	Pressure boundary Throttling	Stainless steel	Containment air Indoor – not air conditioned	None
Orifices Flow meters	Pressure boundary Throttling	Stainless steel	Indoor – not air conditioned	Cracking[1]
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

NOTE: 1. Plant experience has identified the potential for cracking of previously heat-traced components.

TABLE 3.4-5
PRIMARY WATER MAKEUP

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Valves Piping/fittings	Pressure boundary	Stainless steel	Treated water	Loss of material
External Environment				
Valves Piping/fittings	Pressure boundary	Stainless steel	Indoor – not air conditioned Containment air	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure Integrity

TABLE 3.4-6
SAMPLE SYSTEMS

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
NSSS Sample System				
Internal Environment				
Sample heat exchanger shells and covers[1]	Pressure boundary	Carbon steel	Treated water	Loss of material
Valves Piping/fittings Tubing/fittings (pressurizer, reactor coolant system hot legs sample lines)	Pressure boundary	Stainless steel	Treated water - primary	Loss of material Cracking[2]
Valves Piping/fittings Tubing/fittings (safety injection system sample lines)	Pressure boundary	Stainless steel	Treated water - borated	Loss of material

NOTES: 1. Heat transfer is not a license renewal intended function for the sample heat exchangers. Additionally, the tub
exchangers is normally isolated so the tubes do not perform a pressure boundary function.
2. Stainless steel materials exposed to temperature >140ºF are susceptible to stress corrosion cracking (see Ap

TABLE 3.4-6 (continued)
SAMPLE SYSTEMS

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
NSSS Sample System (continued)				
Internal Environment (continued)				
Valves Piping/fittings Tubing/fittings (other sample lines)	Pressure boundary	Stainless steel	Treated water - borated	Loss of material Cracking
Valves Piping/fittings Tubing/fittings (volume control tank cover gas sample lines)	Pressure boundary	Stainless steel	Air/Gas	None

TABLE 3.4-6 (continued)
SAMPLE SYSTEMS

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
NSSS Sample System (continued)				
External Environment				
Sample cooler shells and covers	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Indoor – not air conditioned Containment air	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-6 (continued)
SAMPLE SYSTEMS

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Secondary Sample System				
Internal Environment				
Sample heat exchanger shells and covers	Structural support [1]	Stainless steel	Treated water	Loss of material
Sample heat exchanger tubes [2] (inside diameter)	Pressure boundary Throttling	Stainless steel	Treated water - secondary	Loss of material Cracking
Sample heat exchanger tubes (outside diameter)	Pressure boundary	Stainless steel	Treated water	Loss of material
Valves Piping/fittings	Pressure boundary	Carbon steel	Treated water - secondary	Loss of material
Valves Tubing/fittings	Pressure boundary Throttling	Stainless steel	Treated water - secondary	Loss of material Cracking

NOTES: 1. Pressure boundary not required. Provides structural support for the cooling coils.
2. Required for Main Steam System pressure boundary. Heat transfer is not a license renewal intended functio
exchangers.

TABLE 3.4-6 (continued)
SAMPLE SYSTEMS

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Secondary Sample System (continued)				
External Environment				
Sample heat exchanger shells and covers	Structural support	Stainless steel	Outdoor	None
Valves Piping/fittings	Pressure boundary	Carbon steel	Indoor – not air conditioned Containment air	None
			Borated water leaks	Loss of material
Valves Tubing/fittings	Pressure boundary Throttling	Stainless steel	Outdoor	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-7
WASTE DISPOSAL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Residual heat removal pit sump pumps	Pressure boundary	Stainless steel	Raw water – floor drainage	Loss of material
Waste gas compressor heat exchanger channels, covers, and tube sheets	Pressure boundary	Carbon steel	Treated water	Loss of material
Waste gas compressor heat exchanger tubes[1]	Pressure boundary	Admiralty brass	Treated water (inside diameter)	Loss of material
			Treated water (outside diameter)	Loss of material
Valves Piping/fittings	Pressure boundary	Stainless steel	Raw water – floor drainage	Loss of material
				Fouling (clogging of drain piping)

NOTE: 1. Required for Component Cooling Water System pressure boundary; heat transfer is not a license renewal intend
waste gas compressor heat exchangers.

TABLE 3.4-7 (continued)
WASTE DISPOSAL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None
Valves Piping/fittings	Pressure boundary	Stainless steel	Treated water - borated	Loss of material
		Carbon steel	Air/Gas	None

TABLE 3.4-7 (continued)
WASTE DISPOSAL

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Residual heat removal pit sump pumps	Pressure boundary	Stainless steel	Raw water – floor drainage	Loss of material
Waste gas compressor heat exchanger channels and covers	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Containment air Indoor – not air conditioned	None
Valves Piping/fittings	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Piping	Pressure boundary	Stainless steel	Embedded/Encased	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-8
INSTRUMENT AIR

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Instrument air receiver tanks	Pressure boundary	Carbon steel	Air/Gas (wetted)	Loss of material
Instrument air oil/water separators	Pressure boundary	Stainless steel	Air/Gas (wetted)	Loss of material
Instrument air aftercooler heat exchanger tube sheets, channels, header covers, shells	Pressure boundary	Carbon steel	Air/Gas (wetted)	Loss of material
Instrument air aftercooler heat exchanger tubes	Pressure boundary Heat transfer	Copper alloy	Air/Gas (wetted)	Loss of material
Instrument air filters	Pressure boundary	Carbon steel	Air/Gas	None
Instrument air dryers	Pressure boundary	Carbon steel	Air/Gas	None
Instrument air flasks	Pressure boundary	Carbon steel	Air/Gas	None

TABLE 3.4-8 (continued)
INSTRUMENT AIR

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Instrument air strainers	Pressure boundary Filtration	Carbon steel	Air/Gas (wetted)	Loss of material
Valves Piping/fittings (upstream of air dryers)	Pressure boundary	Carbon steel	Air/Gas (wetted)	Loss of material
Piping/fittings (upstream of air dryers)	Pressure boundary	Carbon steel – galvanized	Air/Gas (wetted)	Loss of material
Valves Tubing/fittings (upstream of air dryers)	Pressure boundary	Stainless steel	Air/Gas (wetted)	Loss of material

TABLE 3.4-8 (continued)
INSTRUMENT AIR

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Tubing/fittings (upstream of air dryers)	Pressure boundary	Copper alloy	Air/Gas (wetted)	Loss of material
Valves Piping/fittings (downstream of air dryers)	Pressure boundary	Carbon steel	Air/Gas	None
Piping/fittings (downstream of air dryers)	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Valves Piping/fittings Tubing/fittings (downstream of air dryers)	Pressure boundary	Stainless steel	Air/Gas	None
Valves Tubing/fittings (downstream of air dryers)	Pressure boundary	Copper alloy	Air/Gas	None
Orifices Flow elements	Pressure boundary Throttling	Stainless steel	Air/Gas	None

TABLE 3.4-8 (continued)
INSTRUMENT AIR

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Instrument air receiver tanks	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Instrument air oil/water separators	Pressure boundary	Stainless steel	Indoor – not air conditioned	None
Instrument air aftercooler heat exchanger covers and shells	Pressure boundary Heat transfer	Carbon steel	Outdoor	Loss of material
Instrument air heat exchanger fins	Heat transfer	Aluminum	Outdoor	Fouling
Instrument air strainers	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Instrument air filters	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Instrument air dryers	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Instrument air flasks	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material

TABLE 3.4-8 (continued)
INSTRUMENT AIR

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves Tubing/fittings	Pressure boundary	Copper alloy	Containment air Indoor – not air conditioned Outdoor	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Containment air Indoor – not air conditioned Outdoor	None
Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air Indoor – not air conditioned Outdoor	Loss of material
			Borated water leaks	Loss of material
Piping/fittings	Pressure boundary	Carbon steel – galvanized	Containment air Indoor – not air conditioned Outdoor	Loss of material
			Borated water leaks	Loss of material

TABLE 3.4-8 (continued)
INSTRUMENT AIR

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Containment air Indoor – not air conditioned	None
Orifices Flow elements	Pressure boundary Throttling	Stainless steel	Containment air Indoor – not air conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-9
NORMAL CONTAINMENT AND CONTROL ROD DRIVE MECHANISM

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Normal containment cooler housings	Pressure boundary Structural support [1]	Carbon steel	Containment air	Loss of material
Control rod drive mechanism cooler housings	Pressure boundary Structural support [1]	Carbon steel	Containment air	Loss of material
Normal containment cooler headers	Pressure boundary	Carbon steel	Treated water	Loss of material
Control rod drive mechanism cooler headers	Pressure boundary	Carbon steel	Treated water	Loss of material
Normal containment cooler tubes	Pressure boundary Heat transfer	Admiralty brass	Treated water (inside diameter)	Loss of material Fouling
Control rod drive mechanism cooler tubes (Unit 3)	Pressure boundary Heat transfer	Stainless steel	Treated water (inside diameter)	Loss of material Fouling

NOTE: 1. Housings provide structural support to ensure integrity of the safety-related Component Cooling Water System p

TABLE 3.4-9 (continued)
NORMAL CONTAINMENT AND CONTROL ROD DRIVE MECHANISM

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Control rod drive mechanism cooler tubes (Unit 4)	Pressure boundary Heat transfer	Admiralty brass	Treated water (inside diameter)	Loss of material Fouling
Ductwork	Pressure boundary	Carbon steel – galvanized	Containment air	None
Ductwork (wetted sections of ductwork near normal containment cooler housings)	Pressure boundary	Carbon steel – galvanized	Containment air	Loss of material
Control rod drive mechanism ductwork shrouds	Pressure boundary	Carbon steel – galvanized	Containment air	None
Normal containment cooling ductwork flexible connectors	Pressure boundary	Neoprene Coated canvas	Containment air	Cracking
Control rod drive mechanism cooling ductwork flexible connectors	Pressure boundary	Neoprene	Containment air	Cracking

TABLE 3.4-9 (continued)
NORMAL CONTAINMENT AND CONTROL ROD DRIVE MECHANISM

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Normal containment cooler housings	Pressure boundary Structural support [1]	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Control rod drive mechanism cooler housings	Pressure boundary Structural support [1]	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Normal containment cooler headers	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Control rod drive mechanism cooler header	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material

NOTE: 1. Housings provide structural support to ensure integrity of the safety-related Component Cooling Water System p

TABLE 3.4-9 (continued)
NORMAL CONTAINMENT AND CONTROL ROD DRIVE MECHANISM

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Normal containment cooler tubes	Pressure boundary Heat transfer	Admiralty brass	Containment air	Loss of material
Normal containment cooler tube fins	Heat transfer	Aluminum	Containment air	Loss of material Fouling
Control rod drive mechanism cooler tubes (Unit 3)	Pressure boundary Heat transfer	Stainless steel	Containment air	Loss of material
Control rod drive mechanism cooler tubes (Unit 4)	Pressure boundary Heat transfer	Admiralty brass	Containment air	Loss of material
Control rod drive mechanism cooler fins	Heat transfer	Aluminum	Containment air	Loss of material
				Fouling
Ductwork	Pressure boundary	Carbon steel – galvanized	Containment air	None
			Borated water leaks	Loss of material

TABLE 3.4-9 (continued)
NORMAL CONTAINMENT AND CONTROL ROD DRIVE MECHANISM

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Ductwork (wetted) (section of ductwork near normal containment cooler housings)	Pressure boundary	Carbon steel – galvanized	Containment air	Loss of material
			Borated water leaks	Loss of material
Control rod drive mechanism ductwork and shrouds	Pressure boundary	Carbon steel – galvanized	Containment air	None
			Borated water leaks	Loss of material
Normal containment cooling ductwork flexible connectors	Pressure boundary	Neoprene Coated canvas	Containment air	Cracking
Control rod drive mechanism ductwork flexible connectors	Pressure boundary	Neoprene	Containment air	Cracking
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-10
AUXILIARY BUILDING AND ELECTRICAL EQUIPMENT ROOM VENT

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Auxiliary building ventilation air handler housings	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Auxiliary building ventilation pre and roughing filter housings	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Ductwork	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
		Carbon steel	Air/Gas	Loss of material
Pressure test point plugs	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None
Flexible connectors	Pressure boundary	Coated canvas	Air/Gas	Cracking

TABLE 3.4-10 (continued)
AUXILIARY BUILDING AND ELECTRICAL EQUIPMENT ROOM VENT

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Auxiliary building ventilation air handler housings	Pressure boundary	Carbon steel – galvanized	Indoor –not air conditioned	None
Auxiliary building ventilation pre and roughing filter housings	Pressure boundary	Carbon steel – galvanized	Indoor – not air conditioned	None
			Borated water leaks	Loss of material
Ductwork	Pressure boundary	Carbon steel – galvanized	Indoor – not air conditioned	None
			Borated water leaks	Loss of material
		Carbon steel	Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Pressure test point plugs	Pressure boundary	Carbon steel – galvanized	Indoor – not air conditioned	None
			Borated water leaks	Loss of material
Tubing/fittings	Pressure boundary	Stainless steel	Indoor – not air conditioned	None

TABLE 3.4-10 (continued)
AUXILIARY BUILDING AND ELECTRICAL EQUIPMENT ROOM VENT

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Flexible connectors	Pressure boundary	Coated canvas	Indoor – not air conditioned	Cracking
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-11
CONTROL BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Cable spreading room and computer room chilled water surge tanks	Pressure boundary	Carbon steel	Treated water	Loss of material
			Air/Gas	Loss of material
Cable spreading room and computer room chilled water pumps	Pressure boundary	Carbon steel	Treated water	Loss of material
Cable spreading room and computer room chiller water boxes	Pressure boundary Heat transfer	Carbon steel	Treated water	Loss of material
Wye strainers Thermowells	Pressure boundary	Carbon steel	Treated water	Loss of material
Valves Piping/fittings Level gauges	Pressure boundary	Carbon steel	Treated water	Loss of material
			Air/Gas	Loss of material
Flow elements	Pressure boundary Throttling	Carbon steel	Treated water	Loss of material

TABLE 3.4-11 (continued)
CONTROL BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Flow elements	Pressure boundary Throttling	Stainless steel	Treated water	Loss of material
Air separators Valves Tubing/fittings	Pressure boundary	Stainless steel	Treated water	Loss of material
			Air/Gas	None
Control room ventilation air handling unit housings	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Control room ventilation recirculation filter housing	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Inverter room and battery room air handling unit housing	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Cable spreading room and computer room air handling unit housings	Pressure boundary	Stainless steel	Air/Gas	None
Cable spreading room and computer room air handling unit headers	Pressure boundary	Stainless steel	Treated water	Loss of material

TABLE 3.4-11 (continued)
CONTROL BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Cable spreading room and computer room air handling unit tubes	Pressure boundary Heat transfer	Copper	Treated water	Loss of material
Cable spreading room and computer room air handling unit air boxes in air handlers	Pressure boundary	Carbon steel	Air/Gas (wetted with condensation)	Loss of material
Ductwork	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Ductwork flexible connectors	Pressure boundary	Coated canvas	Air/Gas	Cracking

TABLE 3.4-11 (continued)
CONTROL BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Cable spreading room and computer room chilled water surge tanks	Pressure boundary	Carbon steel	Outdoor	Loss of material
Cable spreading room and computer room chilled water pumps	Pressure boundary	Carbon steel	Outdoor	Loss of material
Cable spreading room and computer room chiller water boxes	Pressure boundary Heat transfer	Carbon steel	Outdoor	Loss of material
Wye strainer Valves Piping/fittings Level gauges Thermowells	Pressure boundary	Carbon steel	Outdoor	Loss of material
Valves Piping/fittings Thermowells	Pressure boundary	Carbon steel	Indoor – air conditioned	None

TABLE 3.4-11 (continued)
CONTROL BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Flow elements	Pressure boundary Throttling	Carbon steel	Outdoor	Loss of material
			Indoor – air conditioned	None
Valves Tubing/fittings	Pressure boundary	Stainless steel	Indoor – air conditioned	None
Air separators Valves Tubing/fittings	Pressure boundary	Stainless steel	Outdoor	None
Flow elements	Pressure boundary Throttling	Stainless steel	Outdoor Indoor – air conditioned	None
Control room ventilation air handling unit housings	Pressure boundary	Carbon steel – galvanized	Indoor – air conditioned	None
Control room ventilation recirculation filter housing	Pressure boundary	Carbon steel – galvanized	Indoor – air conditioned	None
Inverter room and battery room air handling unit housing	Pressure boundary	Carbon steel – galvanized	Indoor – air conditioned	None

TABLE 3.4-11 (continued)
CONTROL BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Cable spreading room and computer room air handling unit housings	Pressure boundary	Stainless steel	Indoor – air conditioned	None
Cable spreading room and computer room air handling unit headers	Pressure boundary	Stainless steel	Indoor – air conditioned (wetted with condensation)	Loss of material
Cable spreading room and computer room air handling unit tubes	Pressure boundary Heat transfer	Copper	Indoor – air conditioned (wetted with condensation)	Loss of material
Cable spreading room and computer room air handling unit air boxes in air handlers	Pressure boundary	Carbon steel	Indoor – air conditioned (wetted with condensation)	Loss of material
Cable spreading room and computer room air handling unit tube fins	Heat transfer	Aluminum	Indoor – air conditioned (wetted with condensation)	Loss of material
Ductwork	Pressure boundary	Carbon steel – galvanized	Indoor – air conditioned	None
Ductwork flexible connectors	Pressure boundary	Coated canvas	Indoor – air conditioned	Cracking

TABLE 3.4-11 (continued)
CONTROL BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Outdoor Indoor – air conditioned	None

TABLE 3.4-12
EMERGENCY DIESEL GENERATOR BUILDING VENTILATIO

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Ductwork	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
Filter housings	Pressure boundary	Carbon steel – galvanized	Air/Gas	None
External Environment				
Ductwork	Pressure boundary	Carbon steel – galvanized	Indoor – not air conditioned Indoor – air conditioned	None
Filter housings	Pressure boundary	Carbon steel – galvanized	Indoor – not air conditioned Indoor – air conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Indoor – not air conditioned Indoor – air conditioned	None

TABLE 3.4-13
TURBINE BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Chilled water surge tanks	Pressure boundary	Carbon steel	Treated water	Loss of material
			Air/Gas	Loss of material
Chilled water air separators	Pressure boundary	Carbon steel	Treated water	Loss of material
			Air/Gas	Loss of material
Chilled water pumps	Pressure boundary	Carbon steel	Treated water	Loss of material
Chiller water boxes	Pressure boundary Heat transfer	Carbon steel	Treated water	Loss of material
Valves Piping/fittings	Pressure boundary	Carbon steel	Air/Gas	Loss of material
			Treated water	Loss of material

TABLE 3.4-13 (continued)
TURBINE BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Tubing/fittings Level gauges	Pressure boundary	Stainless steel	Treated water	Loss of material
			Air/Gas	None
Flexible hoses	Pressure boundary	Stainless steel	Treated water	Loss of material
Wye strainers Thermowells Test wells	Pressure boundary	Carbon steel	Treated water	Loss of material
Flow elements	Pressure boundary Throttling	Stainless steel Carbon steel	Treated water	Loss of material
Air handling unit housings	Pressure boundary	Carbon steel	Air/Gas	Loss of material
Air handling unit headers	Pressure boundary	Carbon steel	Treated water	Loss of material
Air handling unit heat exchanger tubes	Pressure boundary Heat transfer	Copper	Treated water	Loss of material Fouling
Air handling unit air boxes in air handlers	Pressure boundary	Carbon steel	Air/Gas (wetted with condensation)	Loss of material

TABLE 3.4-13 (continued)
TURBINE BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Chilled water surge tanks	Pressure boundary	Carbon steel	Outdoor	Loss of material
Chilled water air separator	Pressure boundary	Carbon steel	Outdoor	Loss of material
Chilled water pumps	Pressure boundary	Carbon steel	Outdoor	Loss of material
Chiller water boxes	Pressure boundary Heat transfer	Carbon steel	Outdoor	Loss of material
Valves Piping/fittings Wye strainers Thermowells Test wells	Pressure boundary	Carbon steel	Outdoor	Loss of material
Valves Piping/fittings Test wells	Pressure boundary	Carbon steel	Indoor – air conditioned	None

TABLE 3.4-13 (continued)
TURBINE BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves Tubing/fittings Flexible hoses Level gauges	Pressure boundary	Stainless steel	Outdoor	None
Valves Tubing/fittings Flexible hoses	Pressure boundary	Stainless steel	Indoor – air conditioned	None
Flow elements	Pressure boundary Throttling	Carbon steel	Outdoor	Loss of material
			Indoor – air conditioned	None
		Stainless steel	Outdoor	None
Air handling unit housings	Pressure boundary	Carbon steel – galvanized Stainless steel	Indoor – air conditioned	None
Air handling unit headers	Pressure boundary	Carbon steel	Indoor – air conditioned (wetted with condensation)	Loss of material
Air handling unit heat exchanger tubes	Pressure boundary Heat transfer	Copper	Indoor – air conditioned (wetted with condensation)	Loss of material

TABLE 3.4-13 (continued)
TURBINE BUILDING VENTILATION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Air handling unit air boxes	Pressure boundary	Carbon steel	Indoor – air conditioned (wetted with condensation)	Loss of material
Air handling unit heat exchanger fins	Heat transfer	Aluminum	Indoor – air conditioned (wetted with condensation)	Loss of material
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Outdoor Indoor – air conditioned	None

TABLE 3.4-14
FIRE PROTECTION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Raw water tanks	Pressure boundary	Carbon steel	Air/Gas Raw water – city water	Loss of material
Diesel driven fire pump fuel oil tank	Pressure boundary	Carbon steel	Fuel oil	None
			Air/Gas	None
Reactor coolant pump oil collection tanks	Pressure boundary	Carbon steel	Lubricating oil Air/Gas	None
Reactor coolant pump oil collection enclosures and drip pans	Pressure boundary	Stainless steel	Lubricating oil Air/Gas	None
Electric fire pump Diesel fire pump	Pressure boundary	Cast iron	Raw water – city water	Loss of material
Diesel fire pump heat exchanger (radiator) shell	Pressure boundary	Cast iron	Treated water	Loss of material
Diesel fire pump heat exchanger (radiator) tubes and cover	Pressure boundary Heat transfer	Copper alloy	Raw water – city water	Loss of material Fouling

TABLE 3.4-14 (continued)
FIRE PROTECTION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Basket strainers (body)	Pressure boundary	Cast iron	Raw water – city water	Loss of material
Basket strainers (elements)	Filtration	Stainless steel	Raw water – city water	Loss of material
Valves Piping/fittings	Pressure boundary	Carbon steel	Raw water – city water	Loss of material
			Air/Gas (associated with oil tanks) [1]	None
			Air/Gas	Loss of material

NOTE: 1. Internal surfaces of oil tanks are exposed to a combination of oil and air/gas. The components have an oily film

TABLE 3.4-14 (continued)
FIRE PROTECTION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Piping/fittings	Pressure boundary	Carbon steel – galvanized	Raw water – city water	Loss of material
			Air/Gas	None
Valves Piping/fittings	Pressure boundary	Cast iron	Raw water – city water	Loss of material
Valves	Pressure boundary	Cast iron	Air/Gas	Loss of material
Valves Tubing/fittings	Pressure boundary	Copper alloy	Raw water – city water	Loss of material
			Air/Gas	None
Sprinkler heads	Pressure boundary Spray	Copper alloy	Air/Gas	None
			Raw water – city water	Loss of material
Valves Piping/fittings Tubing/fittings Flexible hoses	Pressure boundary	Stainless steel	Raw water – city water	Loss of material

TABLE 3.4-14 (continued)
FIRE PROTECTION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Piping/fittings Tubing/fittings Flexible hoses	Pressure boundary	Stainless steel	Lubricating oil	None
Flow restriction orifices	Pressure boundary Throttling	Stainless steel	Raw water – city water	Loss of material
Flame arrestors	Fire spread prevention	Carbon steel	Air/Gas	None

TABLE 3.4-14 (continued)
FIRE PROTECTION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Raw water tanks	Pressure boundary	Carbon steel	Outdoor	Loss of material
Diesel driven fire pump fuel oil tank	Pressure boundary	Carbon steel	Outdoor	Loss of material
Reactor coolant pump oil collection tank Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Reactor coolant pump oil collection enclosures and drip pans	Pressure boundary	Stainless steel	Containment air	None
Electric fire pump	Pressure boundary	Carbon steel Cast iron	Outdoor	Loss of material
Diesel fire pump	Pressure boundary	Carbon steel Cast iron	Indoor – not air conditioned	Loss of material
Diesel fire pump heat exchanger (radiator) shell	Pressure boundary	Cast iron	Indoor – not air conditioned	Loss of material
Diesel fire pump heat exchanger (radiator) cover	Pressure boundary	Copper alloy	Indoor – not air conditioned	None

TABLE 3.4-14 (continued)
FIRE PROTECTION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Basket strainers (body)	Pressure boundary	Cast iron	Outdoor	Loss of material
Valves Piping/fittings	Pressure boundary	Carbon steel	Outdoor	Loss of material
Valves Piping/fittings	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Piping/fittings	Pressure boundary	Carbon steel – galvanized	Outdoor Indoor – not air conditioned	None
Valves Piping/fittings	Pressure boundary	Cast iron	Outdoor	Loss of material
Valves Piping/fittings	Pressure boundary	Cast iron Carbon steel	Buried	Loss of material
Valves Piping/fittings Tubing/fittings Flexible hoses	Pressure boundary	Stainless steel	Containment air Indoor – not air conditioned Outdoor	None
Valves Tubing/fittings	Pressure boundary	Copper alloy	Outdoor	None
Sprinkler heads	Pressure boundary Spray	Copper alloy	Outdoor	None

TABLE 3.4-14 (continued)
FIRE PROTECTION

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Flow restriction orifices	Pressure boundary Throttling	Stainless steel	Outdoor	None
Flame arrestors	Prevent spread of fire to fuel tank	Carbon steel	Outdoor	Loss of material
Expansion joints	Pressure boundary	Rubber	Indoor – not air conditioned	Cracking
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.4-15
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Air Intake and Exhaust System				
Internal Environment				
Exhaust piping/ fittings Silencers	Pressure boundary	Carbon steel	Air/Gas	Loss of material
Air filter assemblies	Pressure boundary Filtration	Carbon steel	Air/Gas	Loss of material
Expansion joints	Pressure boundary	Stainless steel	Air/Gas	Cracking
Flexible couplings	Pressure boundary	Rubber	Air/Gas	Cracking
Tubing/fittings	Pressure boundary	Stainless steel	Air/Gas	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Air Intake and Exhaust System (continued)				
External Environment				
Exhaust piping/ fittings Silencers	Pressure boundary	Carbon steel	Outdoor (Unit 3) Indoor – not air conditioned (Units 3 and 4)	Loss of material
Air filter assemblies	Pressure boundary Filtration	Carbon steel	Indoor – not air conditioned	Loss of material
Expansion joints	Pressure boundary	Stainless steel	Indoor – not air conditioned	Cracking
Flexible couplings	Pressure boundary	Rubber	Indoor – not air conditioned	Cracking
Tubing/fittings	Pressure boundary	Stainless steel	Indoor – not air conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Outdoor Indoor – not air conditioned	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Air Start System				
Internal Environment				
Air start accumulators	Pressure boundary	Carbon steel Stainless steel	Air/Gas	None
Air start motors	Pressure boundary	Aluminum alloy	Air/Gas	None
Air start system lubricators	Pressure boundary	Carbon steel	Air/Gas	None
Valves Piping/fittings Tubing/fittings Governor bypasses	Pressure boundary	Carbon steel – galvanized Carbon steel Stainless steel Copper alloy Cast iron	Air/Gas	None
Filters	Pressure boundary Filtration	Stainless steel	Air/Gas	None
Flexible hose	Pressure boundary	Rubber braided	Air/Gas	Cracking

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Air Start System (continued)				
External Environment				
Air start accumulators	Pressure boundary	Carbon steel (Unit 3)	Indoor – not air conditioned	Loss of material
		Stainless steel (Unit 4)	Indoor – not air conditioned	None
Air start motors	Pressure boundary	Aluminum alloy	Indoor – not air conditioned	None
Air start system lubricators	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Valves Piping/fittings Tubing/fittings Governor bypasses	Pressure boundary	Carbon steel Cast iron	Indoor – not air conditioned	Loss of material
		Carbon steel (galvanized) Stainless steel Copper alloy	Indoor – not air conditioned	None
Filters	Pressure boundary Filtration	Stainless steel	Indoor – not air conditioned	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Air Start System (continued)				
External Environment (continued)				
Flexible hoses	Pressure boundary	Rubber braided	Indoor – not air conditioned	Cracking
		Stainless steel	Indoor – not air conditioned	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Indoor – not air conditioned	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Fuel Oil System				
Internal Environment				
Unit 3 Diesel Oil Storage Tank	Pressure boundary	Carbon steel	Fuel oil	Loss of material
			Air/Gas	Loss of material
Unit 4 Diesel Oil Storage Tank (liner)	Pressure boundary	Carbon steel	Fuel oil	Loss of material
			Air/Gas	Loss of material
Emergency diesel generator fuel oil pumps	Pressure boundary	Carbon steel Cast iron	Fuel oil	Loss of material

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Fuel Oil System (continued)				
Internal Environment (continued)				
Diesel oil day tanks	Pressure boundary	Carbon steel	Fuel oil	Loss of material
			Air/Gas	None
Diesel oil skid tanks	Pressure boundary	Carbon steel	Fuel oil	Loss of material
			Air/Gas	None
Valves Piping/fittings Sight glasses	Pressure boundary	Carbon steel	Fuel oil	Loss of material
			Air/Gas	None
Valves Piping/fittings Tubing/fittings Flex hoses Filters Sight glasses	Pressure boundary	Stainless steel	Fuel oil	Loss of material
			Air/Gas	None
Tubing/fittings	Pressure boundary	Copper	Fuel oil	Loss of material
Filters	Pressure boundary Filtration	Carbon steel Cast iron Stainless steel	Fuel oil	Loss of material

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Fuel Oil System (continued)				
Internal Environment (continued)				
Flame arrestors	Pressure boundary Fire spread prevention	Carbon steel	Air/Gas	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Fuel Oil System (continued)				
External Environment				
Unit 3 Diesel Oil Storage Tank	Pressure boundary	Carbon steel	Outdoor	Loss of material
Unit 4 Diesel Oil Storage Tank (liner)	Pressure boundary	Carbon steel	Embedded/Encased	None
Diesel oil day tanks	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Diesel oil skid tanks	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Emergency diesel generator fuel oil pumps	Pressure boundary	Carbon steel Cast iron	Outdoor (Unit 3) Indoor – not air conditioned (Unit 4)	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Carbon steel Stainless steel	Outdoor	Loss of material
Valves Piping/fittings Sight glasses	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Fuel Oil System (continued)				
External Environment (continued)				
Valves Piping/fittings Tubing/fittings Flex hoses Filters Sight glasses	Pressure boundary	Stainless steel	Indoor – not air conditioned	Loss of material
Tubing/fittings	Pressure boundary	Copper	Indoor – not air conditioned	None
Filters	Pressure boundary Filtration	Carbon steel Cast iron Stainless steel	Indoor – not air conditioned	Loss of material
Flame arrestors	Pressure boundary Fire spread prevention	Carbon steel	Outdoor	Loss of material
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Outdoor Indoor – not air conditioned	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTE

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Cooling Water System				
Internal Environment				
Diesel generator cooling water expansion tanks	Pressure boundary	Carbon steel	Treated water	Loss of material
Diesel generator cooling water pumps	Pressure boundary	Cast iron	Treated water	Loss of material
Diesel generator cooling water immersion heaters	Pressure boundary	Carbon steel	Treated water	Loss of material
Radiator water boxes (Units 3 and 4)	Pressure boundary	Carbon steel	Treated water	Loss of material
Radiator tubes (Units 3 and 4)	Pressure boundary Heat transfer	Copper alloy	Treated water	Loss of material Fouling
Valves Piping/fittings Tubing/fittings	Pressure boundary	Carbon steel	Treated water	Loss of material

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Cooling Water System (continued)				
Internal Environment (continued)				
Tubing/fittings Flexible hoses	Pressure boundary	Stainless steel	Treated water	Loss of material
Orifices	Pressure boundary Throttling	Stainless steel	Treated water	Loss of material
Valves Sight glasses	Pressure boundary	Copper alloy	Treated water	Loss of material

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Cooling Water System (continued)				
External Environment				
Diesel generator cooling water expansion tanks	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Diesel generator cooling water pumps	Pressure boundary	Cast iron	Indoor – not air conditioned	Loss of material
Diesel generator cooling water immersion heaters	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Radiator water boxes (Units 3 and 4)	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Radiator tubes (Unit 3)	Pressure boundary Heat transfer	Copper alloy	Indoor – not air conditioned	Loss of material
Radiator tubes (Unit 4)	Pressure boundary Heat transfer	Copper alloy	Indoor – not air conditioned	None [1]
Valves Piping/fittings Tubing/fittings	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material

NOTE: 1. Precluded by design, Unit 4 radiators are substantially different from Unit 3 radiators.

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Cooling Water System (continued)				
External Environment (continued)				
Tubing/fittings Flexible hoses	Pressure boundary	Stainless steel	Indoor – not air conditioned	None
Orifices	Pressure boundary Throttling	Stainless steel	Indoor – not air conditioned	None
Valves Sight glasses	Pressure boundary	Copper alloy	Indoor – not air conditioned	None
Flexible hoses	Pressure boundary	Rubber	Indoor – not air conditioned	Cracking
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Indoor – not air conditioned	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Lube Oil System				
Internal Environment				
Diesel generator lube oil pumps	Pressure boundary	Carbon steel Cast iron	Lubricating oil	None
Heat exchanger shells	Pressure boundary	Carbon steel	Lubricating oil	None
Heat exchanger tubing	Pressure boundary Heat transfer	Red brass	Treated water (inside diameter)	Loss of material Fouling
			Lubricating oil (outside diameter)	None
Heat exchanger channel heads	Pressure boundary	Cast iron	Treated water	Loss of material
Valves Piping/fittings Flexible hoses Sight glasses	Pressure boundary	Carbon steel Cast iron	Lubricating oil	None
Filters	Pressure boundary Filtration	Carbon steel Cast iron	Lubricating oil	None
Tubing/fittings	Pressure boundary	Stainless steel	Lubricating oil	None
Orifices	Pressure boundary Throttling	Carbon steel Stainless steel	Lubricating oil	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTE

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Lube Oil System (continued)				
External Environment				
Diesel generator lube oil pumps	Pressure boundary	Carbon steel Cast iron	Indoor – not air conditioned	Loss of material
Heat exchanger shells	Pressure boundary	Carbon steel	Indoor – not air conditioned	Loss of material
Heat exchanger channel heads	Pressure boundary	Cast iron	Indoor – not air conditioned	Loss of material
Valves Piping/fittings Tubing/fittings Flexible hoses Sight glasses	Pressure boundary	Carbon steel Cast iron	Indoor – not air conditioned	Loss of material
Filters	Pressure boundary Filtration	Carbon steel Cast iron	Indoor – not air conditioned	Loss of material
Tubing/fittings	Pressure boundary	Stainless steel	Indoor – not air conditioned	None
Orifices	Pressure boundary Throttling	Carbon steel	Indoor – not air conditioned	Loss of material
		Stainless steel	Indoor – not air conditioned	None

TABLE 3.4-15 (continued)
EMERGENCY DIESEL GENERATORS AND SUPPORT SYSTEI

Component/ Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Lube Oil System (continued)				
External Environment (continued)				
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Indoor – not air conditioned	None

3.5 STEAM AND POWER CONVERSION SYSTEMS

The following systems are included in this section:

- Main Steam and Turbine Generators
- Feedwater and Blowdown
- Auxiliary Feedwater and Condensate Storage

Subsection 2.3.4 provides a description of these systems and identifies the components requiring an aging management review for license renewal. Appendix C contains the process that identified the aging effects requiring management for non-Class 1 components.

3.5.1 MATERIALS AND ENVIRONMENT

The Steam and Power Conversion Systems are exposed to internal environments of treated water - secondary, treated water, lubricating oil, and air/gas; and external environments of outdoor, containment air, buried, and potential borated water leaks (see Tables 3.0-1 and 3.0-2).

The only parts of systems or components considered to be inaccessible for inspection are those that are buried or embedded/encased in concrete. These environments are addressed as part of the aging management review process; see Table 3.0-2, “External Service Environments." Potential aging effects associated with these environments are reviewed and those aging effects requiring management are identified along with the credited aging management program(s). All other parts of systems and components can be accessed, if required. The only Steam and Power Conversion System containing inaccessible piping parts is the Standby Steam Generator Feedwater System, which contains sections of buried stainless steel piping.

The tanks, pumps, heat exchangers, piping, tubing, valves, and associated components and commodity groups for these systems are constructed of carbon steel, stainless steel, low alloy steel, cast iron, and brass. The components and commodity groups, their intended functions, the materials, and environments for the Steam and Power Conversion Systems are summarized in Tables 3.5-1 through 3.5-3.

3.5.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects requiring management and the programs and activities that manage the aging effects for each applicable environment and material combination are provided in Tables 3.5-1 through 3.5-3. The aging effects requiring management for each system are summarized in the following paragraphs.

Main Steam and Turbine Generators - The aging effects requiring management are loss of material for carbon steel and stainless steel components, and cracking for certain stainless steel components and heat exchanger tubing. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Feedwater and Blowdown - The aging effects requiring management are loss of material for carbon steel and stainless steel components, and cracking for certain stainless steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

Auxiliary Feedwater and Condensate Storage - The aging effect requiring management is loss of material for cast iron, admiralty brass, carbon steel, low alloy steel, and stainless steel components.

3.5.3 OPERATING EXPERIENCE

3.5.3.1 INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to Steam and Power Conversion Systems includes the following:

- NRC Bulletin 79-03, "Longitudinal Weld Defects in ASME SA-312 Type 304 Stainless Steel Pipe"
- NRC Bulletin 79-13, “Cracking in Feedwater System Piping”
- NRC Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants"
- NRC Bulletin 87-01, “Thinning of Pipe Walls in Nuclear Power Plants”
- NRC Bulletin 89-02, "Stress Corrosion Cracking of High Hardness Type 410 Stainless Steel Internal Preloaded Bolting In Anchor Darling Model S530W Swing Check Valves or Valves of Similar Design"
- NRC Generic Letter 79-20, “Information Requested on PWR Feedwater Lines”
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety Related Equipment"
- NRC Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning”
- NRC Generic Letter 91-17, "Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants"
- NRC Information Notice 80-05, "Chloride Contamination of Safety Related Piping and Components"
- NRC Information Notice 80-29, “Broken Studs on Terry Turbine Steam Inlet Flanges”
- NRC Information Notice 81-04, “Cracking in Main Steam Lines”

- NRC Information Notice 84-32, "Auxiliary Feedwater Sparger and Pipe Hanger Damage"
- NRC Information Notice 84-87, “Piping Thermal Deflection Induced by Stratified Flow”
- NRC Information Notice 85-56, "Inadequate Environment Control for Components and Systems in Extended Storage"
- NRC Information Notice 86-106, “Feedwater Line Break”
- NRC Information Notice 87-28, "Air Systems Problems at U.S. Light Water Reactors"
- NRC Information Notice 87-36, “Significant Unexpected Erosion of Feedwater Lines"
- NRC Information Notice 88-17, “Summary of Responses to NRC Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants”
- NRC Information Notice 88-37, Flow Blockage of Cooling Water to Safety System Components"
- NRC Information Notice 89-01, "Valve Body Erosion"
- NRC Information Notice 89-53, "Rupture of Extraction Steam Line on High Pressure Turbine"
- NRC Information Notice 89-76, "Biofouling Agent: Zebra Mussel"
- NRC Information Notice 89-80, "Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping"
- NRC Information Notice 90-65, "Recent Orifice Plate Problems"
- NRC Information Notice 91-18, “High-Energy Piping Failures Caused by Wall Thinning”
- NRC Information Notice 91-19, “Steam Generators Feedwater Distribution Piping Damage”
- NRC Information Notice 91-28, “Cracking in Feedwater System Piping”
- NRC Information Notice 91-38, “Thermal Stratification in Feedwater System Piping”
- NRC Information Notice 92-07, “Rapid Flow-Induced Erosion/Corrosion of Feedwater Piping”

- NRC Information Notice 93-20, "Thermal Fatigue Cracking of Feedwater Piping to Steam Generators"
- NRC Information Notice 93-21, "Summary of Observations Compiled During Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs"
- NRC Information Notice 94-59, "Accelerated Dealloying of Cast Aluminum-Bronze Valves Caused by Microbiologically Induced Corrosion"
- NRC Information Notice 94-79, "Microbiologically Influenced Corrosion of Emergency Diesel Generator Service Water Piping"
- NRC Information Notice 95-11, "Failure of Condensate Piping Because of Erosion/Corrosion at a Flow-Straightening Device"
- NRC Information Notice 99-19, "Rupture of the Shell Side of a Feedwater Heater at the Point Beach Plant"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.5.2.

3.5.3.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of Steam and Power Conversion Systems component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.5.2.

3.5.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.5.2. Tables 3.5-1 through 3.5-3 contain the results of the aging management review for the Steam and Power Conversion Systems and summarize the aging effects requiring management.

The aging effects requiring management are adequately managed by the following programs:

- Auxiliary Feedwater Pump Oil Coolers Inspection
- Auxiliary Feedwater Steam Piping Inspection Program
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Field Erected Tanks Internal Inspection
- Flow Accelerated Corrosion Program
- Galvanic Corrosion Susceptibility Inspection Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the Steam and Power Conversion Systems components listed in Tables 3.5-1 through 3.5-3 are maintained consistent with the current licensing basis for the period of extended operation.

TABLE 3.5-1
MAIN STEAM AND TURBINE GENERATORS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Unit 4 Main Steam Isolation Valve instrument air accumulator tanks	Pressure boundary	Carbon steel	Air/Gas	None
Main process piping: Valves Piping/fittings	Pressure boundary	Carbon steel	Treated water - secondary	Loss of material
Steam traps: Valves Piping/fittings Steam traps	Pressure boundary	Carbon steel	Treated water - secondary	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Treated water - secondary	Loss of material Cracking
Valves Tubing/fittings Filters Flex hoses Rupture disks	Pressure boundary	Stainless steel	Air/Gas	None
Instrument air 3-way valves	Pressure boundary	Brass	Air/Gas	None
Flow elements	Pressure boundary Throttling	Carbon steel	Treated water - secondary	Loss of material

TABLE 3.5-1 (continued)
MAIN STEAM AND TURBINE GENERATORS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Unit 4 Main Steam Isolation Valve instrument air accumulator tanks	Pressure boundary	Carbon steel	Outdoor	Loss of material
Main process piping: Valves Piping/fittings	Pressure boundary	Carbon steel	Containment air Outdoor	None[1]
			Borated water leaks	Loss of material
Steam traps: Valves Steam traps Piping/fittings	Pressure boundary	Carbon steel	Outdoor	Loss of material
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Containment air	None
Valves Tubing/fittings Filters Flex hoses Rupture disks	Pressure boundary	Stainless steel	Outdoor	None
Instrument air 3-way valves	Pressure boundary	Brass	Outdoor	None

NOTES: 1. Carbon steel components that normally operate at high temperatures are not susceptible to loss of material.
2. Flow Accelerated Corrosion Program addresses external general corrosion via use of radiographic examinat

TABLE 3.5-1 (continued)
MAIN STEAM AND TURBINE GENERATORS

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Flow elements	Pressure boundary Throttling	Carbon steel	Containment air Outdoor	None[1]
			Borated water leaks	Loss of material
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

NOTE: 1. Carbon steel components that normally operate at high temperatures are not susceptible to loss of material.

TABLE 3.5-2
FEEDWATER AND BLOWDOWN

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Demineralized water storage tank	Pressure boundary	Carbon steel	Air/Gas	Loss of material
			Treated water	Loss of material
Standby steam generator feedwater pumps	Pressure boundary	Carbon steel	Treated water	Loss of material
#6 Feedwater heater shells, tube sheets, and covers	Pressure boundary	Carbon steel	Treated water - secondary	Loss of material
#6 Feedwater heater tubes[1]	Pressure boundary	Stainless steel	Treated water - secondary (inside and outside diameters)	Loss of material Cracking

NOTE: 1. Heat transfer is not a license renewal intended function for this component.

TABLE 3.5-2 (continued)
FEEDWATER AND BLOWDOWN

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Piping/fittings (main feedwater and blowdown)	Pressure boundary	Carbon steel	Treated water - secondary	Loss of material
Tubing/fittings Thermowells (feedwater)	Pressure boundary	Stainless Steel	Treated water – secondary	Loss of material Cracking
Valves Piping/fittings Tubing/fittings Thermowells (blowdown)	Pressure boundary	Stainless steel	Treated water - secondary	Loss of material Cracking
Instrument air solenoid valves	Pressure boundary	Brass	Air/Gas	None
Orifices	Pressure boundary Throttling	Stainless steel	Treated water - secondary	Loss of material Cracking
Strainers	Pressure boundary	Stainless steel	Treated water	Loss of material

TABLE 3.5-2 (continued)
FEEDWATER AND BLOWDOWN

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Piping/fittings Tubing/fittings (standby steam generator feedwater pump suction)	Pressure boundary	Stainless steel	Treated water	Loss of material
Orifices	Pressure boundary	Stainless steel	Treated water	Loss of material

TABLE 3.5-2 (continued)
FEEDWATER AND BLOWDOWN

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Demineralized water storage tank	Pressure boundary	Carbon steel	Outdoor	Loss of material
Standby steam generator feedwater pumps	Pressure boundary	Carbon steel	Outdoor	Loss of material
#6 Feedwater heater shells and covers	Pressure boundary	Carbon steel	Outdoor	None[1]
Valves Piping/fittings (main feedwater and blowdown)	Pressure boundary	Carbon steel	Containment air	None[1]
			Borated water leaks	Loss of material
Tubing/fittings	Pressure boundary	Stainless steel	Containment air	None
Valves Piping/fittings (from standby steam generator feedwater pumps to main feedwater piping)	Pressure boundary	Carbon steel	Outdoor	Loss of material

NOTE: 1. Carbon steel components that normally operate at high temperatures are not susceptible to loss of material.

TABLE 3.5-2 (continued)
FEEDWATER AND BLOWDOWN

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves Piping/fittings (main feedwater and blowdown)	Pressure boundary	Carbon steel	Outdoor Indoor – not air-conditioned	None[1]
Valves Piping/fittings Tubing/fittings Thermowells (blowdown)	Pressure boundary	Stainless steel	Outdoor Indoor – not air-conditioned	None
Valves Piping/fittings Tubing/fittings Strainers (standby steam generator feedwater pump suction)	Pressure boundary	Stainless steel	Outdoor	Loss of material Cracking[2]
Piping (standby steam generator feedwater pump suction)	Pressure boundary	Stainless steel	Buried	None

NOTES: 1. Carbon steel components that normally operate at high temperatures are not susceptible to loss of material.
2. Plant experience has identified the potential for cracking in non-stress relieved heat affected zones of weld joir

TABLE 3.5-2 (continued)
FEEDWATER AND BLOWDOWN

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Instrument air solenoid valves	Pressure boundary	Brass	Outdoor	None
Orifices	Pressure boundary Throttling	Stainless steel	Outdoor	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Borated water leaks	Loss of mechanical closure integrity

TABLE 3.5-3
AUXILIARY FEEDWATER AND CONDENSATE STORAGE

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment				
Condensate storage tanks	Pressure boundary	Carbon steel	Air/Gas	Loss of material
			Treated water	Loss of material
Auxiliary feedwater pumps	Pressure boundary	Low alloy steel	Treated water	Loss of material
Auxiliary feedwater pump turbine casings	Pressure boundary	Carbon steel	Treated water - secondary Air/Gas	Loss of material
Auxiliary feedwater pump lube oil cooler and governor oil cooler tube sheets	Pressure boundary	Carbon steel	Treated water	Loss of material
			Lubricating oil	None

NOTE: 1. Galvanic corrosion only at carbon steel contact points with stainless steel, brass, and low alloy steel for these co

TABLE 3.5-3 (continued)
AUXILIARY FEEDWATER AND CONDENSATE STORAGE

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Auxiliary feedwater pump lube oil cooler and governor oil cooler channels and covers	Pressure boundary	Cast iron	Treated water	Loss of material
Auxiliary feedwater pump lube oil cooler and governor oil cooler shells Lube oil pump casings Lube oil reservoirs Piping/fittings	Pressure boundary	Carbon steel	Lubricating oil	None
Auxiliary feedwater pump lube oil cooler and governor oil cooler tube bundles	Pressure boundary Heat transfer	Admiralty brass Stainless steel	Treated water (inside diameter)	Loss of material
			Lubricating oil (outside diameter)	None
Valves Piping/fittings Tubing/fittings	Pressure boundary	Stainless steel	Treated water	Loss of material

TABLE 3.5-3 (continued)
AUXILIARY FEEDWATER AND CONDENSATE STORAGE

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
Internal Environment (continued)				
Valves Piping/fittings	Pressure boundary	Carbon steel Low alloy steel	Treated water	Loss of material
Valves Piping/ fittings Steam traps	Pressure boundary	Carbon steel	Treated water - secondary Air/Gas	Loss of material
Valves Piping/fittings (upstream of MOVs 3/4 1403, 1404, and 1405	Pressure boundary	Carbon steel	Treated water - secondary	Loss of material
Valves Piping/fittings Tubing/fittings Flex hoses Rupture disks	Pressure boundary	Stainless steel	Air/Gas	None
Orifices	Pressure boundary Throttling	Stainless steel Carbon steel	Treated water	Loss of material

NOTE: 1. Galvanic corrosion only at carbon steel contact points with stainless steel, brass, and low alloy steel for these co

TABLE 3.5-3 (continued)
AUXILIARY FEEDWATER AND CONDENSATE STORAGE

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment				
Condensate storage tanks	Pressure boundary	Carbon steel	Outdoor	Loss of material
Auxiliary feedwater pumps	Pressure boundary	Low alloy steel	Outdoor	Loss of material
Auxiliary feedwater pump turbine casings	Pressure boundary	Carbon steel	Outdoor	Loss of material
Auxiliary feedwater pump lube oil cooler and governor oil cooler shells and channels	Pressure boundary	Carbon steel Cast iron	Outdoor	Loss of material
Valves Piping/fittings Steam traps (non-insulated)	Pressure boundary	Carbon steel	Outdoor	Loss of material
Valves Piping/fittings (insulated)	Pressure boundary	Carbon steel	Outdoor	Loss of material

TABLE 3.5-3 (continued)
AUXILIARY FEEDWATER AND CONDENSATE STORAGE

Component / Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management
External Environment (continued)				
Valves Piping/fittings Tubing/fittings Orifices Rupture disks Flex hoses	Pressure boundary	Stainless steel	Outdoor	None
Orifices	Pressure boundary Throttling	Carbon steel	Outdoor	Loss of material
		Stainless steel	Outdoor	None
Bolting (mechanical closures)	Pressure boundary	Carbon steel	Outdoor	None

3.6 STRUCTURES AND STRUCTURAL COMPONENTS

Structures and their structural components and commodities that are within the scope of license renewal and subject to aging management reviews are discussed in Section 2.4 and summarized in Tables 3.6-2 through 3.6-20.

Determination of the aging effects applicable to structures and their structural components and commodities begins with identification of the aging effects defined in industry literature. From the set of aging effects, the component and commodity materials and operating environments define the aging effects for each structural component or commodity that is subject to an aging management review. These aging effects are validated by a review of industry and Turkey Point Units 3 and 4 operating experiences to provide reasonable assurance that the full set of aging effects are established for the aging management review.

Structural components inaccessible for inspection were evaluated for potential aging effects based on their environment as part of the aging management review. Several structural components that are inaccessible for visual inspection require aging management at Turkey Point. Examples include buried concrete, embedded steel, and structural components blocked by installed equipment or structures. Structural components inaccessible for inspection are managed by inspecting accessible structures with similar materials and environments for aging effects that may be indicative of aging effects for inaccessible structural components. The programs credited for managing aging effects of inaccessible structural components are the ASME Section XI, Subsection IWE Inservice Inspection Program and the Systems and Structures Monitoring Program. These programs are discussed in Appendix B.

3.6.1 CONTAINMENTS

The Westinghouse Generic Topical Report, WCAP-14756, “Aging Management Evaluation for Pressurized Water Reactor Containment Structure,” is not credited and is not incorporated by reference in this Application.

The Containments are divided into two structural classifications, Containment structure, and Containment internal structural components. The components of the structural classifications are grouped by material or function. The Containment structure component groupings are concrete, steel, and post-tensioning system. The Containment internal structural components groupings are concrete and steel.

3.6.1.1 CONTAINMENT STRUCTURE CONCRETE COMPONENTS

The Containment structure concrete components are:

- dome
- cylinder wall
- floor
- foundation mat

3.6.1.1.1 MATERIALS AND ENVIRONMENT

The Containment structure concrete components were designed and constructed in accordance with ACI and American Society for Testing and Materials (ASTM) standards to provide good quality, dense, low permeability concrete. The codes and standards used for design and fabrication of the Containment structure concrete components are provided in Turkey Point UFSAR Subsections 5.1.2 and 5.1.6.

Containment structure concrete components are exposed to several different environments depending on their location. Below grade (buried) Containment structure concrete components can be either above or below the groundwater elevation. Containment structure concrete components that are below grade and above groundwater are exposed to soil/fill. Containment structure concrete components that are below groundwater are exposed to soil/fill and groundwater. The groundwater chemistry is relevant in the determination of the degradation of below groundwater Containment structure concrete components. Based on a review of the Turkey Point Final Environmental Statement [Reference 3.6-1], the groundwater parameters for chlorides and sulfates exceed the threshold limits where

degradation may occur. Above grade external surfaces of the Containment structure are exposed to indoor – not air conditioned and outdoor environments. Internal components of the Containment structure are exposed to the Containment air environment (see Table 3.0-2).

3.6.1.1.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of the intended function(s) for Containment structure concrete components are loss of material, cracking, and change in material properties. Each is discussed below.

LOSS OF MATERIAL

Loss of material is manifested in Containment structure concrete components as scaling, spalling, pitting, and erosion. Aging mechanisms that can lead to loss of material are freeze-thaw, abrasion and cavitation, elevated temperature, aggressive chemical attack, and corrosion of reinforcing and embedded steel.

Freeze-thaw is considered an aging mechanism for concrete structural components that are exposed to severe weather conditions of numerous freeze-thaw cycles with significant amounts of winter rainfall. Turkey Point is located in a subtropical climate with long, warm summers accompanied by abundant rainfall and mild, dry winters with negligible freeze-thaw cycles. Therefore, freeze-thaw is not an aging mechanism that can lead to loss of material for Containment structure concrete components.

Abrasion and cavitation is an aging mechanism that occurs only in concrete structures that are continually exposed to flowing water. The Intake Structure concrete components located below the intake canal water level and the concrete intake cooling water piping are the only concrete components exposed to flowing water. Therefore, abrasion and cavitation is not an aging mechanism that can lead to loss of material for Containment structure concrete components.

Elevated temperature was evaluated as an aging mechanism for Containment structure concrete components. The concrete around hot piping penetrations is subject to extended local heatup. The penetrations were designed and constructed to maintain concrete components below the degradation threshold and localized temperature limits of the ACI standards without forced ventilation. No other Containment structure concrete components are exposed to elevated temperature.

Therefore, elevated temperature is not an aging mechanism that can lead to loss of material for Containment structure concrete components.

Aggressive chemical attack and corrosion of reinforcing and embedded steel are aging mechanisms for Containment structure concrete components exposed to groundwater. As discussed in Subsection 2.4.1.1, design features such as waterproofing membranes and water stops are utilized underneath the foundation mat and outside the lower portions of the Containment structure wall. In addition, a cathodic protection system is provided as a design feature to control galvanic corrosion of the reinforcing and embedded steel as a preventive measure. However, the waterproofing membranes, water stops, and cathodic protection system are not credited in the determination of aging effects requiring management.

Based on the above, loss of material due to aggressive chemical attack and corrosion of reinforcing and embedded steel is an aging effect requiring management for Containment structure concrete components below groundwater elevation.

CRACKING

Cracking is manifested in Containment structure concrete components as complete or incomplete separation of the concrete into two or more parts. Aging mechanisms that can lead to cracking are freeze-thaw, reactions with aggregates, shrinkage, settlement, fatigue, and elevated temperature.

As discussed previously, freeze-thaw is not an aging mechanism that can lead to cracking for Containment structure concrete components.

Turkey Point concrete components were constructed using non-reactive aggregates whose acceptability was based on established industry standards and ASTM tests. Therefore, reaction with aggregates is not an aging mechanism that can lead to cracking for Containment structure concrete components.

When concrete is exposed to air, large portions of the free water evaporate, causing shrinkage. At Turkey Point, the initial concrete water content was kept low, low slump concrete was used, and adequate steel reinforcement was provided, which all minimize shrinkage. Based on industry information, 100% of shrinkage occurs within 20 years. Turkey Point concrete structures and concrete components were constructed more than 20 years ago. Therefore, concrete shrinkage is not an aging

mechanism that can lead to cracking for Containment structure concrete components.

Settlement is based directly on the physical properties of a structure's foundation material. The most pronounced settlement is evidenced in the first several months after construction. Turkey Point concrete structures are founded on fossiliferous limestone bedrock (Miami Oolite) with crushed limestone fill. This foundation material is suitable for foundation systems with no significant structural settlement expected. Therefore, settlement is not an aging mechanism that can lead to cracking for Containment structure concrete components.

Fatigue is a progressive degradation problem for materials subjected to cyclic application of loads that are less than the maximum allowable static loads. Turkey Point concrete components are designed in accordance with ACI standards and have good low-cycle fatigue properties. Although some concrete components are subject to high cycles of low-level repeated load, these components were designed in accordance with ACI standards, which limit the maximum design stress to less than 50% of the static stress of the concrete. The concrete fatigue strength is about 55% of its static strength at extremely high cycles ($>10^7$ cycles) of loading. Therefore, fatigue is not an aging mechanism that can lead to cracking for Containment structure concrete components.

As discussed previously, the Containment structure concrete components do not exceed established threshold limits for degradation due to elevated temperature. Therefore, elevated temperature is not an aging mechanism that can lead to cracking.

Based on the above, cracking is not an aging effect requiring management for containment structure concrete components.

CHANGE IN MATERIAL PROPERTIES

Change in material properties is manifested in concrete as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength. Aging mechanisms that can lead to a change in material properties are leaching, creep, elevated temperature, irradiation embrittlement, and aggressive chemical attack.

Leaching of calcium hydroxide is observed on concrete that is alternately wetted and dried. White deposits that are left on the surface of the concrete are a solution of water, free lime from the concrete, and carbon dioxide that is readily seen on the surface of the concrete. Turkey Point concrete structures and concrete components are constructed of dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the guidance provided by the ACI, and when implemented, degradation caused by leaching of calcium hydroxide is not significant. Therefore, leaching is not an aging mechanism that can lead to change in material properties for Containment structure concrete components.

Creep is significant when new concrete is subjected to load and decreases exponentially with time; and any degradation is noticeable in the first few years. All reinforced concrete components were designed based on the ACI working stress design method. Creep in all concrete components is minimal because of low compressive stresses in concrete and the use of high strength concrete. In addition, creep proceeds at a decreasing rate with age, 96% of creep has occurred within 30 years; and therefore, concrete creep is not an aging mechanism that can lead to change in material properties for Containment structure concrete components.

As discussed previously, the Containment structure concrete components do not exceed established threshold limits for degradation due to elevated temperature. Therefore, elevated temperature is not an aging mechanism that can lead to change in material properties for Containment structure concrete components.

Irradiation embrittlement was evaluated as an aging mechanism that could lead to change in material properties. Shielding from the water in the reactor core and the reactor vessel reduces the neutron flux, resulting in levels of accumulated exposure that are far below the levels necessary to cause degradation. The maximum gamma dose evaluated through the period of extended operation is below the dose required for radiation degradation. Therefore, irradiation embrittlement is not an aging mechanism that can lead to change in material properties for Containment structure concrete components.

As discussed above, aggressive chemical attack is an aging mechanism that can lead to change in material properties for concrete components below groundwater elevation.

Based on the above, change in material properties due to aggressive chemical attack is an aging effect requiring management for Containment structure concrete components below groundwater elevation.

3.6.1.1.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to Containment structural concrete components includes the following:

- NRC Information Notice 97-11, “Cement Erosion from Containment Subfoundations at Nuclear Power Plants”
- NRC Information Notice 98-26, ”Settlement Monitoring and Inspection of Plant Structures Affected by Degradation of Porous Concrete Subfoundations”
- NUREG-1522, ”Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures”
- NUREG/CR-4652, ”Concrete Component Aging and its Significance Relative to Life Extension of Nuclear Power Plants”
- NUREG/CR-6598, ”An Investigation of Tendon Sheathing Filler Migration into Concrete”
- NUREG/CP-0100, Prasad, N., et al., “Concrete Degradation Monitoring and Evaluation,” Proceedings of the International Nuclear Power Plant Aging Symposium, August 30 – September 1, 1998

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.6.1.1.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of Containment structure concrete component aging, in addition to interviews with responsible engineering personnel. No aging effects

requiring management were identified from this review beyond those identified in Subsection 3.6.1.1.2.

3.6.1.1.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.6.1.1.2. Table 3.6-2 contains the results of the aging management review for the Containments, and summarizes the aging effects requiring management for Containment structure concrete components.

The aging effects requiring management are adequately managed by the following program:

- Systems and Structures Monitoring Program

Based on the evaluation provided in Appendix B for the program above, aging effects are adequately managed so that the intended functions of the Containment structure concrete components listed in Table 3.6-2 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.1.2 CONTAINMENT STRUCTURE STEEL COMPONENTS

The Containment structure steel components are:

- liners (including the liner plate, anchors/embedments/attachments, leak chase channels, and moisture barriers)
- penetrations [including mechanical piping, mechanical ventilation, and steel portions (pressure boundary) of the electrical penetration assemblies]
- airlocks and hatches (personnel hatch, equipment hatch, escape hatch, including seals and gaskets)
- fuel transfer tube blind flanges (Note: The fuel transfer tubes, penetration sleeves, and gate valves are addressed in Subsection 3.6.2.2.)

3.6.1.2.1 MATERIALS AND ENVIRONMENT

The Containment structure steel components were designed and constructed in accordance with ASME Section III – 1965 for the pressure boundary, and American Institute of Steel Construction (AISC), Manual of Steel Construction, for structural steel. The codes and standards used for design and fabrication of the Containment

structure steel components are provided in Turkey Point UFSAR Subsections 5.1.2 and 5.1.6.

The gaskets, seals, and moisture barriers that protect the Containment structure steel components are elastomers.

The Containment structure steel components are exposed to containment air, indoor – not air conditioned, outdoor, and embedded/encased environments and potential borated water leaks (see Table 3.0-2).

3.6.1.2.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of the intended function(s) for Containment structure steel components are loss of material, cracking, and change in material properties. Each is discussed below.

LOSS OF MATERIAL

Aging mechanisms that can lead to loss of material are material compatibility, mechanical wear, corrosion, and aggressive chemical attack.

For electrical penetrations, material compatibility tests have been conducted to determine the effects of polymer outgassing on metal corrosion. The test results indicate that the metals and alloys had insignificant amounts of corrosion when exposed to outgasses. The testing conditions are significantly more severe than those experienced by the materials in service in the Turkey Point Containment structures. Therefore, material compatibility is not an aging mechanism that can lead to loss of material for Containment structure steel components.

Mechanical wear can occur at the blind flange closure, located at the inside containment end of each fuel transfer tube that functions as part of the containment system pressure boundary. The moving parts may experience some mechanical wear over time; however, any wear that would interfere with the capability of the blind flange to be removed and replaced would be detected during normal refueling operations and would be addressed in the corrective action program. The Turkey Point airlocks and hatches are opened by mechanical means. The airlocks are designed with an interlock system that ensures the containment pressure boundary is maintained when the airlocks are used for access during reactor operation. The interlock system permits only one of the two airlock doors on each airlock to be open at one time. There are no other structural component metal surfaces in contact with

and moved frequently against each other. Therefore, mechanical wear is not an aging mechanism that can lead to loss of material for Containment structure steel components.

Loss of material in steel may be caused by corrosion. This can be seen as material dissolution, corrosion product buildup, and pitting. It can be uniform or localized. Therefore, corrosion is an aging mechanisms that can lead to loss of material in Containment structure steel components at Turkey Point.

Aggressive chemical attack due to boric acid is an aging mechanism for Containment structure steel components. This form of corrosion is typically localized and is a result of leakage from the Reactor Coolant and other borated systems that can concentrate boric acid and lead to significant material loss of carbon steel components. Although this type of corrosion is event driven (boric acid leak), boric acid corrosion was evaluated as an aging mechanism that can lead to loss of material for Containment structure steel components.

Based on the above, loss of material due to corrosion and aggressive chemical attack, is an aging effect requiring management for Containment structure steel components.

CRACKING

Aging mechanisms that can lead to cracking of Containment structural steel components are stress corrosion and fatigue.

Stress corrosion cracking is an age-related degradation mechanism that affects stainless steels but becomes significant only if tensile stresses and a corrosive environment exist. The stresses may be either applied (external) or residual (internal). The stress corrosion cracks themselves may be either transgranular or intergranular, depending upon the metal and the corrosive agent. As is normal in all cracking, the cracks are perpendicular to the tensile stress. Usually there is little or no obvious visual evidence of corrosion. Stress corrosion cracking is not applicable for the carbon-steel liner plate and, therefore, is not an aging mechanism that can lead to cracking for Containment structure steel components.

Fatigue is a progressive degradation problem for materials subjected to cyclic application of loads that are less than the maximum allowable static loads. For steel components, fatigue is the cumulative effect of microstructural localized plastic deformation in the material section that results each time a stress cycle of sufficient

magnitude occurs. Cracking due to fatigue of the liner plate and penetrations is a Time-Limited Aging Analysis and is discussed in Section 4.6.

Based on the above, cracking is not an aging effect requiring management for Containment structure steel components.

CHANGE IN MATERIAL PROPERTIES

Change in material properties is manifested in Containment structure steel components as a reduction or increase in yield strength, reduction in modulus of elasticity, reduction in ultimate tensile ductility, and an increase in ductile to brittle transition temperature. Aging mechanisms that can lead to change in material properties are elevated temperature, irradiation embrittlement, and embrittlement and permanent set of elastomers.

Elevated temperature was evaluated as an aging mechanism. For Containment structure steel components, a temperature of 700ºF must be reached before significant reductions in the yield strength and modulus of elasticity occur. Since the operating temperatures of the Containment structure steel components are below the threshold temperature, the temperature at which structural integrity of the steel would begin to be affected will not be reached. Therefore, elevated temperature is not an aging mechanism that can lead to change in material properties for Containment structure steel components.

As discussed in Subsection 3.6.1.1.2, irradiation embrittlement was evaluated as an aging mechanism that could lead to change in material properties. Shielding from the water in the reactor core and the reactor vessel reduces the neutron flux, resulting in levels of accumulated exposure that are far below the degradation threshold. Therefore, irradiation embrittlement is not an aging mechanism that can lead to change in material properties for Containment structure steel components.

Elastomers are used as gaskets and seals for airlocks and hatches and as moisture barriers for the Containment liner. Embrittlement and permanent set of elastomers can lead to change in material properties, which can result in loss of pressure retention capability.

Based on the above, change in material properties due to embrittlement and permanent set of elastomers associated with Containment structure steel components is an aging effect requiring management.

3.6.1.2.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to Containment structural steel components includes the following:

- NRC Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants”
- NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to the Reactor Coolant System”
- NRC Bulletin 88-11, “Pressurizer Surge Line Thermal Stratification”
- NRC Generic Letter 80-08, “Examination of Containment Liner Penetration Welds”
- NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants”
- NRC Generic Letter 98-04, “Potential Degradation of the Emergency Core Cooling System and Containment Spray System after a Loss of Coolant Accident because of Construction and Protective Coating Deficiencies and Foreign Material in the Containment”
- NRC Information Notice 86-99 and Information Notice 86-99 Supplement1, “Degradation of Steel Containments”
- NRC Information Notice 88-80, “Unexpected Piping Movement Attributed to Thermal Stratification”
- NRC Information Notice 89-79, “Degraded Coatings and Corrosion of Steel Containment Vessels”
- NRC Information Notice 93-25, “Electrical Penetration Assembly Degradation”
- NRC Information Notice 97-10, “Liner Plate Corrosion in Concrete Containments”
- NRC Information Notice 97-13, “Deficient Conditions Associated with Protective Coatings at Nuclear Power Plants”

- NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.6.1.2.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of Containment structure steel component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.1.2.2.

3.6.1.2.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.6.1.2.2. Table 3.6-2 contains the results of the aging management review for the Containments and summarizes the aging effects requiring management for Containment structure steel components.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsection IWE Inservice Inspection Program
- Boric Acid Wastage Surveillance Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that that the intended functions of the Containment structure steel components listed in Table 3.6-2 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.1.3 CONTAINMENT STRUCTURE POST - TENSIONING SYSTEM

The Containment structure post-tensioning system components are:

- tendon wires
- tendon anchorage

3.6.1.3.1 MATERIALS AND ENVIRONMENT

The codes and standards used for design and fabrication of the Containment structure post-tensioning system are provided in UFSAR Section 5.0 and Appendices 5A and 5B. The post-tensioning system is described in UFSAR Section 5.1.

Each tendon is housed in spirally wrapped, corrugated, thin wall sheathing and capped at each end with a sheathing filler cap. After fabrication, the tendon is shop dipped in grease. The tendon sheathing provides the channel in the concrete through which the tendon is pulled, and contains the tendon sheathing filler material (grease). The tendon anchorages and tendon wires are contained in the sheathing filler material (grease).

3.6.1.3.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of intended function(s) for the Containment post-tensioning system components are loss of material and loss of prestress. Each is discussed below.

LOSS OF MATERIAL

Loss of material may be caused by corrosion.

The effects of corrosion must be considered for both the tendon wires within the grease-filled conduits and for the anchorages providing the tendon terminations. Stressed components of the Containment structure post-tensioning system are normally well protected against corrosion. The tendon and anchorages are enclosed with tendon sheathing and end caps that are filled with grease.

Based on the above, loss of material due to corrosion is an aging mechanism requiring management for Containment structure post-tensioning system components.

LOSS OF PRESTRESS

Aging mechanisms that can lead to loss of prestress are: elevated temperatures, irradiation, stress relaxation of the prestressing wire, shrinkage, creep or elastic deformation of the concrete, anchorage seating losses, and tendon friction. Loss of prestress of the Containment structure post-tensioning system is a Time-Limited Aging Analysis and is discussed in Section 4.5.

3.6.1.3.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to Containment structure post-tensioning system components includes the following:

- NRC Information Notice 85-10, “Post-Tensioned Containment Tendon Anchor Head Failure”
- NRC Information Notice 91-80, “Failure of Anchor Head Threads on Post-Tensioning System During Surveillance Inspection”
- NRC Information Notice 99-10, “Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments”
- NUREG-1522, “Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures”
- NUREG/CR-0092, “Corrosion of Steel Tendons in Concrete Pressure Vessels, Review of Recent Literature and Experimental Investigations”
- NUREG/CR-2719, “Evaluation of Inservice Inspections of Greased Prestressing Tendons”
- NUREG/CR-6598, “An Investigation of Tendon Sheathing Filler Migration into Concrete”

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.6.1.3.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of Containment structure post-tensioning system component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.1.3.2.

3.6.1.3.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.6.1.3.2. Table 3.6-2 contains the results of the aging management review for the Containments and summarizes the aging effects requiring management for Containment structure post-tensioning system components.

The aging effects requiring management are adequately managed by the following program:

- ASME Section XI, Subsection IWL Inservice Inspection Program

Based on the evaluation provided in Appendix B for the program above, aging effects are adequately managed so that that the intended functions for Containment structure post-tensioning system components listed in Table 3.6-2 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.1.4 CONTAINMENT INTERNAL STRUCTURAL CONCRETE COMPONENTS

The Containment internal structural concrete components are:

- reinforced concrete primary shield walls
- reinforced concrete secondary shield walls
- reinforced concrete upper secondary compartment walls (steam generator and pressurizer cubicles)
- reinforced concrete refueling cavity walls
- reinforced concrete containment sumps
- reinforced concrete equipment pads
- reinforced concrete missile shields
- reinforced concrete beams, floors, mats, and walls
- reinforced concrete curbs

3.6.1.4.1 MATERIALS AND ENVIRONMENT

The Containment internal structural concrete components were designed and constructed in accordance with ACI and ASTM standards to provide good quality,

dense, low permeability concrete. The codes and standards used for design and fabrication of the Containment internal structural concrete components are provided in Turkey Point UFSAR Subsections 5.1.2 and 5.1.6.

The Containment internal structural concrete components are exposed to the Containment air environment (see Table 3.0-2).

3.6.1.4.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of the intended function(s) for Containment internal structural concrete components are loss of material, cracking, and change in material properties. Each is discussed below.

LOSS OF MATERIAL

Loss of material is manifested in Containment internal structural concrete components as scaling, spalling, pitting, and erosion. Aging mechanisms that can lead to loss of material are freeze-thaw, abrasion and cavitation, aggressive chemical attack, corrosion of reinforcing and embedded steel, and elevated temperature.

As discussed in Section 3.6.1.1.2, freeze-thaw and abrasion and cavitation were evaluated for concrete components at Turkey Point and determined not to be aging mechanisms that can lead to loss of material for Containment internal structural concrete components.

There are no Containment internal structural concrete components exposed to the groundwater and, therefore, aggressive chemical attack and corrosion of reinforcing and embedded steel are not aging mechanisms that can lead to loss of material for Containment internal structural concrete components.

Elevated temperature was evaluated as an aging mechanism. The primary shield wall concrete is subject to extended local heatup. The combination of component insulation, ventilation systems, concrete wall design, and plant Technical Specifications maintains the temperatures in the primary shield walls below degradation threshold and localized temperature limits of ACI standards. No other concrete components are exposed to elevated temperature. Therefore, elevated temperature is not an aging mechanism that can lead to loss of material for Containment internal structural concrete components.

Based on the above, loss of material is not an aging effect requiring management for Containment internal structural concrete components.

CRACKING

Cracking is manifested in Containment internal structural concrete components as complete or incomplete separation of the concrete into two or more parts. Aging mechanisms that can lead to cracking are freeze-thaw, reactions with aggregates, shrinkage, settlement, fatigue, and elevated temperature.

As previously discussed in Section 3.6.1.1.2, freeze-thaw, reactions with aggregates, shrinkage, settlement, and fatigue were evaluated for concrete components at Turkey Point and are not aging mechanisms that can lead to cracking for Containment internal structural concrete components.

As discussed in Subsection 3.6.1.1.2, the concrete components do not exceed established threshold limits for degradation due to elevated temperature. Therefore, elevated temperature is not an aging mechanism that can lead to cracking for Containment internal structural concrete components.

Based on the above, cracking is not an aging effect requiring management for Containment internal structural concrete components.

CHANGE IN MATERIAL PROPERTIES

Change in material properties is manifested in concrete as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength. Aging mechanisms that can lead to a change in material properties are leaching, creep, aggressive chemical attack, irradiation embrittlement, and elevated temperature.

As previously discussed in Section 3.6.1.1.2, leaching and creep were evaluated for concrete components at Turkey Point and are not aging mechanisms that can lead to change in material properties for Containment internal structural concrete components.

There are no internal components exposed to the groundwater and, therefore, aggressive chemical attack is not an aging mechanism that can lead to change in material properties for Containment internal structural concrete components.

Irradiation embrittlement was evaluated as an aging mechanism that could lead to change in material properties. Shielding from the water in the reactor core and the reactor vessel reduces the neutron flux, resulting in levels of accumulated exposure that are far below the degradation threshold. Neutron flux levels at the primary shield wall are well below the threshold levels for age-related radiation or radiation heating degradation. The maximum gamma dose evaluated through the period of extended operation is below the dose required for radiation degradation. Therefore, irradiation embrittlement is not an aging mechanism that can lead to change in material properties for Containment internal structural concrete components.

As discussed previously, the Containment internal structural concrete components do not exceed established threshold limits for degradation due to elevated temperature. Therefore, elevated temperature is not an aging mechanism that can lead to change in material properties for Containment internal structural concrete components.

Based on the above, change in material properties is not an aging effect requiring management for Containment internal structural concrete components.

3.6.1.4.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history that included reviews of the NRC generic communications and licensee event reports was performed to validate the set of aging mechanisms that cause the aging effects in Subsection 3.6.1.4.2. The industry correspondence that was found applicable to the Containment internal structural concrete components is provided in Subsection 3.6.1.1.3.

No aging effects requiring management were identified from the documents listed in Subsection 3.6.1.1.3 beyond those already identified in Subsection 3.6.1.4.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of Containment internal structural concrete component aging, in addition to interviews with responsible engineering personnel.

No aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.1.4.2.

3.6.1.4.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no aging effects requiring management as discussed in Subsection 3.6.1.4.2. Table 3.6-2 contains the results of the aging management review for the Containments and indicates there are no aging effects requiring management for Containment internal structural concrete components.

Based on the aging management review discussed above, the intended functions for the Containment internal structural concrete components will be maintained consistent with the current licensing basis for the period of extended operation.

3.6.1.5 CONTAINMENT INTERNAL STRUCTURAL STEEL COMPONENTS

The Containment internal structural steel components are:

- equipment component supports
- heating, ventilation and air-conditioning (HVAC) ductwork supports
- piping supports
- pipe whip restraints (excluding Reactor Coolant System loop piping within the scope of Leak Before Break – Subsection 4.2.3)
- cable trays, conduits, and supports
- electrical and instrument panels and enclosures
- anchorages/embedments exposed surfaces
- instrument line supports
- instrument racks and frames
- structural steel beams and columns
- stairs, platforms, and grating
- sump screens
- Lubrite plates
- radiant energy shields

- polar crane
- Reactor Coolant System supports (including reactor vessel supports, steam generator supports, reactor coolant pump supports, pressurizer supports, and the surge line support)
- non-safety related piping between class break and anchor

Table 3.6-2 contains a list of Containment internal structural steel components addressed by this aging management review. Note: The refueling pool liner and structural steel components associated with spent fuel handling equipment located inside the Containment are addressed in Subsection 3.6.2.

3.6.1.5.1 MATERIALS AND ENVIRONMENT

The Containment internal structural steel component design complies with AISC Manual of Steel Construction for the structural steel. The codes and standards used for design and fabrication of the Containment internal structural steel components are provided in Turkey Point UFSAR Subsections 5.1.2 and 5.1.6.

The materials of construction for the Reactor Coolant System supports include structural steels, low alloy steels, and carbon steel pipe. The primary bolting material is carbon steel. The Reactor Coolant System supports are described in Turkey Point UFSAR Section 4.2.

The Containment internal structural steel components are exposed to the containment air environment. In addition, Containment internal structural steel components may be exposed to treated water and treated water – borated environments and potential borated water leaks (see Table 3.0-2).

Pipe segments beyond the safety-related/non-safety related boundaries are constructed of carbon and stainless steel and consist of piping and inline components. The external surfaces of these pipe segments are exposed to the containment air environment. Internal environments of the pipe segments are the same as the internal environments for the systems in which they are installed.

3.6.1.5.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of the intended function(s) for Containment internal structural steel components are loss of material, cracking, and change in material properties. Each is discussed below.

LOSS OF MATERIAL

Aging mechanisms that can lead to loss of material for Containment internal structural steel components are material compatibility, mechanical wear, corrosion, and aggressive chemical attack.

As previously discussed in Subsection 3.6.1.2.2, material compatibility and mechanical wear were evaluated for structural steel components at Turkey Point and determined not to be aging mechanisms that can lead to loss of material for Containment internal structural steel components.

Loss of material in steel may be caused by corrosion. This can be seen as material dissolution, corrosion product buildup, and pitting. It can be uniform or localized.

Aggressive chemical attack due to boric acid is an aging mechanism for Containment internal structural steel components. This form of corrosion is typically localized and is a result of leakage from the Reactor Coolant and other borated systems that can concentrate boric acid and lead to significant material loss of carbon steel components. Although this type of corrosion is event driven (boric acid leak), boric acid corrosion was evaluated as an aging mechanism at Turkey Point.

Based on the above, loss of material due to corrosion and aggressive chemical attack is an aging effect requiring management for Containment internal structural steel components.

CRACKING

Aging mechanisms that can lead to cracking of structural steel components are fatigue and stress corrosion cracking.

For steel components, fatigue is the cumulative effect of microstructural localized plastic deformation in the material section that results each time a stress cycle of sufficient magnitude occurs. Steel components in the containment subjected to high-cycle ($>10^5$ cycles) loading conditions were designed in accordance with AISC standards. Fatigue degradation will have no adverse effects on the continued intended function(s) performance during the license renewal term and no further evaluation for steel components in the Containment is required. Fatigue is not an aging mechanism that can lead to cracking for Containment internal structural steel components.

Stress corrosion cracking is a localized, non-ductile cracking failure from an unfavorable combination of sustained tensile stress, material condition, and environment. Stress corrosion cracking can only occur when all three conditions exist simultaneously. At Turkey Point, it was determined that the combination of conditions required for stress corrosion cracking do not exist simultaneously, therefore, stress corrosion cracking cannot occur in Containment internal structural steel components.

Based on the above, cracking is not an aging effect requiring management for Containment internal structural steel components.

CHANGE IN MATERIAL PROPERTIES

Aging mechanisms that can lead to change in material properties are elevated temperature, irradiation embrittlement, and creep and stress relaxation.

Elevated temperature was evaluated as an aging mechanism. For Containment internal structural steel components, a temperature of 700ºF must be reached before significant reductions in the yield strength and modulus of elasticity occur. Since the operating temperatures of the Containment internal structural steel components are below the threshold temperature, the temperature at which structural integrity of the steel would begin to be affected will not be reached. Therefore, elevated temperature is not an aging mechanism that can lead to change in material properties for Containment internal structural steel components.

Irradiation embrittlement was evaluated as an aging mechanism that could lead to change in material properties. Shielding from the water in the reactor core and the reactor vessel reduces the neutron flux, resulting in levels of accumulated exposure that are far below the degradation threshold. Therefore, irradiation embrittlement is not an aging mechanism that can lead to change in material properties for Containment internal structural steel components.

Creep is a continuous physical deformation with time for a metal under a constant applied stress. Stress relaxation is a reduction in stress with time under a given constant strain. Creep and stress relaxation are strongly affected by temperature. Generally, creep and stress relaxation are insignificant when the service temperature to melting temperature ratio is less than 0.5. Since the temperature in the Reactor Coolant System supports is below 650ºF, creep and stress relaxation do not cause effects requiring aging management for Containment internal structural steel components.

Based on the above, change in material properties is not an aging effect requiring management for Containment internal structural steel components.

3.6.1.5.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to Containment internal structural steel components is provided in Subsection 3.6.1.2.3.

No aging effects requiring management were identified from the documents listed in Subsection 3.6.1.2.3 beyond those already identified in Subsection 3.6.1.5.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of Containment internal structural steel component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.1.5.2.

3.6.1.5.4 REACTOR COOLANT SYSTEM SUPPORTS TECHNICAL REPORT APPLICABILITY

Westinghouse Generic Topical Report, WCAP-14422, “License Renewal Evaluation: Aging Management for Reactor Coolant System Supports” [Reference 3.6-2], has been submitted to the NRC for review and approval. The NRC issued a draft safety evaluation report for WCAP-14422 [Reference 3.6-3] on February 25, 2000. This aging management review is based on that Generic Technical Report, however, WCAP-14422 is not incorporated by reference in this Application.

Turkey Point reviewed the current design and operation of the Reactor Coolant System supports using the process described in Subsection 2.3.1.1.1 and confirmed that the operating environments used in the design of the Turkey Point Reactor Coolant System supports are consistent with the description contained in

WCAP-14422. The Reactor Coolant System supports contain materials beyond those listed in Table 2-4 of WCAP-14422. These additional materials were included in the Reactor Coolant System supports aging management review described in Subsection 3.6.1.5. The component intended functions for the Turkey Point Reactor Coolant System supports are consistent with the intended functions identified in WCAP-14422.

As a result of the NRC review of WCAP-14422, several open items and license renewal applicant action items were identified. These open items and applicant action items are described in the draft NRC safety evaluation of WCAP-14422. Turkey Point-specific responses to those open items and applicant action items relevant to the identification of Reactor Coolant System supports subject to aging management review are provided in Tables 2.4-1 and 2.4-2. Major component supports are described in UFSAR Subsection 5.1.9.

Based on the results of Turkey Point's review to identify Time-Limited Aging Analyses (see Section 4.0), there are no Time-Limited Aging Analyses associated with Reactor Coolant System supports. This is consistent with the discussion of Time-Limited Aging Analyses in WCAP-14422.

The aging management programs described in WCAP-14422 include six attributes and are established on an aging mechanism basis. The Turkey Point aging management programs referred to in this aging management review and described in Appendix B contain ten attributes, and are established on a program basis.

Additionally, WCAP-14422 identifies stress corrosion cracking of support bolting as an aging effect requiring management. Based on the discussion in Subsection 3.6.1.5.2 under "Cracking," Turkey Point has concluded that stress corrosion cracking of support bolting does not require aging management.

3.6.1.5.5 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.6.1.5.2. Table 3.6-2 contains the results of the aging management review for the Containments and summarizes the aging effects requiring management for Containment internal structural steel components.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsection IWF Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that that the intended functions of the Containment internal structural steel components listed in Table 3.6-2 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.2 OTHER STRUCTURES

This aging management review identifies and evaluates aging effects on Turkey Point passive, long-lived structures and structural components (other than the Containments and selected structural components). Structures and structural components within the scope of license renewal and subject to aging management reviews are discussed in Section 2.4 and include:

- Auxiliary Building
- Cold Chemistry Lab
- Control Building
- Cooling Water Canals
- Diesel Driven Fire Pump Enclosure
- Discharge Structure
- Electrical Penetration Rooms
- Emergency Diesel Generator Buildings
- Fire Protection Monitoring Station
- Fire Rated Assemblies
- Intake Structure
- Main Steam and Feedwater Platforms
- Plant Vent Stack
- Spent Fuel Storage and Handling
- Turbine Building
- Turbine Gantry Cranes
- Turkey Point Units 1 and 2 Chimneys
- Yard Structures

Tables 3.6-3 through 3.6-20 contain the specific structural component and commodity groups, materials, intended functions, environments, aging effects, and aging management programs for each of the structures listed above. Structural components are grouped by material and environment for each structure. The structural component groups are:

- Steel in air
- Steel in fluid
- Concrete
- Miscellaneous

3.6.2.1 STEEL IN AIR STRUCTURAL COMPONENTS

Steel in air structural components include:

- framing, bracing, and connections
- decking, grating, and checkered plate
- stairs and ladders
- exposed anchors and embedments
- piping, duct, and component supports
- non-safety related piping between class break and anchor
- crane rails and girders
- cable trays, conduits, and electrical enclosures
- cable tray, conduit, and electrical enclosure supports
- instrumentation supports
- instrument racks and frames

3.6.2.1.1 MATERIALS AND ENVIRONMENT

Steel in air structural components were designed and constructed in accordance with AISC standards. The codes and standards used for the design and fabrication are identified in UFSAR Chapter 5. Steel in air structural components are constructed of painted or galvanized carbon steel and stainless steel. Turkey Point steel in air structural components are exposed to environments of containment air, outdoor, indoor – not air conditioned, indoor – air conditioned, and potential borated water leaks (see Table 3.0-2). The specific materials and environments for steel in air structural components for each structure are contained in Tables 3.6-3 through 3.6-20.

Pipe segments beyond the safety-related/non-safety related boundaries are constructed of carbon and stainless steel and consist of piping and inline

components. The external surfaces of these pipe segments are exposed to the Indoor – not air conditioned and outdoor environments and potential borated water leaks. Internal environments of the pipe segments are the same as the internal environments for the systems in which they are installed.

3.6.2.1.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of the intended function(s) of steel in air structural components are loss of material, cracking, and change in material properties. Each is discussed below.

LOSS OF MATERIAL

Aging mechanisms that can lead to loss of material are mechanical wear, corrosion, and aggressive chemical attack. This may be seen as material dissolution, corrosion product buildup, and pitting. Loss of material may be uniform or localized.

Mechanical wear is associated with close fitting mechanical components and is not applicable to structural steel. Accordingly, mechanical wear is not an aging mechanism that can lead to loss of material in steel in air structural components.

Loss of material in steel may be caused by corrosion. Carbon steel in an air environment is susceptible to corrosion except in the following situations: the steel is located in an air conditioned environment, or the steel is galvanized and not wetted. Stainless steel structural components are not subject to corrosion for the environments at Turkey Point. Accordingly, with the exceptions above, corrosion is an aging mechanism that can lead to loss of material in steel in air structural components.

Aggressive chemical attack due to boric acid is an aging mechanism for steel in air structural components. This form of corrosion is typically localized and is a result of leakage from borated water systems that can concentrate boric acid and lead to significant material loss of carbon steel components. Although this type of corrosion is event driven (boric acid leaks), boric acid corrosion was evaluated as an aging mechanism at Turkey Point.

Based on the above, loss of material due to corrosion and aggressive chemical attack is an aging effect requiring management for steel in air structural components.

CRACKING

Aging mechanisms that can lead to cracking of steel in air structural components are stress corrosion and fatigue.

Stress corrosion cracking is an age-related degradation mechanism that affects stainless steels but becomes significant only if tensile stresses and a corrosive environment exist. The stresses may be either applied (external) or residual (internal). The stress corrosion cracks themselves may be either transgranular or intergranular, depending upon the metal and the corrosive agent. As is normal in all cracking, the cracks are perpendicular to the tensile stress. Usually there is little or no obvious visual evidence of corrosion. Stress corrosion cracking is not an aging mechanism that can lead to cracking for steel in air structural components.

Fatigue is a progressive degradation problem for materials subjected to cyclic application of loads that are less than the maximum allowable static loads. For steel components, fatigue is the cumulative effect of microstructural localized plastic deformation in the material section that results each time a stress cycle of sufficient magnitude occurs. Fatigue is not an aging mechanism that can lead to cracking in steel in air structural components.

Based on the above, cracking is not an aging effect requiring management for steel in air structural components.

CHANGE IN MATERIAL PROPERTIES

Aging mechanisms that can cause change in material properties are thermal and irradiation embrittlement. Steel in air structural components outside the Containment are not exposed to the elevated temperatures or fluences that would cause reduction in fracture toughness. The only steel in air structural components inside Containment and within the scope of this subsection are those associated with fuel handling equipment. These structural components are also not subject to the elevated temperatures and fluences necessary for reduction in fracture toughness. Accordingly, change in material properties is not an aging effect requiring management for steel in air structural components.

3.6.2.1.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to steel in air structural components includes the following:

- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"
- NRC Information Notice 89-07, "Failures of Small-Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Render Emergency Diesels Inoperable"
- NRC Information Notice 89-80, "Potential For Water Hammer, Thermal Stratification, and Steam Binding in High Pressure Coolant Injection Piping"
- NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures"
- NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal"

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.6.2.1.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of steel in air structural component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.2.1.2.

3.6.2.1.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.6.2.1.2. Tables 3.6-3 through 3.6-20 contain the

results of the aging management review for the other structures and summarize the aging effects requiring management for steel in air structural components.

The aging effects requiring management are adequately managed by the following programs:

- ASME Section XI, Subsection IWF Inservice Inspection Program
- Boric Acid Wastage Surveillance Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the steel in air structural components listed in Tables 3.6-3 through 3.6-20 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.2.2 STEEL IN FLUID STRUCTURAL COMPONENTS

This subsection includes steel structural components that are exposed to fluids and those steel components that are exposed to both fluids and air. Steel structural components that are exposed to only an air environment were discussed in Subsection 3.6.2.1 above. Steel in fluid structural components include:

- refueling pool cavity liner plates
- spent fuel pool liner plates
- spent fuel handling equipment and tools
- spent fuel pool keyway gates
- fuel transfer tubes, penetration sleeves, and gate valves
- reactor cavity seal rings
- spent fuel storage racks and Boraflex
- spent fuel pool anchorages and embedments
- intake structure traveling screens

3.6.2.2.1 MATERIALS AND ENVIRONMENT

Steel in fluid structural components were designed and constructed in accordance with AISC standards. The codes and standards used for the design and fabrication

of the steel in fluid structural components are identified in Turkey Point UFSAR Chapter 5.

Steel in fluid structural components are constructed of painted carbon steel or stainless steel. In addition, the spent fuel storage racks contain Boraflex panels.

Turkey Point steel in fluid structural components are exposed to fluid environments of raw water – cooling canals and treated water – borated, and air environments of Containment air, indoor – not air conditioned, and outdoor (see Tables 3.0-1 and 3.0-2). The specific materials and environments for steel in fluid structural components for each structure are contained in Tables 3.6-3 through 3.6-20.

3.6.2.2.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of intended function(s) for steel in fluid structural components are loss of material, cracking, and change in material properties. The aging mechanisms that could lead to these aging effects in steel in fluid structural components were evaluated using the methodology provided in Appendix C. The results are provided below.

LOSS OF MATERIAL

Aging mechanisms that can lead to loss of material are leaching, aggressive chemical attack, mechanical wear, and corrosion.

Based on the evaluation using the methodology described in Appendix C, leaching, aggressive chemical attack, and mechanical wear are not aging mechanisms that can lead to loss of material in steel in fluid structural components.

Loss of material for steel in fluid structural components may be caused by corrosion. This may be seen as material dissolution, corrosion product buildup, and pitting. Loss of material may be uniform or localized.

Based on the above, loss of material due to corrosion is an aging effect requiring management for steel in fluid structural components.

CRACKING

Aging mechanisms that can lead to cracking of steel in fluid structural components are fatigue, hydrogen damage, and stress corrosion cracking.

Based on the evaluation using the methodology described in Appendix C, fatigue, hydrogen damage, and stress corrosion cracking were evaluated for steel in fluid structural components at Turkey Point and determined not to lead to cracking requiring management. Accordingly, cracking is not an aging effect requiring management for steel in fluid structural components.

CHANGE IN MATERIAL PROPERTIES

Aging mechanisms that can cause change in material properties are creep and stress relaxation, and thermal and irradiation embrittlement.

Based on the evaluation using the methodology described in Appendix C, creep and stress relaxation were evaluated for steel in fluid structural components at Turkey Point and determined not to lead to change in material properties.

Steel in fluid structural components outside Containment are not exposed to the elevated temperatures or fluences that would cause reduction in fracture toughness. The only steel in fluid structural components inside Containment and within the scope of this subsection are the reactor refueling cavity liner and those associated with fuel handling equipment. These structural components are also not subject to the elevated temperatures and fluences necessary for reduction in fracture toughness. Accordingly, change in material properties is not an aging effect requiring management for steel in fluid structural components.

Boraflex is a neutron absorber inserted between the fuel storage cells in high-density fuel storage racks. Irradiation results in embrittlement of the Boraflex inserts.

Based on the above, change in material properties due to irradiation of fuel storage rack Boraflex inserts is an aging effect requiring management.

3.6.2.2.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to steel in fluid structural components includes the following:

- NRC Bulletin 94-01, “Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1”

- NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants”
- NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”
- NRC Generic Letter 96-04, “Boraflex Degradation in Spent Fuel Pool Storage Racks”
- NRC Information Notice 87-43, “Gaps in Neutron-Absorbing Material in High Density Spent Fuel Storage Racks”
- NRC Information Notice 89-07, “Failures of Small-Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Render Emergency Diesels Inoperable”
- NRC Information Notice 89-80, “Potential For Water Hammer, Thermal Stratification, and Steam Binding in High Pressure Coolant Injection Piping”
- NRC Information Notice 93-70, “Degradation of Boraflex Neutron Absorber Coupons”
- NRC Information Notice 95-38, “Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks”
- NUREG-1522, “Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures”

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.6.2.2.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of steel in fluid structural component aging, in addition to interviews with responsible engineering personnel. No additional aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.2.2.2.

3.6.2.2.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond

those discussed in Subsection 3.6.2.2.2. Tables 3.6-3 through 3.6-20 contain the results of the aging management review for the other structures and summarize the aging effects requiring management for steel in fluid structural components.

The aging effects requiring management are adequately managed by the following programs:

- Boraflex Surveillance Program
- Chemistry Control Program
- Periodic Surveillance and Preventive Maintenance Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of the steel in fluid structural components listed in Tables 3.6-3 through 3.6-20 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.2.3 CONCRETE STRUCTURAL COMPONENTS

Concrete structural components include:

- foundations
- columns
- walls
- floors
- roofs
- equipment pads
- electric duct banks
- manholes
- trenches
- masonry block walls
- embedded steel
- embedded anchors
- concrete piping

3.6.2.3.1 MATERIALS AND ENVIRONMENT

Concrete structural components were designed and constructed in accordance with ACI and ASTM standards. The codes and standards used for the design and fabrication of the concrete structural components are identified in UFSAR Chapter 5.

Turkey Point concrete structural components are exposed to environments of outdoor, indoor – not air conditioned, indoor – air conditioned, buried, raw water – cooling canals, and embedded/encased (see Table 3.0-2). The specific materials and environments for concrete structural components for each structure are contained in Tables 3.6-3 through 3.6-20.

3.6.2.3.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of intended function(s) for concrete structural components are loss of material, cracking, and change in material properties. Each is discussed below.

LOSS OF MATERIAL

Loss of material is manifested in concrete structural components as scaling, spalling, pitting and erosion. Aging mechanisms that can lead to loss of material are freeze-thaw, abrasion and cavitation, elevated temperature, aggressive chemical attack, and corrosion of reinforcing and embedded/encased steel.

Freeze-thaw is considered an aging mechanism for concrete structural components that are exposed to severe weather conditions of numerous freeze-thaw cycles with significant amounts of winter rainfall. Turkey Point is located in a subtropical climate with long, warm summers accompanied by abundant rainfall and mild, dry winters with negligible freeze-thaw cycles. Therefore, freeze-thaw is not an aging mechanism that can lead to loss of material for concrete structural components.

Abrasion and cavitation is an aging mechanism that occurs only in concrete structures that are continually exposed to flowing water. The Intake Structure concrete components located below the intake canal water level and the intake cooling water piping are the only concrete components exposed to flowing water. The velocity in the intake canal and the velocity in the intake cooling water piping are significantly less than the threshold limits at which abrasion and cavitation degradation occurs. Therefore, abrasion and cavitation is not an aging mechanism that can lead to loss of material for concrete structural components.

Concrete structural components outside Containment are not exposed to elevated temperatures that exceed ACI threshold limits and, therefore, elevated temperature is not an aging mechanism that can lead to loss of material for concrete structural components.

Aggressive chemical attack, leading to corrosion of reinforcing steel and embedded steel, was identified as an age-related degradation mechanism for concrete structural components. At Turkey Point, this is applicable to concrete structural components exposed to the groundwater, saltwater flow, or saltwater splash (Intake Cooling Water System discharge). The structures with concrete structural components located below groundwater elevation are the Intake Structure, Discharge Structure, and the floors and lower wall portions of the residual heat removal pump and heat exchanger rooms in the Auxiliary Building. The concrete structural components at the Intake and Discharge Structures are also exposed to high chlorides due to the flow of saltwater. The Discharge Structure headwalls are exposed to discharge canal and Intake Cooling Water System discharge splash.

Based on the above, loss of material due to aggressive chemical attack and corrosion of reinforcing and embedded steel is an aging effect that requires aging management for concrete structural components below groundwater elevation, exposed to saltwater flow, or exposed to saltwater splash.

CRACKING

Cracking is manifested in concrete structural components as complete or incomplete separation of the concrete into two or more parts. Aging mechanisms that can lead to cracking are freeze-thaw, reactions with aggregates, fatigue, shrinkage, settlement, and elevated temperature.

As discussed previously, freeze-thaw is not an aging mechanism that can lead to cracking for concrete structural components.

Turkey Point concrete components were constructed using non-reactive aggregates whose acceptability was based on established industry standards and ASTM tests. Therefore, reaction with aggregates is not an aging mechanism that can lead to cracking for concrete structural components.

Fatigue is a progressive degradation problem for materials subjected to cyclic application of loads that are less than the maximum allowable static loads. Turkey Point concrete components are designed in accordance with ACI standards and

have good low-cycle fatigue properties. Although some concrete components are subject to high cycles of low-level repeated load, these components were designed in accordance with ACI standards, which limit the maximum design stress to less than 50% of the static stress of the concrete. The concrete fatigue strength is about 55% of its static strength at extremely high cycles (>10^7 cycles) of loading. Therefore, fatigue is not an aging mechanism that can lead to cracking for concrete structural components.

When concrete is exposed to air, large portions of the free water evaporate, causing shrinkage. At Turkey Point, the initial concrete water content was kept low, low slump concrete was used, and adequate steel reinforcement was provided, which all minimize shrinkage. Based on industry information, 100% of concrete shrinkage occurs within 20 years. Turkey Point concrete structures and concrete components were constructed more than 20 years ago; therefore, concrete shrinkage is not an aging mechanism that can lead to cracking for concrete structural components.

Settlement is based directly on the physical properties of a structure's foundation material. The most pronounced settlement is evidenced in the first several months after construction. Turkey Point concrete structures are founded on fossiliferous limestone bedrock (Miami Oolite) with crushed limestone fill. This foundation material is suitable for foundation systems with no significant structural settlement expected. Therefore, settlement is not an aging mechanism that can lead to cracking for concrete structural components.

Shrinkage and settlement of supporting structures can cause cracking of unreinforced masonry block walls. Cracking could reduce the structural strength of the wall. Any cracks that affected the structural integrity and could consequently impact the intended function(s) of the masonry block walls were identified in response to NRC Bulletin 80-11 and associated inspections.

Concrete structural components outside Containment are not exposed to elevated temperatures that exceed ACI threshold limits and, therefore, elevated temperature is not an aging mechanism that can lead to cracking for concrete structural components.

Based on the above, cracking due to shrinkage and settlement of unreinforced masonry block walls is an aging effect requiring management for concrete structural components.

CHANGE IN MATERIAL PROPERTIES

Change in material properties is manifested in concrete as increased permeability, increased porosity; reduction in pH; reduction in tensile, compressive, or bond strength; and reduction in modulus of elasticity. Aging mechanisms that can lead to a change in material properties are leaching, creep, elevated temperature, irradiation embrittlement, and aggressive chemical attack.

Leaching of calcium hydroxide is observed on concrete that is alternately wetted and dried. White deposits that are left on the surface of the concrete are a solution of water, free lime from the concrete, and carbon dioxide that is readily seen on the surface of the concrete. Turkey Point concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio that is characteristic of concrete having low permeability. This is consistent with the guidance provided by the ACI, and when implemented, degradation caused by leaching of calcium hydroxide is not significant. Therefore, leaching is not an aging mechanism that can lead to change in material properties for concrete structural components.

Creep is significant when new concrete is subjected to load and decreases exponentially with time; and any degradation is noticeable in the first few years. All reinforced concrete components were designed based on the ACI working stress design method. Creep in all concrete components is minimal because of low compressive stresses in concrete and the use of high strength concrete. In addition, creep proceeds at a decreasing rate with age, 96% of creep has occurred within 30 years; and therefore, concrete creep is not an aging mechanism that can lead to change in material properties for concrete structural components.

Concrete structural components outside Containment are not exposed to elevated temperature or fluences that would cause embrittlement.

Concrete structural components subject to loss of material due to aggressive chemical attack would also be subject to change in material properties due to the same aging mechanism.

Based on the above, change in material properties due to aggressive chemical attack is an aging effect requiring management for concrete structural components below groundwater elevation, exposed to saltwater flow, or exposed to saltwater splash.

3.6.2.3.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to concrete structural components includes the following:

- NRC Bulletin 80-11, “Masonry Wall Design”
- NRC Information Notice 97-11, “Cement Erosion from Containment Subfoundations at Nuclear Power Plants”
- NRC Information Notice 98-26, “Settlement Monitoring and Inspection of Plant Structures Affected by Degradation of Porous Concrete Subfoundations”
- NUREG-1522, “Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures”
- NUREG/CR-4652, “Concrete Component Aging and its Significance Relative to Life Extension of Nuclear Power Plants”
- NUREG/CP-0100, Prasad, N., et al., “Concrete Degradation Monitoring and Evaluation”, Proceedings of the International Nuclear Power Plant Aging Symposium, August 30 – September 1, 1998

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.6.2.3.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of concrete structural component aging, in addition to interviews with responsible engineering personnel. No additional aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.2.3.2.

3.6.2.3.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.6.2.3.2. Tables 3.6-3 through 3.6-20 contain the results of the aging management review for the other structures and summarize the aging effects requiring management for concrete structural components.

The aging effects requiring management are adequately managed by the following program:

- Systems and Structures Monitoring Program

Based on the evaluation provided in Appendix B for the program above, aging effects are adequately managed so that the intended functions of the concrete structural components listed in Tables 3.6-3 through 3.6-20 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.2.4 MISCELLANEOUS STRUCTURAL COMPONENTS

Miscellaneous structural components include:

- fire rated assemblies (fire penetration seals, fire retardant coatings, and fire doors)
- Cooling Water Canals
- weatherproofing (structures and sealants)
- flood protection seals and stop logs
- Control Room ceiling and raised floor

3.6.2.4.1 MATERIALS AND ENVIRONMENT

The miscellaneous structural components consist of a variety of materials, depending on their location and function. Materials used include painted and galvanized carbon steel, aluminum, earth/rock, wood, gypsum board, acoustical panels, weatherproofing materials (silicone caulking, sealants and foams, neoprene gaskets, asphalt, felt, and membrane roofing), and fire protection materials (silicone foams, elastomers and gels, mineral wool, alumina-silica, ceramic fiber blankets, ceramic fiber boards, calcium silicate board, flexible boots, Thermo-Lag, Flamemastic, and cementitious coatings).

The miscellaneous structural components are exposed to different environments, depending on their location and function. Environments include outdoor, indoor – not air conditioned, and indoor – air conditioned (see Table 3.0-2). The specific materials and environments for miscellaneous structural components for each structure are contained in Tables 3.6-3 through 3.6-20.

3.6.2.4.2 AGING EFFECTS REQUIRING MANAGEMENT

The aging effects that could cause loss of intended function(s) for miscellaneous structural components are loss of material and loss of seal. Both are discussed below.

LOSS OF MATERIAL

Aging mechanisms that can lead to loss of material are wear, weathering, corrosion, organic decomposition.

Wear, weathering, organic decomposition, and corrosion were evaluated for loss of material of the fire rated assemblies, cooling water canals, weatherproofing (structures and sealants), Control Room ceiling and raised floor, and the Fire Protection Monitoring Station miscellaneous structural components. Except for some fire retardant coatings and fire doors, wear, weathering, organic decomposition, and corrosion do not lead to loss of material for miscellaneous structural components.

Fire rated assemblies, except for some fire retardant coatings and fire doors, when cured, become monolithic solids taking the form of the system to which each is injected or applied. In addition, SECY 96-146 concludes that penetration seals are not subjected to aging effects.

Raceway material fire retardant coatings, installed as part of the 10 CFR 50 Appendix R upgrades, were provided by Thermo-Lag. Thermo-Lag locations exposed to an outdoor environment have experienced topcoat damage and, therefore, loss of material due to weathering is an aging effect requiring management for Thermo-Lag insulation materials.

Fire doors are passive features to seal passageways through fire barriers and are constructed of carbon steel. Carbon steel in outdoor and indoor – not air conditioned environments is susceptible to corrosion. Therefore, loss of material due to corrosion is an aging effect requiring management for certain fire doors.

The Cooling Water Canals provide a source of cooling water for plant shutdown. The heat load for shutdown is a small percentage of the normal operating heat load. Weathering and organic decomposition are not aging mechanisms that could lead to loss of material of the Cooling Water Canals that could cause loss of intended function(s).

The Control Room ceiling and raised floor and the Fire Protection Monitoring Station are indoor – air conditioned environments that are occupied 24 hours per day. Should indication of an aging effect arise in these areas it would be identified and corrected.

The miscellaneous structural components that provide flood protection include wooden and aluminum stop logs, and the pipe trench penetration seals. Aluminum is highly resistant to corrosion. Aluminum stop logs and pipe trench penetrations have been evaluated for loss of material and determined not to require aging management. Wooden stop logs are subject to loss of material due to organic decomposition and, therefore, loss of material due to organic decomposition is an aging effect requiring management for wooden stop logs.

Based on the above, loss of material due to weathering, corrosion, and organic decomposition is an aging effect requiring management for miscellaneous structural components.

LOSS OF SEAL

The aging mechanism that can lead to loss of seal is weathering.

The covers of manholes containing redundant safe shutdown cables are sealed to prevent the spread of flammable or combustible liquids into manholes. Sealant material is chemically resistant to combustible liquid and does not burn readily. Sealant material is subject to loss of seal when exposed to weathering and, therefore, loss of seal due to weathering is an aging effect requiring management for manholes and associated sealants.

Weatherproofing includes both roofing systems and structural sealants. These components protect internal components from the effects of exposure to weather, primarily rainwater. Inspections are performed to ensure weatherproofing is functioning properly. Weatherproofing is exposed to environmental degradation that may potentially damage the sealant material. Loss of seal due to weathering is an aging effect requiring management for weatherproofing components.

The miscellaneous structural components that provide flood protection include wooden and aluminum stop logs, and the pipe trench penetration seals. Wooden and aluminum stop logs have been evaluated for loss of seal and determined not to require aging management. The penetration seals in the pipe trench are subject to loss of seal from exposure to weather and exhibit signs of decreased elasticity (drying out) and an increase in hardness and, thus, loss of seal.

Based on the above, loss of seal due to weathering is an aging effect requiring management for miscellaneous structural components.

3.6.2.4.3 OPERATING EXPERIENCE

INDUSTRY EXPERIENCE

A review of industry operating history and a review of NRC generic communications were performed to validate the set of aging effects that require management. The industry correspondence that was reviewed for operating experience related to miscellaneous structural components includes the following:

- NRC Bulletin 92-01, “Failure Of Thermo-Lag 330 Fire Barrier System To Maintain Cabling In Wide Cable Trays And Small Conduits Free From Fire Damage”
- NRC Bulletin 92-01, Supplement 1, “Failure of Thermo-Lag 330 Fire Barrier System to Perform Its Specified Fire Endurance Function“
- NRC Generic Letter 92-08, “Thermo-Lag 330-1 Fire Barriers”
- NRC Information Notice 88-04, “Inadequate Qualification and Documentation of Fire Barrier Penetration Seals”
- NRC Information Notice 88-56, “Potential Problems with Silicone Foam Fire Barrier Penetration Seals”
- NRC Information Notice 91-47, “Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test”
- NRC Information Notice 91-79, “Deficiencies in the Procedures for Installing Thermo-Lag Fire Barrier Materials”
- NRC Information Notice 91-79, Supplement 1, “Deficiencies Found in Thermo-Lag Fire Barrier Installation”

- NRC Information Notice 92-46, “Thermo-Lag Fire Barrier Material Special Review Team Final Report Findings, Current Fire Endurance Tests, And Ampacity”
- NRC Information Notice 92-55, “Current Fire Endurance Test Results For Thermo-Lag Fire Barrier Material”
- NRC Information Notice 92-82, “Results of Thermo-Lag 330-1 Combustibility Testing”
- NRC Information Notice 94-22, “Fire Endurance and Ampacity Derating Test Results for 3-hour Fire-Rated Thermo-Lag 330-1 Fire Barriers”
- NRC Information Notice 94-28, “Potential Problems With Fire-Barrier Penetration Seals”
- NRC Information Notice 94-34, “Thermo-Lag 330-660 Flexi-Blanket Ampacity Derating Concerns
- NRC Information Notice 95-32, “Thermo-Lag 330-1 Flame Spread Test Results”
- NRC Information Notice 95-49 and Supplement 1, “Seismic Adequacy of Thermo-Lag Panels”
- NRC Information Notice 97-70, “Potential Problems with Fire Barrier Penetration Seals”
- SECY-96-146, “Technical Assessment of Fire Barrier Penetration Seals in Nuclear Power Plants”

No aging effects requiring management were identified from the above documents beyond those already identified in Subsection 3.6.2.4.2.

PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was also reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of miscellaneous structural component aging, in addition to interviews with responsible engineering personnel. No aging effects requiring management were identified from this review beyond those identified in Subsection 3.6.2.4.2.

3.6.2.4.4 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.6.2.4.2. Tables 3.6-3 through 3.6-20 contain the results of the aging management review for the other structures and summarize the aging effects requiring management for miscellaneous structural components.

The aging effects requiring management are adequately managed by the following programs:

- Fire Protection Program
- Periodic Surveillance and Preventive Maintenance Program
- Systems and Structures Monitoring Program

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions for the miscellaneous structural components listed in Tables 3.6-3 through 3.6-20 are maintained consistent with the current licensing basis for the period of extended operation.

3.6.3 REFERENCES

3.6-1 Final Environmental Statement Related to Operation of Turkey Point Plant, Florida Power and Light Company, Dockets No. 50-250 and 50-251, July 1972, United States Atomic Energy Commission Directorate of Engineering.

3.6-2 WCAP-14422, “License Renewal Evaluation: Aging Management for Reactor Coolant System Supports,” Revision 2, March 1997.

3.6-3 C. I. Grimes (NRC) letter to R. A. Newton (WOG), “Draft Safety Evaluation Concerning the Westinghouse Owners Group License Renewal Evaluation: Aging Management for Reactor Coolant System Supports, WCAP-14422, Revision 2,” February 25, 2000.

TABLE 3.6-1
STRUCTURAL COMPONENT INTENDED FUNCTIONS

1. Provide pressure boundary and/or fission product barrier.
2. Provide structural support to safety-related components.
3. Provide shelter/protection to safety-related components (including radiation shielding).
4. Provide rated fire barrier to retard spreading of a fire.
5. Provide a source of cooling water for plant shutdown.
6. Provide missile barrier.
7. Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment o safety-related functions.
8. Provide flood protection barrier.
9. Provide filtration of process fluid to protect downstream equipment.
10. Provide structural support and/or shelter to components required for fire protection, anticipated transients without scrar station blackout (SBO) events.

    NOTE: Although not credited in the analyses for these events, these components have been conservatively included ir renewal.
11. Provide pipe whip restraint and/or jet impingement protection.

TABLE 3.6-2
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete Walls above groundwater elevation (dome and cylinder walls)	2, 3, 4, 6, 7, 8, 10	Concrete	Buried Outdoor Indoor – not air conditioned	None
Reinforced concrete Walls below groundwater elevation (cylinder walls and foundation mat)	2, 3, 7, 10	Concrete	Buried	Loss of material Change in material properties
Anchorages/ embedments above groundwater elevation	1, 2, 7, 10	Carbon steel	Embedded/Encased	None
Anchorages/ embedments below groundwater elevation	1, 2, 7, 10	Carbon steel	Embedded/Encased	Loss of material
Internal reinforced concrete components: Beams Floor slabs Shield walls Secondary compartment walls Refueling cavity walls Equipment pads Missile shields Curbs Containment sumps Miscellaneous	2, 3, 6, 7, 8, 10	Concrete	Containment air	None

TABLE 3.6-2 (continued)
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Carbon steel liner plates	1, 2, 7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Liner plate anchorages/ attachments exposed surfaces	1, 2, 7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Liner plate anchorages/ attachments	1, 2, 7, 10	Carbon steel	Embedded/Encased	None
Mechanical piping penetrations	1, 2, 4	Carbon steel	Containment air Outdoor Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material

TABLE 3.6-2 (continued)
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Mechanical ventilation penetrations	1, 2, 4	Carbon steel	Containment air Outdoor	Loss of material
			Borated water leaks	Loss of material
Electrical penetrations	1, 2, 4	Carbon steel	Containment air Indoor – not air conditioned	Loss of material
			Borated water leaks	Loss of material
Containment personnel hatches Emergency escape hatches Equipment hatches	1, 4	Carbon steel	Containment air Outdoor	Loss of material
			Borated water leaks	Loss of material
Seals and gaskets (hatches)	1	Elastomers	Containment air Outdoor	Change in material properties
Moisture barriers	3	Elastomers	Containment air	Change in material properties

TABLE 3.6-2 (continued)
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Fuel transfer tube blind flanges [1]	1, 4	Stainless steel	Containment air	None
Cable trays Conduits	2, 7, 10	Carbon steel – galvanized	Containment air	None
			Borated water leaks	Loss of material
Conduit and cable tray supports	2, 7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Electrical, instrument panels and enclosures	2, 3, 7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
		Carbon steel – galvanized	Containment air	None
			Borated water leaks	Loss of material
		Stainless steel	Containment air	None
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material

NOTE: 1. The fuel transfer tube, penetration sleeves, and gate valves are addressed in Table 3.6-16.

TABLE 3.6-2 (continued)
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Instrument line supports	2, 7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Structural steel: Beams Columns Elevators Stairs Platforms Grating	2, 7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Sump screens	9	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Safety-related piping and component supports	2, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material

TABLE 3.6-2 (continued)
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Non-safety related piping and component supports	7, 10	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Pipe whip restraints	11	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Non-safety related pipe segments between class break and seismic anchor	2, 7, 10	Stainless steel	Containment air	None
		Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Radiant energy shields	4	Stainless steel	Containment air	None
Miscellaneous structural components	2, 7, 10	Carbon steel - galvanized	Containment air	None
			Borated water leaks	Loss of material
Tendon wires	2	Carbon steel	In grease (sheathing filler material)	Loss of material

TABLE 3.6-2 (continued)
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Tendon anchorage components	2	Carbon steel	In grease (sheathing filler material)	Loss of material
Polar cranes: [1] Runway rail brackets Runway rails Main girders Cabs Footwalks & railings End connectors (fasteners) Electrical enclosures Trolley rails Trolley structures Control conductor supports	2, 7	Carbon steel	Containment air	Loss of material
Reactor vessel supports	2	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
		Stainless steel	Containment air	None

NOTE: 1. Not exposed to potential borated water leakage.

TABLE 3.6-2 (continued)
CONTAINMENTS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Steam generator supports	2	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
		Lubrite	Containment air	None
Pressurizer supports	2	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Reactor coolant supports	2	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Surge line supports	2	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material

TABLE 3.6-3
AUXILIARY BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Structural steel: Beams Columns Connections	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
			Borated water leaks	Loss of material
Miscellaneous steel: Stairs Platforms Grating	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Indoor – air conditioned	None
		Carbon steel Carbon steel – galvanized	Borated water leaks	Loss of material

TABLE 3.6-3 (continued)
AUXILIARY BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Borated water leaks	Loss of material
			Indoor – air conditioned	None
Safety-related supports (pipe supports and component supports)	2, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Borated water leaks	Loss of material
			Indoor – air conditioned	None
Non-safety related supports	7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Borated water leaks	Loss of material
			Indoor – air conditioned	None

TABLE 3.6-3 (continued)
AUXILIARY BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Non-safety related pipe segments between class break and seismic anchor	2, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Borated water leaks	Loss of material
		Stainless steel	Indoor - not air conditioned	None
Cable trays Conduits	2, 3, 7, 10	Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None
			Borated water leaks	Loss of material
Cable tray and conduit supports	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor –air conditioned	None
		Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None
		Carbon steel Carbon steel – galvanized	Borated water leaks	Loss of material

TABLE 3.6-3 (continued)
AUXILIARY BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Electrical enclosures	2, 3, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized Stainless steel	Indoor - not air conditioned Indoor – air conditioned	None
		Carbon steel Carbon steel – galvanized	Borated water leaks	Loss of material
Electrical component supports	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None
		Carbon steel Carbon steel – galvanized	Borated water leaks	Loss of material

TABLE 3.6-3 (continued)
AUXILIARY BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Instrument racks and frames	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None
		Carbon steel Carbon steel – galvanized	Borated water leaks	Loss of material
HVAC duct supports	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None
		Carbon steel Carbon steel – galvanized	Borated water leaks	Loss of material
Fan/filter intake hoods	3	Stainless steel	Outdoor	None

TABLE 3.6-3 (continued)
AUXILIARY BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Pipe whip restraints	11	Carbon steel	Indoor - not air conditioned	Loss of material
			Borated water leaks	Loss of material
Reinforced concrete: Foundations (above groundwater elevation)	2, 7,10	Concrete	Buried	None
Reinforced concrete: Foundations and walls (residual heat removal pump and heat exchanger rooms below groundwater elevation)	2, 7, 10	Concrete	Buried	Loss of material Change in material properties
Reinforced concrete: Beams Columns Walls Floors/slabs (above groundwater elevation)	2, 3, 4, 6, 7, 8, 10	Concrete	Indoor - not air conditioned Indoor – air conditioned Outdoor	None
Reinforced block walls	2, 3, 4, 10	Concrete	Indoor - not air conditioned	None

TABLE 3.6-3 (continued)
AUXILIARY BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Unreinforced block walls	2, 3, 4, 10	Concrete	Indoor - not air conditioned	Cracking
Anchorages/ embedments (below groundwater elevation)	2, 7, 10	Steel	Embedded/Encased	Loss of material
Anchorages/ embedments (above groundwater elevation)	2, 7, 10	Steel	Embedded/Encased	None
Weatherproofing	3	Caulking/ sealant Waterproofing membrane with concrete topping Asphalt roll roofing Flashing Roofing felt	Outdoor	Loss of seal
Pipe trench penetrations	8	Promatec flexible seal	Outdoor	Loss of seal
Stop logs	8	Wood	Outdoor	Loss of material
		Aluminum	Outdoor	None

TABLE 3.6-4
COLD CHEMISTRY LAB

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete: Foundations (above groundwater elevation)	7	Concrete	Buried	None
Reinforced concrete: Walls and roof	7	Concrete	Indoor - air conditioned Outdoor	None

TABLE 3.6-5
CONTROL BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management	F
Structural steel: Beams Columns Connections	2, 7, 10	Carbon steel	Indoor – air conditioned	None	N
Miscellaneous steel	2, 3, 6, 7, 10	Carbon steel	Indoor – air conditioned	None	N
			Outdoor	Loss of material	S S M F
Stairs Platforms Grating	7	Carbon steel Carbon steel – galvanized	Indoor – air conditioned	None	N
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Indoor – air conditioned	None	N
Safety-related supports (pipe supports and component supports)	2, 10	Carbon steel	Indoor – air conditioned	None	N
Non-safety related supports	7, 10	Carbon steel	Indoor – air conditioned	None	N
Cable trays Conduits	2, 3, 7, 10	Carbon steel – galvanized	Indoor – air conditioned	None	N

TABLE 3.6-5 (continued)
CONTROL BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Cable tray and conduit supports	2, 7, 10	Carbon steel Carbon steel – galvanized	Indoor – air conditioned	None
Electrical enclosures (includes control panels)	2, 3, 7, 10	Carbon steel Carbon steel – galvanized Stainless steel	Indoor – air conditioned	None
Electrical component supports	2, 7, 10	Carbon steel Carbon steel – galvanized	Indoor – air conditioned	None
Instrument racks and frames	2, 7, 10	Carbon steel Carbon steel – galvanized	Indoor – air conditioned	None
HVAC supports	2, 7, 10	Carbon steel Carbon steel – galvanized	Indoor – air conditioned	None
Reinforced concrete: Foundations (above groundwater elevation)	2, 7, 10	Concrete	Buried	None

TABLE 3.6-5 (continued)
CONTROL BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete: Beams Columns Walls Floors/Slabs (above groundwater elevation)	2, 3, 4, 6, 7, 10	Concrete	Outdoor Indoor – air conditioned	None
Reinforced masonry block walls	2, 3, 4, 6, 7, 10	Concrete	Indoor – air conditioned	None
Unreinforced masonry block walls	2, 3, 4, 6, 7, 10	Concrete	Indoor – air conditioned	Cracking
Anchorages/ embedments	2, 7, 10	Steel	Embedded/Encased	None
Control Room ceiling	7	Gypsum board Acoustical panels	Indoor – air conditioned	None
Control Room raised floor	7	Tee cor panel Micarta Cove base Steel supports	Indoor – air conditioned	None
Weatherproofing	3	Caulking/sealant Roofing material	Outdoor	Loss of seal

TABLE 3.6-6
COOLING WATER CANALS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Earthen canal	5	Earth/rock	Outdoor	None

TABLE 3.6-7
DIESEL DRIVEN FIRE PUMP ENCLOSURE

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Anchorages/ embedments exposed surfaces	10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None
Reinforced concrete foundations (above groundwater elevation)	10	Concrete	Buried	None
Anchorages/ embedments (above groundwater elevation)	10	Carbon steel Carbon steel – galvanized	Embedded/Encased	None
Pipe supports	10	Carbon steel	Indoor – not air conditioned	Loss of material
Manufactured structure	10	Steel Aluminum	Outdoor	Loss of seal
Doors	10	Aluminum	Outdoor	None
Louvers	10	Aluminum	Outdoor	None

TABLE 3.6-8
DISCHARGE STRUCTURE

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete: North pipe headwall	7	Concrete	Raw water – cooling canals (submerged or exposed to splash)	Loss of material Change in material properties
			Outdoor	None
Reinforced concrete: South pipe headwall	7	Concrete	Raw water – cooling canals (submerged or exposed to splash)	Loss of material Change in material properties
			Outdoor	None

TABLE 3.6-9
ELECTRICAL PENETRATION ROOMS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Anchorages/ embedments (above groundwater elevation)	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None
Cable trays Conduits	2, 3, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None
Cable trays and conduits supports	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None
Electrical enclosures	2, 3, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None
Electrical component supports	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None

TABLE 3.6-9 (continued)
ELECTRICAL PENETRATION ROOMS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Instrument racks	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
Structural steel	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
Ladders Platforms	7	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None
Reinforced concrete: Foundations (above groundwater elevation)	2, 7, 10	Concrete	Outdoor Buried	None
Reinforced concrete: (above groundwater elevation) Walls Floors Roof	2, 3, 4, 6, 7, 10	Concrete	Indoor - not air conditioned Outdoor	None
Anchorages/ embedments (above groundwater elevation)	2, 7, 10	Steel	Embedded/Encased	None

TABLE 3.6-9 (continued)
ELECTRICAL PENETRATION ROOMS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Weatherproofing	3	Caulking/sealant Membrane roof with concrete topping Flashing	Outdoor	Loss of seal

TABLE 3.6-10
EMERGENCY DIESEL GENERATOR BUILDINGS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Structural steel: Beams Columns Connections	2, 6, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
Miscellaneous steel	2, 3, 6, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
Stairs Platforms Grating	7	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Indoor – air conditioned Indoor - not air conditioned	None
		Stainless steel	Indoor - not air conditioned	None

TABLE 3.6-10 (continued)
EMERGENCY DIESEL GENERATOR BUILDINGS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Indoor – air conditioned Indoor - not air conditioned	None
Safety-related supports (pipe supports and component supports)	2, 10	Carbon steel	Indoor - not air conditioned	Loss of material
Non-safety related pipe supports	7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
Non-safety related pipe segments between class break and seismic anchor	2, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor - not air conditioned	None
Cable trays Conduits	2, 3, 7, 10	Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None

TABLE 3.6-10 (continued)
EMERGENCY DIESEL GENERATOR BUILDINGS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Cable tray and conduit supports	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel – galvanized	Indoor – air conditioned	None
Electrical component supports	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None
Electrical enclosures	2, 3, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
		Carbon steel	Indoor – air conditioned	None
		Carbon steel – galvanized Stainless steel	Indoor - not air conditioned Indoor – air conditioned	None

TABLE 3.6-10 (continued)
EMERGENCY DIESEL GENERATOR BUILDINGS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Instrument racks and frames	2, 7, 10	Carbon steel	Indoor - not air conditioned	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized Stainless steel	Indoor - not air conditioned Indoor – air conditioned	None
HVAC supports	2, 7, 10	Carbon steel	Indoor - not air conditioned Outdoor	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Outdoor	None
HVAC roof hoods (Unit 4)	2, 7, 10	Stainless steel	Outdoor	None
Reinforced concrete: Foundations (above groundwater elevation)	2, 7, 10	Concrete	Buried	None

TABLE 3.6-10 (continued)
EMERGENCY DIESEL GENERATOR BUILDINGS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete: (above groundwater elevation) Beams Columns Walls Floor/slabs	2, 3, 4, 6, 7, 8, 10	Concrete	Indoor - not air conditioned Outdoor	None
Reinforced masonry block walls	2, 3, 4, 6, 7, 8, 10	Concrete	Indoor - not air conditioned Outdoor	None
Unreinforced masonry block walls	2, 3, 4, 6, 7, 8, 10	Concrete	Indoor - not air conditioned	Cracking
Anchorages/ embedments (above groundwater elevation)	2, 7, 10	Steel	Embedded/Encased	None
Weatherproofing	3	Chase foam Silicone Caulking	Outdoor	Loss of seal
Louvers	3	Aluminum	Outdoor	None

TABLE 3.6-11
FIRE PROTECTION MONITORING STATION

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Structural steel roof support	10	Carbon steel	Indoor - air conditioned	None
Anchorages/ embedments exposed surfaces	10	Carbon steel Carbon steel - galvanized	Indoor - air conditioned	None
Reinforced concrete floor and roof	10	Concrete	Outdoor Indoor – air conditioned	None
Unreinforced masonry block walls	10	Concrete	Outdoor Indoor - air conditioned	Cracking
Anchorages/ embedments (above groundwater elevation)	10	Carbon steel	Embedded/Encased	None
Doors	10	Aluminum	Outdoor Indoor - air conditioned	None
Membrane roof	10	W.R. Grace membrane roofing system	Outdoor	None

TABLE 3.6-12 [1]
FIRE RATED ASSEMBLIES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Fire Barriers				
Raceway fireproofing protection	4	Thermo-Lag 3M Interam FireDam Caulk Marinite board Cerafiber Quelpyre mastic Silicone foam Flamemastic	Outdoor	Loss of material
			Indoor - not air conditioned Indoor – air conditioned	None
Fire retardant coating	4	Flamemastic	Indoor - not air conditioned Indoor – air conditioned	None
Structural steel fireproofing	4	Cementitious fireproofing	Indoor - not air conditioned Indoor – air conditioned	None
Manhole seals	4	Sealant	Outdoor	Loss of seal

NOTE: 1. Concrete and steel structural components that serve as fire rated barriers are addressed with each structure. R
are addressed with the Containments.

TABLE 3.6-12 (continued)
FIRE RATED ASSEMBLIES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Fire Barriers (continued)				
Fire sealed isolation joint	4	Cerafiber	Outdoor	Loss of seal
Fire doors	4	Carbon steel	Indoor - air conditioned	None
			Outdoor Indoor – not air conditioned	Loss of material
Control Room fire doors	3, 4	Carbon steel	Indoor - air conditioned	None
			Outdoor	Loss of material

TABLE 3.6-12 (continued)
FIRE RATED ASSEMBLIES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Fire Seals				
Type M-1 mechanical penetration seals	4	Silicone elastomer	Indoor - not air conditioned Indoor – air conditioned	None
Type M-4 sheet metal sleeve extensions	4	Sheet metal sleeve		
Type M-8 mechanical penetration seals	4	Silicone foam		
Type M-11 penetration requiring movement	4	Silicone elastomer Sheet metal sleeve		
Type M-15 boot and fiber	4	Cerafiber and boot Sheet metal sleeve		
Type E-1 electrical penetration seal	4	Silicone elastomer	Indoor - not air conditioned Indoor – air conditioned	None
Type E-16 wireway through an I-beam web	4	Silicone elastomer		
Type EM-1 boxing detail	4	Carbon steel – galvanized		
Type M-18 instrument tubing penetration seal	4	Hydrosil		
Type M-3 mechanical penetration seal	4	High density gel		

TABLE 3.6-12 (continued)
FIRE RATED ASSEMBLIES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Fire Seals (continued)				
PVC conduit	4	Silicone foam	Indoor - not air conditioned Indoor – air conditioned	None
Type GFS penetration seal	4	OZ Gedney fireseal		
Type CF penetration seal	4	Ceramic fiber		
Type CLK penetration seal	4	Cerafiber with caulk		
Type foam penetration seal	4	Silicone foam		
Open ended conduit (empty) Threaded metal cap	4	Carbon steel – galvanized	Indoor - not air conditioned Indoor – air conditioned	None
Electrical conduit seal	4	Silicone RTV foam	Indoor – not air conditioned Indoor – air conditioned	None

TABLE 3.6-13
INTAKE STRUCTURE

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Structural steel: Beams Columns Connections	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Stairs Platforms Grating	7	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized Stainless steel	Outdoor	None
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor - wetted	
		Carbon steel – galvanized	Outdoor	None
Non-safety related supports	7, 10	Carbon steel	Outdoor	Loss of material
Non-safety related pipe segments between class break and seismic anchor	2, 10	Stainless steel	Outdoor	None

TABLE 3.6-13 (continued)
INTAKE STRUCTURE

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Cable trays Conduits	2, 3, 7, 10	Carbon steel – galvanized	Outdoor	None
			Outdoor – wetted	Loss of material
Cable tray and conduit supports	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor – wetted	
		Carbon steel – galvanized	Outdoor	None
Electrical enclosures	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor – wetted	
		Carbon steel – galvanized Stainless steel	Outdoor	None
Electrical component supports	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor - wetted	
		Carbon steel – galvanized	Outdoor	None

TABLE 3.6-13 (continued)
INTAKE STRUCTURE

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Instrument racks and frames	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor - wetted	
		Carbon steel – galvanized	Outdoor	None
Intake bridge crane	7	Carbon steel	Outdoor	Loss of material
Intake structure traveling screen frames	7	Carbon steel	Outdoor Raw water – cooling canals	Loss of material
Intake structure traveling screen cloth	9	Stainless steel	Outdoor Raw water – cooling canals	Loss of material
Reinforced concrete: Foundations Beams Columns Walls Floors/slabs (below intake canal level)	2, 5, 7, 10	Concrete	Raw water – cooling canals (submerged)	Loss of material Change in material properties

TABLE 3.6-13 (continued)
INTAKE STRUCTURE

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete: Beams Columns Walls Floors/slabs (above intake canal level)	2, 5, 7, 8, 10	Concrete	Outdoor	None
Anchorages/ embedments (below intake canal level)	2, 7, 10	Steel	Embedded/Encased	Loss of material
Anchorages/ embedments (above intake canal level)	2, 7, 8, 10	Steel	Embedded/Encased	None

TABLE 3.6-14
MAIN STEAM AND FEEDWATER PLATFORMS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Structural steel: Beams Columns Connections	2, 6, 7, 10	Carbon steel	Outdoor	Loss of material
Stairs Platforms Grating	7	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Safety-related supports (pipe and component supports)	2, 10	Carbon steel	Outdoor	Loss of material
Non-safety related supports	7, 10	Carbon steel	Outdoor	Loss of material
Pipe whip restraints	11	Carbon steel	Outdoor	Loss of material
Cable trays Conduits	2, 3, 7, 10	Carbon steel – galvanized	Outdoor	None

TABLE 3.6-14 (continued)
MAIN STEAM AND FEEDWATER PLATFORMS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Cable tray and conduit supports	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Electrical enclosures	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized Stainless steel	Outdoor	None
Electrical component supports	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Instrument racks and frames	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Reinforced concrete foundations (above groundwater elevation)	2, 7, 10	Concrete	Outdoor Buried	None

TABLE 3.6-14 (continued)
MAIN STEAM AND FEEDWATER PLATFORMS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete: (above groundwater elevation) Walls Floors Roof	2, 3, 6, 7, 10	Concrete	Outdoor	None
Anchorages/ embedments (above groundwater elevation)	2, 7, 10	Steel	Embedded/Encased	None

TABLE 3.6-15
PLANT VENT STACK

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Structural steel supports/ restraints	7	Carbon steel	Outdoor	Loss of material
Anchorages/ embedments exposed surfaces	7	Carbon steel	Outdoor	Loss of material
		Carbon steel - galvanized	Outdoor	None
Conduits and conduit supports	7	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Electrical enclosures and supports	7	Carbon steel – galvanized Stainless steel	Outdoor	None
Steel vent stack	7	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None

TABLE 3.6-15 (continued)
PLANT VENT STACK

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Pedestal and grout cover (above groundwater elevation)	7	Concrete	Outdoor Buried	None
Anchorages/embedments (above groundwater elevation)	7	Carbon steel	Embedded/Encased	None

TABLE 3.6-16
SPENT FUEL STORAGE AND HANDLING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Manipulator cranes	2	Carbon steel	Containment air	Loss of material
			Borated water leaks	Loss of material
Spent fuel bridge cranes	2	Carbon steel	Indoor - not air conditioned	Loss of material
Spent fuel cask crane	2	Carbon steel	Outdoor	Loss of material
Fuel transfer sheave frames	2	Carbon steel	Indoor - not air conditioned Containment air	Loss of material
			Borated water leaks	Loss of material
Spent fuel pools, transfer canals, and refueling pool liners keyway gates embedded plates	1	Stainless steel	Containment air Indoor - not air conditioned	None
			Treated water – borated	Loss of material

TABLE 3.6-16 (continued)
SPENT FUEL STORAGE AND HANDLING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Fuel transfer tubes Fuel transfer tube penetration sleeves Fuel transfer tube gate valves	1[1]	Stainless steel	Containment air Indoor - not air conditioned (internal and external surfaces)	None
			Treated water – borated (internal and external surfaces)	Loss of material
		Carbon steel	Embedded/Encased	None
Spent fuel handling equipment and tools	2	Stainless steel	Containment air	None
			Treated water – borated	Loss of material
Reactor cavity seal rings	1	Stainless steel	Containment air	None
			Treated water – borated	Loss of material
Spent fuel storage racks	2	Stainless steel	Treated water – borated	Loss of material

NOTE: 1. The pressure boundary function of the portion of the fuel transfer tubes inside Containment and the penetration
Containment integrity. The pressure boundary function of the portion of the fuel transfer tubes outside Containr
transfer tube gate valves is spent fuel pool integrity when the spent fuel pool keyway gates are removed during

TABLE 3.6-16 (continued)
SPENT FUEL STORAGE AND HANDLING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Boraflex	3	Boron impregnated polymer	Treated water – borated	Change in material properties
Reinforced concrete overhead sliding doors	3, 4, 6	Concrete	Outdoor	None

TABLE 3.6-17
TURBINE BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Outdoor	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Outdoor Indoor – air conditioned	None
Structural steel and miscellaneous steel (including stairs, platforms and grating)	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Non-safety related supports	7, 10	Carbon steel	Outdoor	Loss of material
Non-safety related pipe segments between class break and seismic anchor	2, 10	Carbon steel	Outdoor	Loss of material
Cable trays Conduits	2, 3, 7, 10	Carbon steel – galvanized	Outdoor Indoor – air conditioned	None

TABLE 3.6-17 (continued)
TURBINE BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Cable tray and conduit supports	2, 7, 10	Carbon steel	Outdoor	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Outdoor Indoor – air conditioned	None
Electrical enclosures	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Outdoor Indoor – air conditioned	None
		Stainless steel	Outdoor	None
Electrical component supports	2, 7, 10	Carbon steel	Outdoor	Loss of material
			Indoor – air conditioned	None
		Carbon steel – galvanized	Outdoor Indoor – air conditioned	None

TABLE 3.6-17 (continued)
TURBINE BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Instrument rack and frames	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Reinforced concrete: Foundations (including switchgear/load center enclosures)	2, 4, 6, 7, 10	Concrete	Buried	None
Reinforced concrete: Walls Floors Roof (including switchgear/load center enclosures)	2, 3, 4, 6, 7, 10	Concrete	Outdoor Indoor – air conditioned	None
Reinforced masonry block walls (above groundwater elevation)	2, 3, 4, 7, 10	Concrete	Indoor – air conditioned	None
Unreinforced masonry block walls	2, 3, 4, 7, 10	Concrete	Outdoor	Cracking
Perimeter flood walls	8	Concrete	Outdoor	None
Anchorages/ embedments	2, 7, 10	Steel	Embedded/Encased	None
Weatherproofing	3	Pourable caulking Polymer sealant	Outdoor	Loss of seal

TABLE 3.6-17 (continued)
TURBINE BUILDING

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Pipe trench penetrations	8	Promatec flexible seal	Outdoor	Loss of seal
Stop logs	8	Wood	Outdoor	Loss of material
		Aluminum	Outdoor	None

TABLE 3.6-18
TURBINE GANTRY CRANES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Runway rails Rail anchorages/ embedments Leg truck beams Leg end frames Main girders Cab Ladders and stairways Platforms Railings Trolley rails Trolley structure	7	Carbon steel	Outdoor	Loss of material
Electrical enclosures	7	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None

TABLE 3.6-19
TURKEY POINT UNITS 1 AND 2 CHIMNEYS

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete chimney	7	Concrete	Outdoor	None
Reinforced concrete foundations (above groundwater elevation)	7	Concrete	Buried	None

TABLE 3.6-20
YARD STRUCTURES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Anchorages/ embedments exposed surfaces	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Miscellaneous steel	7	Carbon steel	Outdoor	Loss of material
			Borated water leaks	Loss of material
		Carbon steel – galvanized	Outdoor	None
			Borated water leaks	Loss of material
Safety-related supports (pipe supports and component supports)	2, 10	Carbon steel	Outdoor	Loss of material
			Borated water leaks	Loss of material
Non-safety related pipe segments between class break and seismic anchor	2	Carbon steel	Outdoor	Loss of material
			Borated water leaks	Loss of material
		Stainless steel	Outdoor	None

TABLE 3.6-20 (continued)
YARD STRUCTURES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Non-safety related supports	7, 10	Carbon steel	Outdoor	Loss of material
			Borated water leaks	Loss of material
Cable trays Conduits	2, 3, 7, 10	Carbon steel – galvanized	Outdoor	None
			Outdoor - wetted	Loss of material
			Borated water leaks	Loss of material
Cable tray, conduit and electrical supports	2, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor – wetted	Loss of material
			Outdoor	None
		Carbon steel Carbon steel – galvanized	Borated water leaks	Loss of material
Electrical enclosures	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material

TABLE 3.6-20 (continued)
YARD STRUCTURES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Instrument racks and frames	2, 3, 7, 10	Carbon steel	Outdoor	Loss of material
		Carbon steel – galvanized	Outdoor	None
Pipe whip restraints	11	Carbon steel	Outdoor	Loss of material
Reinforced concrete foundations: (above groundwater elevation) 3A and 3B emergency diesel generator fuel oil transfer pumps Unit 3 emergency diesel generator fuel oil storage tank Refueling water storage tanks Condensate storage tanks Auxiliary feedwater pumps	2, 10	Concrete	Outdoor	None

TABLE 3.6-20 (continued)
YARD STRUCTURES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete foundations: (above groundwater elevation) Demineralized water tank Diesel driven instrument air compressors Diesel driven standby steam generator feedwater pump Raw water tanks Diesel fire pump fuel storage tank Electric fire pump Fire water jockey pump	10	Concrete	Outdoor	None
Reinforced concrete foundations for pipe supports (above groundwater elevation)	2, 7, 10	Concrete	Outdoor	None
Reinforced concrete: Electrical duct banks Electrical manholes (above groundwater elevation)	3, 10	Concrete	Outdoor	None

TABLE 3.6-20 (continued)
YARD STRUCTURES

Component/ Commodity Group	Intended Function (See Table 3.6-1)	Material	Environment	Aging Effects Requiring Management
Reinforced concrete intake cooling water piping (above groundwater elevation)	7	Concrete	Buried	None
Reinforced concrete pipe trenches (above groundwater elevation)	2, 10	Concrete	Outdoor	None
Anchorages/ embedments (above groundwater elevation)	2, 7, 10	Steel	Embedded/Encased	None

3.7 ELECTRICAL AND INSTRUMENTATION AND CONTROLS

Section 2.5 provides a description of the electrical/I&C components requiring aging management review for license renewal. This section provides the results of the aging management review of the electrical/I&C components. The results of this section are also summarized in Table 3.7-5.

3.7.1 AGING EFFECTS REQUIRING MANAGEMENT

3.7.1.1 NON-ENVIRONMENTALLY QUALIFIED INSULATED CABLES AND CONNECTIONS, AND ELECTRICAL/I&C PENETRATIONS

An evaluation published by the Department of Energy (DOE), "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations" [Reference 3.7-1, (DOE Cable AMG)], provides a comprehensive compilation and evaluation of information on the topics of insulated cables and connections, spliced connections, and terminal blocks. The electrical/I&C non-metallic materials are evaluated with the cable and connector materials in this evaluation. The DOE Cable AMG evaluated the stressors acting on cable and connection components, industry data on aging and failure of these components, and the maintenance activities performed on cable systems. Also evaluated were the main subsystems within cables, including the conductors, insulation, shielding, tape wraps, and jacketing, as well as all subcomponents associated with each type of connection.

The principal aging mechanisms and anticipated effects resulting from environmental and operating stresses were identified, evaluated, and correlated with plant experience to determine whether the predicted effects are consistent with field experience. As such, the information, evaluations, and conclusions contained in the DOE Cable AMG are used for the evaluation of aging effects in this subsection.

The most significant and observed aging mechanisms for insulated cables and connections are listed in the DOE Cable AMG, Table 4-18. The aging mechanisms from that table are used in this subsection as the starting point for identifying aging effects for insulated cables and connections. The potential aging effects along with the applicable stressors that are evaluated for insulated cables and connections are presented in Table 3.7-1 and are discussed in the following subsections.

3.7.1.1.1 LOW-VOLTAGE METAL CONNECTOR CONTACT SURFACES MOISTURE AND OXYGEN

The DOE Cable AMG, Section 3.7.2.1.3, states that 3% of all low-voltage metal connector failures were identified as being caused by moisture intrusion. In each case, the source of moisture was precipitation. Based on the total number of reported connector failures in the DOE Cable AMG, moisture intrusion accounted for only 10 failures in all of the operating plants in the United States.

Structures where electrical/I&C components may be exposed to moisture are indicated in Table 3.7-2. The potential moisture sources from Table 3.7-2 that are applicable to connectors at Turkey Point are precipitation and potential boric acid leaks. All metal connectors are located in enclosures or protected from the environment with Raychem splices. Thus, aging effects related to moisture and oxygen do not require management for low-voltage connectors at Turkey Point.

Note: Electrical enclosures are treated as structural components and are discussed with each structure, as applicable, in Section 3.6.

3.7.1.1.2 LOW-VOLTAGE METAL COMPRESSION FITTINGS –- VIBRATION AND TENSILE STRESS

The aging mechanism of mechanical stress will not result in aging effects requiring management for the following reasons:

- Damage to cables during installation at Turkey Point is unlikely due to standard installation practices, which include limitations on cable pulling tension and bend radius. Even though installation damage is unlikely, most (including all safety related) cables are tested after installation and before operation. Failures induced by installation damage generally occur within a short time after the damaged cable is energized.
- NRC resolution of License Renewal Issue No.98-0013 [Reference 3.7-2], which states, “Based on the above evaluation, the staff concludes that the issue of degradation induced by human activities need not be considered as a separate aging effect and should be excluded from an aging management review.”
- Mechanical stress due to forces associated with electrical faults is mitigated by the fast action of circuit protective devices at high currents. However, mechanical stress due to electrical faults is not considered an aging mechanism since such faults are infrequent and random in nature.
- Vibration is generally induced in cables and connections by the operation of external equipment, such as compressors, fans, and pumps. Vibration can affect cable connections at a running motor by producing fatigue damage of the metallic cable or termination components in the immediate vicinity of the connection point. Normally, there has to be some physical damage as well to have an effect (e.g., a nicked connector). Terminations at equipment are part of the equipment and are inspected and maintained along with the

equipment. These terminations are not within the evaluation boundary for insulated cable and connections and are not included in the insulated cable and connection review.

- Manipulation of cables is not considered an aging mechanism since such manipulation occurs during maintenance activities. Such activities require post-maintenance testing to detect any deficiencies in the cables. Any evidence of cable abnormalities would result in the condition being addressed under the corrective action program.

3.7.1.1.3 MEDIUM-VOLTAGE CABLE AND CONNECTIONS AND ELECTRICAL/I&C PENETRATION INSULATION MOISTURE AND VOLTAGE STRESS

The DOE Cable AMG, Section 3.7.4, describes a survey of 25 fossil and nuclear power plants that was conducted to determine the number and types of medium-voltage cable failures that have occurred. The survey identified only 27 failures in almost 1000 plant-years of experience. The failures that occurred, other than moisture-produced water trees, were related to wetting in conjunction with manufacturing defects or damaged terminations due to improper installation, and were not related to aging effects.

Electrical/I&C penetrations are not located in structures exposed to outside ambient conditions and, therefore, not subjected to moisture.

Structures where electrical/I&C cable and connectors may be exposed to moisture are indicated in Table 3.7-2. The effects of moisture-produced water trees on medium-voltage cable were examined in Section 4.1.2.5 of the DOE Cable AMG. Water trees occur when the insulating materials are exposed to long-term, continuous electrical stress and moisture. These trees eventually result in breakdown of the dielectric materials and ultimate failure. The growth and propagation of water trees is somewhat unpredictable and few occurrences have been noted for cables operated below 15kV. Water treeing is a long-term degradation and failure phenomenon that is documented only for medium-voltage electrical cable with cross-linked polyethylene (XLPE) or high molecular weight polyethylene (HMWPE) insulation. However, some cables are located in structures exposed to outside ambient conditions and are evaluated for the potential of moisture-produced water trees.

Turkey Point Units 3 and 4 medium-voltage applications, defined as 2kV to 15kV, use lead sheath cable to prevent effects of moisture on the cables. In addition,

Turkey Point does not use XLPE or HMWPE insulated cables in medium-voltage applications. Therefore, aging effects related to cable exposed to moisture and voltage stress do not require management at Turkey Point.

3.7.1.1.4 MEDIUM- AND LOW-VOLTAGE CABLE AND CONNECTIONS AND ELECTRICAL/I&C PENETRATION INSULATION RADIATION AND OXYGEN

The DOE Cable AMG, Section 4.1.4, Table 4-7, provides a threshold value and a moderate dose for various insulating materials. The threshold value is the amount of radiation that causes incipient to mild insulation damage. Once this threshold is exceeded, damage to the insulation increases from mild to moderate to severe as the total dose increases. The moderate damage value indicates the value at which the insulating material has been damaged but is still functional. Turkey Point evaluations use the moderate damage dose from the DOE Cable AMG as the limiting radiation value shown in Table 3.7-3, unless otherwise noted in the table.

The maximum operating dose shown in Table 3.7-3 includes the maximum 60-year normal exposure for inside Containment. This is conservative, especially for cables located outside Containment.

A comparison of the maximum operating dose and the moderate damage doses in Table 3.7-3 shows that all of the insulation materials included in this aging management review will not exceed the moderate damage doses. Therefore, aging effects caused by radiation exposure will not adversely affect the intended function of insulated cables and connections and electrical/I&C penetrations during the extended period of operation. Therefore, aging effects related to radiation do not require management for cables and connections and electrical/I&C penetrations included in the aging management review.

3.7.1.1.5 MEDIUM- AND LOW-VOLTAGE CABLE AND CONNECTIONS AND ELECTRICAL/I&C PENETRATION INSULATION HEAT AND OXYGEN

A maximum operating temperature was developed for each insulation type based on cable applications at Turkey Point Units 3 and 4. The maximum operating temperature indicated in Table 3.7-4 incorporates a conservative value for self-heating for power applications combined with the maximum design ambient temperature.

The Arrhenius method, as described in EPRI NP-1558, "A Review of Equipment Aging Theory and Technology" [Reference 3.7-3], was used to determine the

maximum continuous temperature to which the insulation material can be exposed so that the material has an indicated "endpoint of 60 years." These limiting temperatures for 60 years of service are provided in Table 3.7-4.

A comparison of the maximum operating temperature to the maximum 60-year continuous use temperature for the various insulation materials indicates that, except for polyethylene (PE) and Butyl used in power applications, all of the insulation materials used in low- and medium-voltage power cables and connections can withstand the maximum operating temperatures for at least 60 years.

PE AND BUTYL CABLE INSULATION

The maximum operating temperatures, including self-heating, for PE and Butyl are 138.7°F and 132.6°F, respectively. The maximum temperatures for a 60-year life are 131.0°F for PE and 125.1°F for Butyl, which are 7.7°F and 7.5°F, respectively, less than the maximum operating temperatures. This difference is very small and is considered to be within the conservatisms incorporated in the maximum operating temperatures and the maximum 60-year continuous use temperatures, as discussed below.

Research funded by the NRC and published in NUREG/CR-6384, "Literature Review of Environmental Qualification of Safety-Related Electric Cables" [Reference 3.7-4], determined that the retention-of-elongation of most cable insulation materials can be reduced to 0% and the insulation will still be capable of withstanding a postulated loss-of-coolant accident (LOCA) and remain functional. In addition, preliminary results of environmental qualification research on low-voltage electrical cables were presented at a NRC public meeting on March 19, 1999. Preliminary conclusions from LOCA tests 1, 2, and 3 of the NRC research program indicate that, “Electric cables with insulation elongation-at-break values as low as 5% performed acceptably under accident conditions” [Reference 3.7-5]. Therefore, the maximum 60-year continuous use temperatures for typical cable insulation are significantly higher than the maximum temperatures shown in Table 3.7-4.

Butyl and PE insulated cables and connections are not used in containment and are not subjected to an accident environment. Therefore, the endpoints chosen for this aging management review are extremely conservative and the 60-year endpoint values can be reduced without a loss of function, thus resulting in higher maximum 60-year continuous use temperatures.

The maximum operating temperatures in Table 3.7-4 include a calculated self-heating temperature rise that assumes normal operation 100% of the time since receipt of the original operating licenses. In addition, as identified in Table 3.0-2, the actual daily and seasonal temperatures vary from 30ºF – 95ºF, which is less than the 104ºF limit assumed in the calculation of 60-year lifetime for Butyl and PE. The Turkey Point units have historically operated less than 90% of the time since receipt of the original operating licenses. This amount of shutdown time lessens the amount of aging actually occurring and thus extends the lives of the materials.

Given these conservatisms, there is reasonable assurance that PE and Butyl insulated cables will not thermally age to the point at which they will not be able to perform their intended function during the period of extended operation. Aging effects related to heat and oxygen do not require management for cables and electrical/I&C penetrations included in the aging management review.

3.7.1.2 UNINSULATED GROUND CONDUCTORS

The ground cable material used at Turkey Point Units 3 and 4 is copper. Copper is a good choice for this application because of its high electrical conductivity, high fusing temperature, and high corrosion resistance. Copper is also relatively strong, and it is easy to join by welding, compression, or clamping. Ground connections are commonly made with welds or mechanical type connectors, which include compression-, bolted-, and wedge-type devices.

Review of available industry technical information regarding material aging revealed that there are no aging effects requiring management for copper grounding materials. In addition, a review of industry and plant operating experiences did not identify any failures of copper ground systems due to aging effects. Also, several underground portions of the Turkey Point grounding system were inspected during plant modifications to add two additional emergency diesel generators, in 1990 and 1991, and no aging-related effects were identified. The system was approximately 20 years old at the time of that inspection. The portion of the grounding system inspected is buried in the same type of soil as other underground portions of the grounding system. Therefore, based on industry and plant-specific experiences, no aging effects requiring management were identified for the plant grounding system.

3.7.2 OPERATING EXPERIENCE

3.7.2.1 INDUSTRY EXPERIENCE

The DOE Cable AMG review includes an industry-wide operating experience review of failures and aging effects of electrical cables and terminations. No aging effects were identified from the DOE Cable AMG beyond those already identified in Subsection 3.7.1.

An incident occurred at the Davis-Besse Nuclear Generating Station on October 2, 1999. A component cooling water pump tripped as a result of a phase-to-ground fault on a medium-voltage 3-phase power cable. The cable was installed in a 4-inch polyvinyl chloride (PVC) conduit, which runs partially underground, and had been in service for about 23 years.

As noted above, all medium-voltage applications (2kV to 15kV) at Turkey Point use lead sheath cable to prevent the effects of moisture on the cables. Based on Turkey Point's medium-voltage cable design, this incident is not applicable to medium-voltage cables at Turkey Point.

3.7.2.2 PLANT-SPECIFIC EXPERIENCE

Turkey Point Units 3 and 4 operating experience was reviewed to validate the identified aging effects requiring management. This review included a survey of Turkey Point non-conformance reports, licensee event reports, and condition reports for any documented instances of electrical/I&C component aging, in addition to interviews with responsible engineering personnel. No aging effects were identified from this review beyond those identified in Subsection 3.7.1. In particular, the review did not identify any instances where insulated cables or connections have failed due to heat-, radiation-, or moisture-related aging effects.

3.7.3 CONCLUSION

The review of industry information, NRC generic communications, and Turkey Point Units 3 and 4 operating experience identified no additional aging effects beyond those discussed in Subsection 3.7.1. Table 3.7-5 contains the results of the aging management review for electrical/I&C components and summarizes that there are no aging effects requiring management for electrical/I&C components. Based on the aging management review, the intended functions of electrical/I&C components will be maintained consistent with the current licensing basis for the period of extended operation.

3.7.4 REFERENCES

3.7-1 SAND 96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations," Sandia National Laboratories for the U. S. Department of Energy, September 1996.

3.7-2 C. I. Grimes (NRC) letter to D. J. Walter (NEI), "License Renewal Issue No. 98-0013, Degradation Induced Human Activities," June 5, 1998.

3.7-3 EPRI NP-1558, “A Review of Equipment Aging Theory and Technology,” Electric Power Research Institute, September 1980.

3.7-4 NUREG/CR-6384, “Literature Review of Environmental Qualification of Safety-Related Electric Cables,” Vol. 1, Brookhaven National Laboratory for the U.S. Nuclear Regulatory Commission, April 1996.

3.7-5 NRC Public Meeting Handouts, Presented by Brookhaven National Laboratory, "Environmental Qualification Research on Low-Voltage Electric Cables," Sponsored by the U.S. NRC, Office of Nuclear Regulatory Research, March 19, 1999.

TABLE 3.7-1
POTENTIAL AGING EFFECTS
ADAPTED FROM DOE CABLE AMG TABLE 4-18

Voltage Category[1]	Component	Applicable Stressor	Potential Aging Effects
Low voltage	Metal connector contact surfaces	Moisture and oxygen	Increased resistance and heating; loss of circuit continuity
	Compression fitting	Vibration Tensile stress	Loss of circuit continuity High resistance
Medium voltage	Cable and connections, electrical/I&C penetration insulation	Moisture and voltage stress	Electrical failure (breakdown of insulation)
Medium and low voltage	Cable and connections, electrical/I&C penetration insulation	Radiation and oxygen	Reduced insulation resistance; electrical failure
		Heat and oxygen	Reduced insulation resistance; electrical failure

NOTE: 1. Low Voltage: less than 2kV; Medium Voltage: 2kV to 15kV

TABLE 3.7-2
MOISTURE EXPOSURE SOURCES

Structure	Environment	Potential Moisture Exposure Source
Turbine Building Intake Structure Main Steam and Feedwater Platforms Yard Structures	Outdoor	Precipitation
Yard Structures	Indoor – not air conditioned (wetted)	Standing water in cable trenches
	Buried	Surface water drainage and soil moisture
Containments Auxiliary Building Yard Structures	Borated water leaks	Systems containing boric acid

TABLE 3.7-3
INSULATION MATERIAL
RADIATION EXPOSURE COMPARISON

Insulation Material	Maximum Operating Dose	Moderate Damage Dose	Additional Information
EP, EPR, FR, FR2, FR3, EPDM, FR-EPR	5.26×10^5 rads	5×10^7 rads	
Epoxy	5.26×10^5 rads	8×10^{16} rads	
Fiberglass (Mineral Insulated)	5.26×10^5 rads	None	Fiberglass is spun glass and, except for some changes in color, is not affected by radiation.
Kerite-HTK	5.26×10^5 rads	1×10^8 rads	Although no value for Kerite is listed in DOE Cable AMG, Table 4-7, the insulation material has been tested many times for the nuclear industry at total doses in excess of 1×10^8 rads. This value is used as the moderate damage dose.
Phenolic	5.26×10^5 rads	~4×10^7 rads	The radiation resistance of phenolic varies depending on what it is “filled” with (e.g., glass, asbestos). The values for “unfilled” phenolic are chosen since it is the weakest.
Silicon rubber	5.26×10^5 rads	3×10^6 rads	Silicone rubber is only used in electrical/I&C penetration pigtail cable insulation.
XLP, XLPE, FR-XLPE, Vulkene	5.26×10^5 rads	1×10^8 rads	
Butyl	5.26×10^5 rads	5×10^6 rads	
Hypalon	5.26×10^5 rads	2×10^6 rads	
Kapton	5.26×10^5 rads	2×10^8 rads	
Nylon	5.26×10^5 rads	2×10^6 rads	There are many formulations of nylon, a material originally developed by the DuPont Company. The values used here are for the most common formulation (general purpose) of nylon that is referred to as Nylon 66 and is designated Zytel 101. Zytel is the DuPont trademark for many different nylon resins.
PE	5.26×10^5 rads	2×10^7 rads	
PVC	5.26×10^5 rads	2×10^7 rads	
Micatemp – pressurizer heaters	5.26×10^5 rads	1×10^9 rads	

TABLE 3.7-4
INSULATION MATERIAL
TEMPERATURE EXPOSURE COMPARISON

Insulation Material	Maximum Operating Temperature [1]	Maximum Temperature For 60-Year Life	60-Year Endpoint
Phenolic	165.6°F (74.2°C)	220.5°F (104.7°C)	50% Retention of Impact Strength
XLPE	165.6°F (74.2°C)	188.1°F (86.7°C)	60% Retention-of-Elongation
Kapton	147.6°F (64.2°C)	248°F (120.0°C)	Failure
EPR, EPDM	147.6°F (64.2°C)	154.9°F (68.3°C)	40% Retention-of-Elongation
FR3	136.8°F (58.2°C)	166.6°F (74.8°C)	20% Retention-of-Elongation
Kerite-HTK	136.0°F (57.8°C)	185.4°F (85.2°C)	20% Retention-of-Elongation
PE	138.7°F (59.3°C)	131°F (55.0°C)	T_{75} Induction Period
Butyl	132.6°F (55.9°C)	125.1°F (51.7°C)	40% Retention-of-Elongation
FR	125.8°F (52.0°C)	141.5°F (60.8°C)	50% Retention-of-Elongation
EP	123.3°F (50.7°C)	154.9°F (68.3°C)	40% Retention-of-Elongation
Epoxy	122.0°F (50.0°C)	399.2°F (204.0°C)	50% Retention of Impact Strength
Silicon rubber	122.0°F (50.0°C)	273°F (133.9°C)	50% Retention-of-Elongation
Vulkene	122.0°F (50.0°C)	188.1°F (86.7°C)	60% Retention-of-Elongation
FR2	122.0°F (50.0°C)	192.5°F (89.2°C)	20% Retention-of-Elongation
FR-EP	122.0°F (50.0°C)	154.9ºF (68.3ºC)	40% Retention-of-Elongation
XLP, FR-XLPE	104.0°F (40.0°C)	188.1°F (86.7°C)	60% Retention-of-Elongation
Nylon	104.0°F (40.0°C)	129.9°F (54.4°C)	28% Retention of Tensile Strength
PVC	104.0°F (40.0°C)	112.0°F (44.4°C)	Mean-Time-To-Failure
Hypalon	104.0°F (40.0°C)	154ºF (67.8ºC)	50% Retention-of-Elongation
Fiberglass, Micatemp-PHC	Not required	Does not age from heat	Not applicable

NOTE: 1. Includes applicable self-heating temperature rise.

TABLE 3.7-5
ELECTRICAL/I&C COMPONENTS AGING MANAGEMENT REVIEW SU

Component / Commodity Group	Intended Function	Insulation Material	Environment [1]	Aging Effect Requiring Management	
Non-environmentally qualified cables and connections, and electrical/I&C penetrations (electrical power circuits)	To electrically connect specified sections of an electrical circuit to deliver voltage, current, or signal	Butyl, EP, EPR, EPDM, FR, FR2, FR3, Kapton, PE, Kerite-HTK, XLPE, Phenolic, Micatemp-PHC, Epoxy, and Silicone rubber	Moisture Temperature Elevated Temperature Ohmic Heating Radiation	None	No
Non-environmentally qualified cables and connections, and electrical/I&C penetrations (instrumentation and control circuits)	To electrically connect specified sections of an electrical circuit to deliver voltage, current, or signal	Butyl, EP, EPR, EPDM, FR, FR2, FR3, FR-EPR, Nylon, Fiberglass, Hypalon, Kapton, PE Kerite-HTK, Phenolic, PVC, XLP, XLPE, Vulkene, FR-XLPE, Epoxy, and Silicone rubber	Moisture Temperature Elevated Temperature Radiation	None	No
Uninsulated ground conductors	To electrically connect specified sections of an electrical circuit to deliver voltage, current, or signal	Uninsulated copper	Moisture Temperature Elevated Temperature Radiation	None	No

NOTE: 1. All environments are external except ohmic heating, which is considered an internal environment.

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 Turkey Point Integrated Plant Assessment, which is described in Chapters 2 and 3. The second area of technical review required for license renewal is the identification and evaluation of plant-specific time-limited aging analyses and exemptions, which are provided in this chapter. The evaluations included in this chapter meet the requirements contained in 10 CFR 54.21(c) and allow the NRC to make the finding contained in 10 CFR 54.29(a)(2).

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

10 CFR 54.21(c) requires an evaluation of time-limited aging analyses be provided as part of the application for a renewed license. Time-limited aging analyses 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 Turkey Point-specific time-limited aging analyses 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" [Reference 4.1-1]. Calculations and evaluations that meet the six criteria of 10 CFR 54.3 were identified from the Technical Specifications, UFSAR, docketed licensing correspondence, and applicable Westinghouse WCAPs. The calculations and evaluations that meet all six criteria of 10 CFR 54.3 are the Turkey Point-specific time-limited aging analyses listed in Table 4.1-1.

As required by 10 CFR 54.21(c)(1), an evaluation of Turkey Point-specific time-limited aging analyses 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 function(s) will be adequately managed for the period of extended operation.

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

4.1.2 IDENTIFICATION OF EXEMPTIONS

The requirements at 10 CFR 54.21(c) also stipulate 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 time-limited aging analyses as defined in 10 CFR 54.3. The identification was performed by evaluating the basis for each active 10 CFR 50.12 exemption to determine whether the exemption was based on a time-limited aging analysis. No 10 CFR 50.12 exemptions involving a time-limited aging analysis as defined in 10 CFR 54.3 were identified for Turkey Point Units 3 and 4.

4.1.3 REFERENCES

4.1-1 NEI 95-10, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Revision 1, Nuclear Energy Institute, January 2000.

TABLE 4.1-1
TIME-LIMITED AGING ANALYSES

TLAA Category	Analysis	Resolution [10 CFR 54.21(c)(1) Section]	Section
Reactor Vessel Irradiation Embrittlement	Pressurized Thermal Shock	(ii) projected to the end of the period of extended operation	4.2.1
	Upper-Shelf Energy	(ii) projected to the end of the period of extended operation	4.2.2
	Pressure-Temperature Limits	(ii) projected to the end of the period of extended operation	4.2.3
Metal Fatigue	ASME Section III, Class 1 Components	(i) remains valid for the period of extended operation	4.3.1
	Reactor Vessel Underclad Cracking	(ii) projected to the end of the period of extended operation	4.3.2
	Reactor Coolant Pump Flywheel	(i) remains valid for the period of extended operation	4.3.3
	ANSI B31.1 Piping	(i) remains valid for the period of extended operation	4.3.4
Environmental Qualification	Anaconda Cables	(i) remains valid for the period of extended operation	4.4.1.1
	AIW Cables	(i) remains valid for the period of extended operation	4.4.1.2
	ASCO Solenoid Valves	(i) remains valid for the period of extended operation	4.4.1.3
	Brand Rex Coaxial Cables	(i) remains valid for the period of extended operation	4.4.1.4
	Brand Rex Instrument Cables	(i) remains valid for the period of extended operation	4.4.1.5
	Conax Conduit Seals	(i) remains valid for the period of extended operation	4.4.1.6
	Conax Penetrations	(i) remains valid for the period of extended operation	4.4.1.7
	Conax Unitized Resistance Temperature Detectors	(i) remains valid for the period of extended operation	4.4.1.8
	Champlain Cables	(i) remains valid for the period of extended operation	4.4.1.9
	Crouse Hinds Penetrations	(i) remains valid for the period of extended operation	4.4.1.10
	General Atomic Radiation Monitors	(i) remains valid for the period of extended operation	4.4.1.11
	General Cables	(i) remains valid for the period of extended operation	4.4.1.12

TABLE 4.1-1 (continued)
TIME-LIMITED AGING ANALYSES

TLAA Category	Analysis	Resolution [10 CFR 54.21(c)(1) Section]	Section
Environmental Qualification (continued)	General Electric Cables	(i) remains valid for the period of extended operation	4.4.1.13
	General Electric Terminal Blocks	(i) remains valid for the period of extended operation	4.4.1.14
	Joy Emergency Containment Cooler and Emergency Containment Filtration Fan Motors	(i) remains valid for the period of extended operation	4.4.1.15
	Limitorque Valve Operators with Reliance Motors For Use Inside Containment	(i) remains valid for the period of extended operation	4.4.1.16
	Limitorque Valve Operators with Reliance Class H(RH) Insulation For Use Inside Containment	(i) remains valid for the period of extended operation	4.4.1.17
	Limitorque Valve Operators with Reliance Motors For Use Outside Containment	(i) remains valid for the period of extended operation	4.4.1.18
	Limitorque Valve Operators with Peerless Motors For Use Outside Containment	(i) remains valid for the period of extended operation	4.4.1.19
	Okonite Cables	(i) remains valid for the period of extended operation	4.4.1.20
	Raychem Heat Shrink Sleeving	(i) remains valid for the period of extended operation	4.4.1.21
	Raychem Cables	(i) remains valid for the period of extended operation	4.4.1.22
	MacWorth Rees Pushbutton Stations	(i) remains valid for the period of extended operation	4.4.1.23
	Rockbestos Cables	(i) remains valid for the period of extended operation	4.4.1.24
	Samuel Moore Cables	(ii) projected to the end of the period of extended operation	4.4.1.25
	3M Insulating Tape and Scotchfil	(i) remains valid for the period of extended operation	4.4.1.26
	Westinghouse Residual Heat Removal Pump Motors	(i) remains valid for the period of extended operation	4.4.1.27
	Westinghouse Containment Spray Pump Motors	(i) remains valid for the period of extended operation	4.4.1.28

TABLE 4.1-1 (continued)
TIME-LIMITED AGING ANALYSES

TLAA Category	Analysis	Resolution [10 CFR 54.21(c)(1) Section]	Section
Environmental Qualification (continued)	Westinghouse Safety Injection Pump Motors	(i) remains valid for the period of extended operation	4.4.1.29
	Combustion Engineering Mineral Insulated Cables and Connectors	(i) remains valid for the period of extended operation	4.4.1.30
	Kerite HTK/FR Cables	(i) remains valid for the period of extended operation	4.4.1.31
	Kerite FR2/FR Cables	(i) remains valid for the period of extended operation	4.4.1.32
	Kerite FR/FR Cables	(i) remains valid for the period of extended operation	4.4.1.33
	Kerite HTK/FR Power Cables	(i) remains valid for the period of extended operation	4.4.1.34
	Teledyne Thermatics Cables	(i) remains valid for the period of extended operation	4.4.1.35
	Weed Resistance Temperature Detectors	(i) remains valid for the period of extended operation	4.4.1.36
	Amerace NQB Terminal Blocks	(i) remains valid for the period of extended operation	4.4.1.37
	Patel/EGS Conformal Splices	(i) remains valid for the period of extended operation	4.4.1.38
	Patel/EGS Grayboot Connectors	(i) remains valid for the period of extended operation	4.4.1.39
Containment Tendon Loss of Prestress	Containment Tendon Loss of Prestress	(ii) projected to the end of the period of extended operation	4.5
Containment Liner Plate Fatigue	Containment Liner Plate Fatigue	(i) remains valid for the period of extended operation	4.6
Other Plant-Specific Time-Limited Aging Analyses	Bottom Mounted Instrumentation Thimble Tube Wear	(iii) the effects of aging on the intended function will be adequately managed for the period of extended operation	4.7.1
	Emergency Containment Cooler Tube Wear	(iii) the effects of aging on the intended function will be adequately managed for the period of extended operation	4.7.2
	Leak-Before-Break for Reactor Coolant System Piping	(ii) projected to the end of the period of extended operation	4.7.3
	Crane Load Cycle Limit	(i) remains valid for the period of extended operation	4.7.4

4.2 REACTOR VESSEL IRRADIATION EMBRITTLEMENT

This group of time-limited aging analyses concerns the effect of irradiation embrittlement on the belt-line regions of the Turkey Point Units 3 and 4 reactor vessels, and how this mechanism affects analyses that provide operating limits or address regulatory requirements. The calculations discussed in this section use predictions of the cumulative effects on the reactor vessels from irradiation embrittlement. The calculations are based on periodic assessment of the neutron fluence and resultant changes in the reactor vessel material fracture toughness.

The intermediate and lower shells and welds that join them in the beltline region (adjacent to the reactor core) of the reactor vessel are fabricated from low alloy steels. These ferritic steels exhibit a ductile-brittle transition that results in fracture toughness property changes as a function of both temperature and irradiation. The material property of particular importance in assessing reactor vessel integrity is fracture toughness, which can be defined as the capability of a material to resist sudden failure caused by crack propagation. Fracture toughness is reduced by neutron irradiation. The measure of fracture toughness of the reactor vessel materials when the reactor vessel is above the brittle fracture/ductile failure transition temperature is referred to as upper-shelf energy. Upper-shelf energy is related to the ability of a material to resist ductile tearing. In addition, the temperature at which the brittle fracture/ductile failure transition occurs increases with increasing radiation. This shift in the transition temperature is referred to as the shift in reference nil ductility transition temperature (RT_{NDT}).

The effect of embrittlement due to neutron bombardment is evaluated for reactor vessel temperatures throughout the range of normal operating values. Heatup and cooldown curves consider normal, relatively slow thermal transients. Pressurized thermal shock transients are characterized by a rapid and significant decrease in reactor coolant temperature with high pressure in the reactor vessel. The high reactor vessel thermal stresses, when combined with the pressure stresses, are assumed to initiate the propagation of a small flaw that is postulated to exist in the reactor vessel beltline. Postulated high pressures could cause propagation of the flaw through the reactor vessel wall.

The welds in the reactor vessel are basically the same material as the parts being joined and may be considered to be included in the preceding discussions. The chemistry differences between weld metal and base metal affect the material

properties that are degraded by embrittlement; therefore, the welds are evaluated separately when considering the aforementioned aging effect.

4.2.1 PRESSURIZED THERMAL SHOCK

The requirements in 10 CFR 50.61 provide rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of the maximum nil ductility reference temperature (RT_{PTS}) whenever a significant change occurs in projected values of RT_{PTS}, or upon request for a change in the expiration date for the operation of the facility.

The methods for calculating RT_{PTS} values are given in 10 CFR 50.61 and are consistent with the methods in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [Reference 4.2-1]. These accepted methods were used to calculate the RT_{PTS} for the Turkey Point reactor vessel limiting materials at 48 effective full power years, the end of the license renewal period. The calculated RT_{PTS} values for the Turkey Point reactor vessels at the end of the period of extended operation (48 effective full power years) are:

	Unit 3	Unit 4
Lower Shell	108. 4°F	64.7°F
Intermediate Shell	78.3°F	129.4°F
Circumferential Weld	297.4°F	297.4°F

The calculated RT_{PTS} values at 48 effective full power years for the Turkey Point reactor vessels are less than the 10 CFR 50.61(b)(2) screening criteria of 270° F for intermediate and lower shells and 300° F for the circumferential welds. Based upon the revised calculations, additional measures will not be required for the Turkey Point reactor vessels during the license renewal period.

The analysis associated with pressurized thermal shock has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.2 UPPER-SHELF ENERGY

The requirements on reactor vessel Charpy upper-shelf energy are included in 10 CFR 50, Appendix G. Specifically, 10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any of the reactor vessel material is predicted to drop below 50 ft-lb., as measured by Charpy V-notch specimen testing. The Turkey Point circumferential weld material previously fell below the 10 CFR 50, Appendix G, requirement of 50 ft-lb. At that time, a fracture mechanics evaluation was performed to demonstrate acceptable equivalent margins of safety against fracture. The NRC reviewed these evaluations, as documented in October 19, 1993 [Reference 4.2-2] and May 9, 1994 [Reference 4.2-3] letters to FPL. These references approved plant operation through the current license term (32 effective full power years).

A fracture mechanics evaluation was performed in accordance with Appendix K of ASME Section XI to demonstrate continued acceptable equivalent margins of safety against fracture through 48 effective full power years [Reference 4.2-4]. This evaluation concluded that the limiting weld for the Turkey Point reactor vessels satisfies the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix K, for ductile flaw extension and tensile instability.

The analysis associated with upper-shelf energy has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.3 PRESSURE-TEMPERATURE LIMITS

The requirements in 10 CFR 50, Appendix G stipulate that heatup and cooldown of the reactor pressure vessel be accomplished within established pressure-temperature limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced. Operation of the Reactor Coolant System is also limited by the net positive suction curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. Therefore, in order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G pressure-temperature limits and the net positive suction curves.

To address the period of extended operation, the 48 effective full power year projected fluences and the Turkey Point-specific reactor vessel material properties were used to determine the limiting material and calculate pressure-temperature limits for heatup and cooldown. The limiting material at all temperatures for the period of extended operation is the circumferential girth weld.

FPL has recently submitted to NRC a proposed license amendment for Turkey Point Units 3 and 4 to extend the service period for the pressure-temperature limit curves to a maximum of 32 effective full power years (EFPY), the end of the current license period [Reference 4.2-5]. The proposed license amendment also includes pressure-temperature limit and Low Temperature Overpressure Protection (LTOP) setpoints for 48 EFPY, the end of the period of extended operation. FPL has not requested NRC approval of the 48 EFPY pressure-temperature limit curves and LTOP setpoints at this time. A separate license amendment specifically requesting approval of the 48 EFPY pressure-temperature limit curves and LTOP setpoints will be submitted to the NRC in the future and prior to expiration of the proposed 32 EFPY pressure-temperature limit curves.

The analysis associated with reactor vessel pressure-temperature limit curves has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.2.4 REFERENCES

4.2-1 Regulatory Guide 1.99, Revision 2 , “Radiation Embrittlement of Reactor Vessel Materials,” U.S. Nuclear Regulatory Commission, February 1986.

4.2-2 L. Raghavan (NRC) letter to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Review of Babcock and Wilcox Owners Group Materials Committee Reports - Upper-Shelf Energy," October 19, 1993.

4.2-3 Richard P. Croteau (NRC) letter to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Generic Letter (GL) 92-01, Revision 1, Reactor Vessel Structural Integrity," May 9, 1994.

4.2-4 BAW-2312, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of Turkey Point Units 3 and 4 for Extended Life Through 48 Effective Full Power Years," Babcock and Wilcox, November 1997.

4.2-5 R. J. Hovey (FPL) letter to U. S. Nuclear Regulatory Commission, "Revised Pressure-Temperature (P/T) Curves, and Cold Overpressure Mitigation System (CMOS) Setpoints," July 7, 2000.

4.3 METAL FATIGUE

The thermal and mechanical fatigue analyses of plant mechanical components have been identified as time-limited aging analyses for Turkey Point. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the Turkey Point UFSAR.

4.3.1 ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, CLASS 1 COMPONENTS

The reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the Turkey Point Units 3 and 4 Nuclear Steam Supply System components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various Nuclear Steam Supply System components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycles applicable to the Class 1 components were assembled. The actual frequency of occurrence for the design basis cycles was determined and compared to the design cycle set. The severity of the actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis the design cycle profiles envelope actual plant operation. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation.

The analyses associated with verifying the structural integrity of the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

For license renewal, continuation of the Turkey Point Fatigue Monitoring Program into the period of extended operation will assure that the design cycle limits are not exceeded. The Fatigue Monitoring Program is considered a confirmatory program and is described in Appendix B.

4.3.2 REACTOR VESSEL UNDERCLAD CRACKING

In early 1971, an anomaly identified as grain boundary separation, perpendicular to the direction of the cladding weld overlay, was identified in the heat-affected zone of reactor vessel base metal. A generic fracture mechanics evaluation demonstrated that the growth of underclad cracks during a 40-year plant life is insignificant.

The evaluation was extended to 60 years using fracture mechanics evaluations based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service life. The 60-year evaluation shows insignificant growth of the underclad cracks.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.3.3 REACTOR COOLANT PUMP FLYWHEEL

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to potentially produce high-energy missiles in the unlikely event of failure. Conditions which may result in overspeed of the reactor coolant pump increase both the potential for failure and the kinetic energy. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway. An evaluation of the probability of failure over the extended period of operation was performed. It demonstrates that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack extension over a 60-year service life.

The analysis associated with the structural integrity of the reactor coolant pump flywheel has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.4 ANSI B31.1 PIPING

The Reactor Coolant System primary loop piping and balance-of-plant piping are designed to the requirements of ANSI B31.1, Power Piping. The exceptions are the Units 3 and 4 pressurizer surge lines and the Unit 4 Emergency Diesel Generator safety-related piping.

The pressurizer surge lines have been designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 1 and are included in Subsection 4.3.1.

The Unit 4 Emergency Diesel Generator safety-related piping has been designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 3, which is essentially the same as ANSI B31.1 design requirements. The evaluation of the Unit 4 Emergency Diesel Generator safety-related piping fatigue is, therefore, included in this subsection.

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was performed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and vary operating temperatures only during plant heatup and cooldown, during plant transients, or during periodic testing. The results of the evaluation for ANSI B31.1 piping systems demonstrate that the number of assumed thermal cycles will not be exceeded in 60 years of plant operation.

The analyses associated with ANSI B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.5 ENVIRONMENTALLY ASSISTED FATIGUE

Generic Safety Issue (GSI) 190 [Reference 4.3-1], was identified by the NRC staff because of concerns about the potential effects of reactor water environments on Reactor Coolant System component fatigue life during the period of extended operation. GSI-190 was closed in December 1999 [Reference 4.3-2], and concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, the NRC concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs.

Fatigue calculations that include consideration of environmental effects to establish cumulative usage factors could be treated as time-limited aging analyses (TLAAs) under 10 CFR Part 54 or they could be utilized to establish the need for an aging management program. In other words, the determination of whether a particular component location is to be included in a program for managing the effects of fatigue, and the characteristics of that program, should incorporate reactor water environmental effects.

To qualify as a TLAA, the analysis of concern must satisfy all six criteria defined in 10 CFR 54.3. Failure to satisfy any one of these criteria eliminates the analysis from further consideration as a TLAA. Fatigue design for Turkey Point Units 3 and 4 has been determined to be a TLAA, even though the design limits are based on cycles rather than an explicit time period. However, reactor water environmental effects, as described in GSI-190, are not included in the Turkey Point current licensing basis (CLB), such that the criterion specified in 10 CFR 54.3(a)(6) is not satisfied. Nevertheless, environmental effects on Class 1 component fatigue have been evaluated separately for Turkey Point to determine if any additional actions are required for the period of extended operation.

The Turkey Point approach to address reactor water environmental effects accomplishes two objectives, as illustrated in Figure 4.3-1. First, the TLAA on fatigue design has been resolved by confirming that the original transient design limits remain valid for the 60-year operating period. Confirmation by the Fatigue Monitoring Program will ensure these transient design limits are not exceeded. Second, reactor water environmental effects on fatigue life are examined using the most recent data from laboratory simulation of the reactor coolant environment. These two aspects of fatigue design are kept separate, since fatigue design for

Turkey Point is part of the plant CLB and a TLAA, while the consideration of reactor water environmental effects on fatigue life, as described in GSI-190, is not considered part of the Turkey Point CLB.

It is important to note that there are three areas of margin included in the Turkey Point Fatigue Monitoring Program that are worthy of consideration. These areas include margins resulting from actual cycle experience, severity of transients, and moderate environmental effects.

Margin Due to Actual Cycles: It has been concluded that the original 40-year design transient set for Class 1 components is valid for the 60-year extended operating period. Conservative projections conclude that the design transient limits will not be exceeded. Additional margin is available in the current Class 1 component fatigue analyses since fatigue usage factors for all Class 1 components remain below the allowable value of 1.0.

Margin Due to Transient Severity: Much of the conservatism in the fatigue calculational methodology is due to design basis transient definitions. It has been concluded that the severity of the original Turkey Point design transients bounds actual plant operation. Additional industry fatigue studies [References 4.3-3 through 4.3-6] conclude that the fatigue impact of conservative design basis transient definitions by themselves bound the contributing impact of reactor water environmental effects.

Margin Due to Moderate Environmental Effects: A portion of the safety factors applied to the ASME Code Section III fatigue design curves includes moderate environmental effects. While there is debate over exactly how much margin this represents, it is noteworthy to recognize this safety factor in this qualitative discussion of margin.

Considering the three margins above, the Turkey Point Fatigue Monitoring Program is conservative from an overall perspective. Nevertheless, specific assessment of potential environmental effects on fatigue is addressed below.

As a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratories (INEL) evaluated, in NUREG/CR-6260 [Reference 4.3-7], fatigue-sensitive component locations at plants designed by all four U. S. nuclear steam supply system vendors. The pressurized water reactor calculations, especially the early-vintage Westinghouse calculations, are directly relevant to Turkey Point. The

description of the "Older Vintage Westinghouse Plant" evaluated in NUREG/CR-6260 matches Turkey Point with respect to design code. In addition, the transient cycles considered in the evaluation match or bound Turkey Point design.

The critical fatigue-sensitive component locations chosen in NUREG/CR-6260 for the early-vintage Westinghouse plant were:

1) The reactor vessel shell and lower head
2) The reactor vessel inlet and outlet nozzles
3) The pressurizer surge line (including the pressurizer and hot leg nozzles)
4) The Reactor Coolant System piping charging system nozzle
5) The Reactor Coolant System piping safety injection nozzle
6) The Residual Heat Removal System Class 1 piping

Note that for the latter three component locations, INEL performed representative design basis fatigue calculations, because early-vintage Westinghouse plants, including Turkey Point, utilize ANSI B31.1 design methodology for the majority of the Class 1 piping.

NUREG/CR-6260 calculated fatigue usage factors for these locations utilizing the interim fatigue curves provided in NUREG/CR-5999 [Reference 4.3-8]. The results of the NUREG/CR-6260 analyses were then utilized to scale up the Turkey Point plant-specific usage factors for the same locations to account for environmental effects. Generic industry studies performed by EPRI and NEI were also considered in this aspect of the evaluation [References 4.3-3 through 4.3-6], as well as environmental data that have been collected and published subsequent to the generic industry studies [References 4.3-9 and 4.3-10]. Based on these adjustments, only the pressurizer surge line piping required further evaluation for the period of extended operation. The pressurizer surge line piping is addressed below.

In lieu of additional analyses to refine the usage factor for the pressurizer surge line, Turkey Point has selected aging management to address pressurizer surge line fatigue during the period of extended operation. In particular, the potential for crack initiation and growth, including reactor water environmental effects, is adequately managed during the extended period of operation by the Turkey Point ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program.

The Turkey Point surge lines are 12-inch schedule 140 lines connected to the pressurizer surge nozzle at one end and to the hot leg surge line nozzle at the other end. The surge lines contain nine welds. A sample of these surge line welds is currently examined every ten years in accordance with the requirements of the ASME Section XI, Subsection IWB. Surge line welds selected for the inservice examinations, by nature of their size, require a volumetric examination. A number of the welds have been examined ultrasonically during the first three inservice examination intervals at Turkey Point, and a larger sample of welds is proposed to be examined ultrasonically during subsequent ten-year intervals. The increased sample of pressurizer surge line welds is based on the risk informed inservice inspection (RI-ISI) methodology. A request to revise the Unit 3 ASME Section XI, Subsection IWB inspection scope for Class 1 piping to RI-ISI has been submitted to the NRC [Reference 4.3-11]. Based on expert panel input into the Turkey Point Unit 3 RI-ISI proposal, which included consideration of environmental effects, the examination scope includes all of the pressurizer surge line welds. A similar change is proposed for Unit 4 and will be submitted to the NRC after approval of the Unit 3 submittal.

Pressurizer surge line examinations performed to date for Turkey Point Unit 3 include three surge line welds that have each been ultrasonically examined twice. For Turkey Point Unit 4, one weld has been ultrasonically examined three times, two welds have been ultrasonically examined twice, and an additional weld has been ultrasonically examined once. No reportable indications have been found.

Note that upon approval of the proposed RI-ISI programs for Turkey Point Units 3 and 4, all pressurizer surge line welds will be inspected during the fourth ISI interval and prior to the license renewal period. The results of these inspections will be utilized to assess the current 10-year inspection interval for continued use throughout the period of extended operation. As such, the potential effects of the reactor water environment have been evaluated for the extended period of operation as required by the resolution of GSI-190. The proposed RI-ISI enhancements to the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provide reasonable assurance that potential environmental effects of fatigue will be managed such that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the extended period of operation.

4.3.6 REFERENCES

4.3-1 Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," U. S. Nuclear Regulatory Commission.

4.3-2 Memorandum, Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to William D. Travers, Executive Director of Operations – “Closeout of Generic Safety Issue 190, Fatigue Evaluation of Metal Components for 60 Year Plant Life," U. S. Nuclear Regulatory Commission, December 26, 1999.

4.3-3 EPRI Report No. TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," Electric Power Research Institute, January 1998.

4.3-4 EPRI Report No. TR-110043, "Evaluation of Environmental Fatigue Effects for a Westinghouse Nuclear Power Plant," Electric Power Research Institute, April 1998.

4.3-5 EPRI Report No. TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," Electric Power Research Institute, April 1998.

4.3-6 EPRI Report No. TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," Electric Power Research Institute, May 1998.

4.3-7 NUREG/CR-6260 (INEL-95/0045), "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," U. S. Nuclear Regulatory Commission, March 1995.

4.3-8 NUREG/CR-5999 (ANL-93/3), "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments," U. S. Nuclear Regulatory Commission, August 1993.

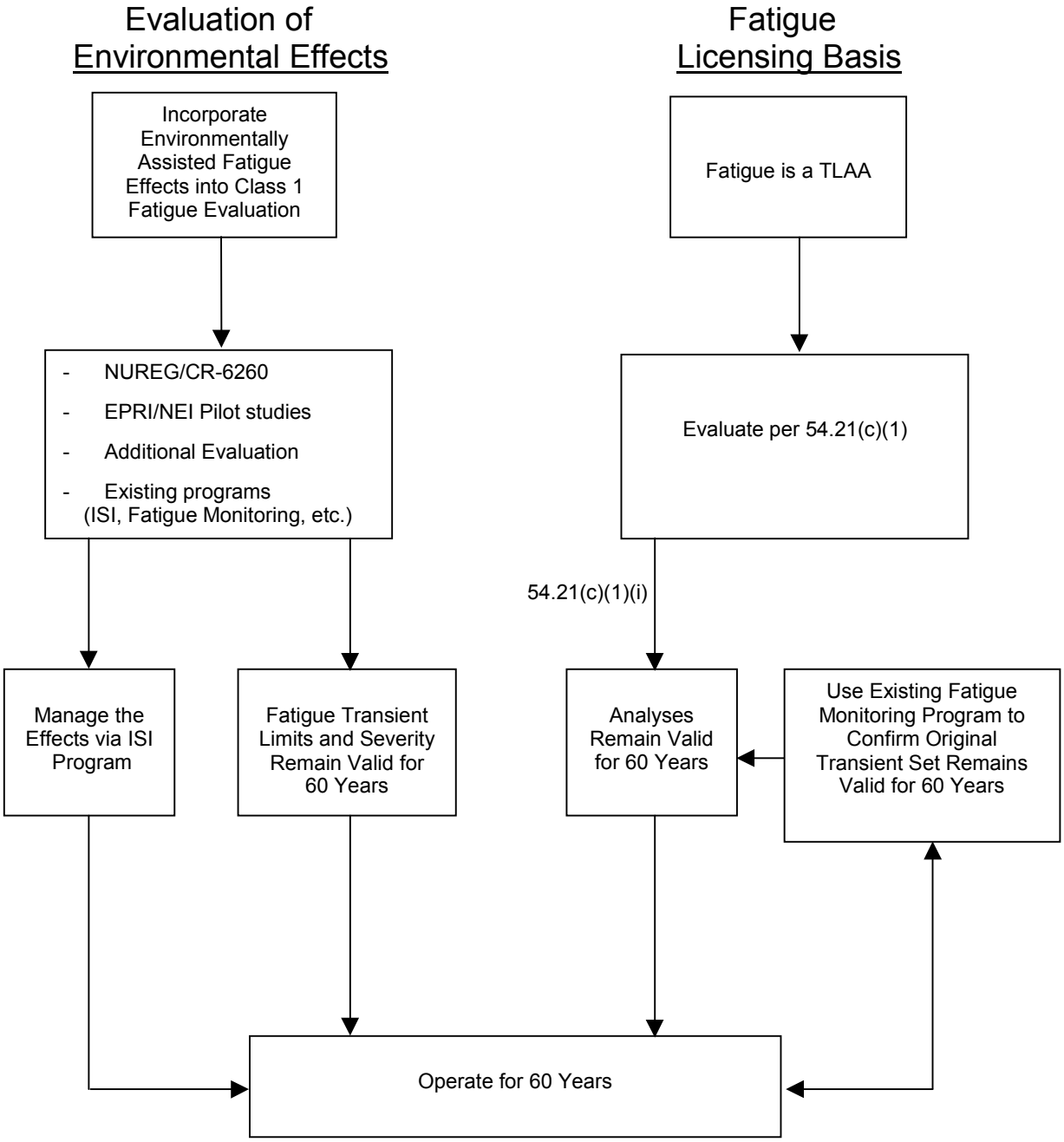
4.3-9 NUREG/CR-6583 (ANL-97/18), “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” U. S. Nuclear Regulatory Commission, March 1998.

4.3-10 NUREG/CR-5704 (ANL-98/31), “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,” U. S. Nuclear Regulatory Commission, April 1999.

4.3-11 FPL Letter, L-2000-010 to U. S. Nuclear Regulatory Commission, Turkey Point Unit 3, Docket No. 50-250, “Risk Informed Inservice Inspection Program," January 19, 2000.

FIGURE 4.3-1

GSI-190 EVALUATION PROCESS



4.4 ENVIRONMENTAL QUALIFICATION

The thermal, radiation, and wear cycle aging analyses of plant electrical and I&C components required to meet 10 CFR 50.49 have been identified as time-limited aging analyses for Turkey Point Units 3 and 4.

The Nuclear Regulatory Commission has established nuclear station environmental qualification requirements in 10 CFR 50, Appendix A, and in 10 CFR 50.49. The requirements in 10 CFR 50.49 specify that an environmental qualification program be established to demonstrate that certain electrical and I&C components located in "harsh" plant environments (i.e., those areas of the plant that could be subject to the harsh environment effects of a loss-of-coolant accident, high energy line break, or post loss-of-coolant accident radiation) are qualified to perform their safety function in those harsh environments after the effects of in-service aging. Further, 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

All operating plants must meet the requirements of 10 CFR 50.49 for certain electrical and I&C 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 environmental conditions. The requirements in 10 CFR 50.49(e)(5) contain provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e) also requires component replacement or refurbishment prior to the end of designated life unless additional life is established through ongoing qualification. 10 CFR 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions. The requirements in 10 CFR 50.49 (k) and (l) permit different criteria to apply based on plant and component vintage. Supplemental environmental qualification regulatory guidance for compliance with these different qualification criteria is provided in the DOR Guidelines, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" [Reference 4.4-1], NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment" [Reference 4.4-2], and Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants" [Reference 4.4-3]. Compliance with 10 CFR 50.49 provides

evidence that the component will perform its intended functions during accident conditions after experiencing the effects of in-service aging.

The Turkey Point Environmental Qualification Program, which complies with all applicable regulations, includes three main elements: identifying applicable equipment and environmental requirements, establishing the qualification, and maintaining (or preserving) that qualification.

The first element involves establishment and control of the Environmental Qualification List of components and the service conditions for the harsh environment plant areas. The second element involves establishment and control of the components' environmental qualification documentation, including vendor test reports, vendor correspondence, calculations, evaluations of component tested conditions to plant required conditions, and determinations of configuration and maintenance requirements. The third element includes preventive maintenance processes (for replacing parts and components at specified intervals), design control processes (ensuring changes to the plant are evaluated for impact to the Environmental Qualification Program), procurement processes (ensuring new and replacement components are purchased to applicable environmental qualification requirements), and corrective action processes in accordance with the FPL Quality Assurance Program.

Components included in the Turkey Point Environmental Qualification Program have been evaluated to determine if existing environmental qualification aging analyses remain valid for the period of extended operation. Qualification for the license renewal period will be treated the same as for components currently qualified at Turkey Point for 40 years or less.

The Turkey Point Environmental Qualification Program manages component thermal, radiation, and wear cycle aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, environmentally qualified components must be refurbished, replaced, or their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for environmentally qualified components that specify a qualification of at least 40 years are considered time-limited aging analyses for license renewal.

4.4.1 ELECTRICAL AND I&C COMPONENT ENVIRONMENTAL QUALIFICATION ANALYSES

Age-related service conditions that are applicable to environmentally qualified components (i.e., 60 years of exposure versus 40 years) were evaluated for the period of extended operation to verify that the current environmental qualification analyses were bounding. Temperature and radiation values assumed for service conditions in the environmental qualification analyses are the maximum design operating values for Turkey Point. The following paragraphs describe the thermal, radiation, and wear cycle aging effects that were evaluated.

THERMAL CONSIDERATIONS - The component qualification temperatures were calculated for 60 years using the Arrhenius method, as described in EPRI NP-1558, "A Review of Equipment Aging Theory and Technology" [Reference 4.4-4]. The Turkey Point Environmental Qualification Program temperature for inside Containment is 50°C and for areas outside Containment is up to 40°C. For conservatism, a temperature rise of 10°C was added to the maximum design operating temperature for continuous duty power cables to account for ohmic heating. This results in maximum design operating temperatures of 60ºC inside Containment and 50ºC outside Containment for these power cables and penetrations. If the component qualification temperature bounded the maximum design operating temperatures, then no additional evaluation was required.

In connection with plant modifications, in 1991, some new environmentally qualified components that will not experience 60 years of thermal aging by the end of the license renewal period were installed at Turkey Point. In these cases, credit may be taken for less than 60 years of aging. This applies to two environmental qualification analyses, Patel/EGS conformal splices and Patel/EGS Grayboot connectors, described in Subsections 4.4.1.38 and 4.4.1.39, respectively.

RADIATION CONSIDERATIONS - The Turkey Point Environmental Qualification Program has established bounding radiation dose qualification values for all environmentally qualified components. These bounding radiation dose values were determined by component vendors through testing. To verify that the bounding radiation values are acceptable for the period of extended operation, 60-year integrated dose values were determined and then compared to the bounding values. The total integrated dose for the 60-year period is determined by adding the established accident dose to the 60-year normal operating dose for the component.

WEAR CYCLE CONSIDERATIONS - The wear cycle aging effect is only applicable to ASCO solenoid valves for Turkey Point (see Subsection 4.4.1.3). ASCO has established a wear cycle limit of 40,000 cycles for these valves. The projected cycles for 60 years for these valves were determined, and then compared to the limit provided by the vendor to establish acceptability for the period of extended operation.

The values for margin identified in Section 6.3.1.5 of IEEE 323-1974 were used as criteria in the Turkey Point Environmental Qualification Program. The only regular exception to the IEEE 323-1974 margins was for radiation. As identified in Item 1.4 of NUREG-0588, additional margin need not be added to the radiation parameters if the methods identified in Appendix D of NUREG-0588 are utilized. The methods used to determine the Turkey Point radiation parameters are consistent with the Appendix D methodology. Hence, the radiation margins required by Section 6.3.1.5 of IEEE 323-1974 are not necessary. Accordingly, margin is adequately addressed in the Turkey Point Environmental Qualification Program.

The following Subsections (4.4.1.1 through 4.4.1.39) provide a description for each of the environmental qualification analyses for the period of extended operation.

4.4.1.1 ANACONDA CABLES

Anaconda cables are installed in instrumentation, control, and power applications, both inside and outside the Containments at Turkey Point. Cable insulation for environmentally qualified Anaconda cables is either FR-EP or EP.

THERMAL ANALYSIS

The qualified life analysis for Anaconda instrumentation and control cables shows the cables are qualified for continuous operation for 60 years at a temperature of 67.9°C. Instrumentation and control cables are subject to minimal temperature rise with a resulting maximum design operating temperature of 50°C.

The qualified life analysis for Anaconda power cables shows the cables are qualified for continuous operation for 60 years at a temperature of 72.0°C. The power cables have a maximum design operating temperature of 60°C (includes 10°C of ohmic heating).

RADIATION ANALYSIS

The qualified life analysis for Anaconda power and instrumentation/control cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $5x10^7$ rads.

CONCLUSION

Anaconda instrumentation, control, and power cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.2 AIW CABLES

AIW cables are installed in outside Containment instrumentation circuits at Turkey Point. This includes all outdoor areas. Cable insulation for environmentally qualified AIW cables is PE.

THERMAL ANALYSIS

The qualified life analysis for AIW cables shows the cables are qualified for continuous operation for 60 years at a temperature of 53.8°C. AIW instrumentation cables are subject to minimal temperature rise with a resulting maximum design operating temperature of 40°C.

RADIATION ANALYSIS

The qualified life analysis for AIW cables shows the cables are qualified for 7.5×10^7 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 7.56×10^6 rads.

CONCLUSION

AIW instrumentation cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.3 ASCO SOLENOID VALVES

Normally de-energized ASCO solenoid valves are installed both inside and outside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for normally de-energized ASCO solenoid valves shows the solenoid valves are qualified for greater than 60 years at ambient temperatures of 50°C inside Containment and 40°C outside Containment. Normally de-energized ASCO solenoid valves have maximum design operating temperatures of 50°C inside Containment and 40°C outside Containment.

RADIATION ANALYSIS

The qualified life analysis for normally de-energized ASCO solenoid valves shows the solenoid valves are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

WEAR/CYCLES ANALYSIS

The qualified life analysis for normally de-energized ASCO solenoid valves shows the solenoid valves are qualified for 40,000 cycles. The maximum projected usage is less than 1000 cycles.

CONCLUSION

Normally de-energized ASCO solenoid valves are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.4 BRAND REX COAXIAL CABLES

Brand Rex coaxial cables are installed for instrumentation applications inside the Containments at Turkey Point. Cable insulation for environmentally qualified Brand Rex coaxial cables is XLPE.

THERMAL ANALYSIS

The qualified life analysis for Brand Rex coaxial cables shows the cables are qualified for continuous operation for 60 years at a temperature of 53.8°C. The cables have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Brand Rex coaxial cables shows the cables are qualified for 1.8×10^8 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 4×10^7 rads.

CONCLUSION

Brand Rex coaxial cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.5 BRAND REX INSTRUMENT CABLES

Brand Rex 600 volt instrumentation cables are installed both inside and outside the Containments at Turkey Point. Cable insulation for environmentally qualified Brand Rex 600 volt instrumentation cables is XLPE.

THERMAL ANALYSIS

The qualified life analysis for Brand Rex instrument cables shows the cables are qualified for continuous operation for greater than 60 years at a temperature of 50.0°C. Instrumentation and control cables are subject to minimal temperature rise with a resulting maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Brand Rex instrument cables shows the cables are qualified for 1.8×10^8 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 4×10^7 rads.

CONCLUSION

Brand Rex instrument cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.6 CONAX CONDUIT SEALS

Conax conduit seal assemblies are installed inside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Conax conduit seal assemblies shows the seal assemblies are qualified for continuous operation for greater than 60 years at a temperature of 50°C. The seal assemblies have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Conax conduit seal assemblies shows the seal assemblies are qualified for 2.25×10^8 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 4×10^7 rads.

CONCLUSION

Conax conduit seal assemblies are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.7 CONAX PENETRATIONS

Conax electrical penetrations are installed inside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Conax power penetrations shows the penetrations are qualified for continuous operation for 60 years at a temperature of 93.7°C. The qualified life analysis for Conax instrumentation and control penetrations shows the penetrations are qualified for continuous operation for greater than 60 years at a temperature of 50°C. The Conax power penetrations have a maximum design operating temperature of 60°C (includes 10°C of ohmic heating). The Conax instrumentation and control penetrations have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Conax electrical penetrations shows the electrical penetrations are qualified for 1.2×10^8 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 4×10^7 rads.

CONCLUSION

Conax electrical penetrations are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.8 CONAX UNITIZED RESISTANCE TEMPERATURE DETECTORS

Conax Unitized resistance temperature detectors are installed inside the Containments at Turkey Point for the containment atmosphere temperature instrument loops.

THERMAL ANALYSIS

The qualified life analysis for Conax Unitized resistance temperature detectors shows the RTDs are qualified for continuous operation for greater than 60 years at a temperature of 50°C. The resistance temperature detectors have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Conax Unitized resistance temperature detectors shows the resistance temperature detectors are qualified for 2.2×10^8 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 4×10^7 rads.

CONCLUSION

Conax Unitized resistance temperature detectors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.9 CHAMPLAIN CABLES

Champlain cables are installed outside the Containments at Turkey Point. Cable insulation for environmentally qualified Champlain cables is XLPE.

THERMAL ANALYSIS

The qualified life analysis for Champlain cables shows the cables are qualified for continuous operation for 60 years at a temperature of 85°C. The cables have a maximum design operating temperature of 40°C.

RADIATION ANALYSIS

The qualified life analysis for Champlain cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $7.56x10^6$ rads.

CONCLUSION

Champlain cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.10 CROUSE HINDS PENETRATIONS

Crouse Hinds penetrations are installed inside the Containments for containment electrical penetrations at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Crouse Hinds penetrations shows the penetrations are qualified for continuous operation for 60 years at temperatures of 50°C for instrumentation and control penetrations and 60°C for power penetrations. The penetrations have maximum design operating temperatures of 50°C for instrumentation and control penetrations and 60°C (includes 10°C of ohmic heating) for power penetrations.

RADIATION ANALYSIS

The qualified life analysis for Crouse Hinds penetrations shows the penetrations are qualified for $1x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

Crouse Hinds penetrations are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.11 GENERAL ATOMIC RADIATION MONITORS

General Atomic radiation monitors are installed inside the Containments at Turkey Point.

THERMAL ANALYSIS

The General Atomic radiation monitor detectors are composed entirely of inorganic materials and not susceptible to thermal degradation.

RADIATION ANALYSIS

The General Atomic radiation monitor detectors are composed of inorganic components. The only component susceptible to radiation aging is the ionization chamber. The ionization chambers are qualified for $1x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose for the ionization chambers is $4x10^7$ rads.

CONCLUSION

General Atomic radiation monitors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.12 GENERAL CABLES

General cables are installed for 5kV power cables outside the Containments at Turkey Point. Cable insulation for environmentally qualified General cables is butyl rubber.

THERMAL ANALYSIS

The qualified life analysis for General cables shows the cables are qualified for continuous operation for 60 years at a temperature of 66.6°C. The cables have a maximum design operating temperature of 50°C (includes 10°C of ohmic heating).

RADIATION ANALYSIS

The qualified life analysis for General cables shows the cables are qualified for $3x10^6$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $2.46x10^6$ rads.

CONCLUSION

General cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.13 GENERAL ELECTRIC CABLES

General Electric cables are installed as instrumentation cables and jumper wires both inside and outside the Containments at Turkey Point. Cable insulation for environmentally qualified General Electric cables is XLPE.

THERMAL ANALYSIS

The qualified life analysis for General Electric instrumentation cables shows the cables are qualified for continuous operation for 60 years at a temperature of 70°C. The cables have a maximum design operating temperature of 50°C.

The qualified life analysis for General Electric jumper wires shows the jumper wires are qualified for 40 years at a temperature of 64.7°C, plus 20 years at a temperature of 75°C (due to space heaters that were energized during the first twenty years of plant operation). The jumper wires have a maximum design operating temperature of 60°C (includes 10°C for ohmic heating). An additional 15°C is also added for the first 20 years of operation to account for space heaters being energized.

RADIATION ANALYSIS

The qualified life analysis for General Electric instrumentation cables and jumper wires shows they are qualified for $5x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $5x10^7$ rads.

CONCLUSION

General Electric cables, installed as instrumentation cables and jumper wires, are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.14 GENERAL ELECTRIC TERMINAL BLOCKS

General Electric terminal blocks are installed outside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for General Electric terminal blocks shows the terminal blocks are qualified for continuous operation for greater than 60 years at a temperature of 40°C. The terminal blocks have a maximum design operating temperature of 40°C.

RADIATION ANALYSIS

The qualified life analysis for General Electric terminal blocks shows the terminal blocks are qualified for $2x10^8$ rads (EB-5) and $1.2x10^7$rads (EB-25). The maximum projected post accident plus 60-year normal operation radiation dose is $7.5x10^6$ rads.

CONCLUSION

General Electric terminal blocks are qualified for the period of extended operation based on the determination that the analyses remain remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.15 JOY EMERGENCY CONTAINMENT COOLER AND EMERGENCY CONTAINMENT FILTRATION FAN MOTORS

Joy emergency containment cooler and emergency containment filtration fan motors are installed inside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Joy emergency containment cooler and emergency containment filtration fan motors shows the motors are qualified for 0.25 years of operation at 115°C and 59.75 years of standby operation at 101.6°C. The Joy emergency containment cooler and emergency containment filtration fan motors have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Joy emergency containment cooler and emergency containment filtration fan motors shows the motors are qualified for $3x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

Joy emergency containment cooler and emergency containment filtration fan motors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.16 LIMITORQUE VALVE OPERATORS WITH RELIANCE MOTORS FOR USE INSIDE CONTAINMENT

Limitorque valve operators with Reliance motors are installed inside the Containments for various motor-operated valves at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Limitorque valve operators with Reliance motors shows the actuators are qualified for greater than 60 years at an ambient temperature of 55°C. A temperature of 5°C was added to the maximum design operating temperature of 50°C inside Containment to account for the space heaters being energized during the first 20 years of plant life. The space heaters were disconnected at that point. The actuators have a maximum design operating temperature of 55°C (includes an adjustment of 5°C as described above).

RADIATION ANALYSIS

The qualified life analysis for Limitorque valve operators with Reliance motors shows the actuators are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

Limitorque valve operators with Reliance motors for use inside Containment are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.17 LIMITORQUE VALVE OPERATORS WITH RELIANCE MOTORS WITH CLASS H(RH) INSULATION FOR USE INSIDE CONTAINMENT

Limitorque valve operators with Reliance motors with Class H(RH) insulation are installed inside the Containments for various motor-operated valves at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Limitorque valve operators with Reliance motors with Class H(RH) insulation shows the actuators are qualified for greater than 60 years at an ambient temperature of 55°C. A temperature of 5°C was added to the maximum design operating temperature of 50°C inside containment to account for the space heaters being energized during the first 20 years of plant life. The space heaters were disconnected at that point. The actuators have a maximum design operating temperature of 55°C (includes an adjustment of 5°C as described above).

RADIATION ANALYSIS

The qualified life analysis for Limitorque valve operators with Reliance motors with Class H(RH) insulation shows the actuators are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

Limitorque valve operators with Reliance motors with Class H(RH) insulation for use inside Containment are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.18 LIMITORQUE VALVE OPERATORS WITH RELIANCE MOTORS FOR USE OUTSIDE CONTAINMENT

Limitorque valve operators with Reliance motors are installed outside the Containments for various motor-operated valves at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Limitorque valve operators with Reliance motors shows the actuators are qualified for greater than 60 years at an ambient temperature of 40°C. The actuators have a maximum design operating temperature of 40°C.

RADIATION ANALYSIS

The qualified life analysis for Limitorque valve operators with Reliance motors shows the actuators are qualified for $2x10^7$ rads. The maximum projected post accident plus 60 year normal operation radiation dose is $2.0x10^6$ rads.

CONCLUSION

Limitorque valve operators with Reliance motors for use outside Containment are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.19 LIMITORQUE VALVE OPERATORS WITH PEERLESS MOTORS FOR USE OUTSIDE CONTAINMENT

Limitorque valve operators with Peerless motors are installed outside the Containments for various motor-operated valves at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Limitorque valve operators with Peerless motors shows the actuators are qualified for greater than 60 years at an ambient temperature of 40°C. The actuators have a maximum design operating temperature of 40°C.

RADIATION ANALYSIS

The qualified life analysis for Limitorque valve operators with Peerless motors shows the actuators are qualified for $1x10^7$ rads. These actuators are located in a mild radiation environment at Turkey Point, resulting in a maximum projected post accident plus 60 year normal operation radiation dose of $1x10^5$ rads.

CONCLUSION

Limitorque valve operators with Peerless motors for use outside Containment are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.20 OKONITE CABLES

Okonite X-Olene/Okoseal 600V power and control cables are installed inside and outside the Containments at Turkey Point. Okonite Okonex/Okoseal 600V power cables are installed outside the Containments at Turkey Point. Okonite Okolene/Okoseal 600V instrumentation and control cables are installed outside the Containments at Turkey Point. Cable insulation for environmentally qualified Okonite X-Olene/Okoseal 600V power and control cables is XLPE. Cable insulation for environmentally qualified Okonite Okonex/Okoseal 600V power cables is butyl rubber. Cable insulation for environmentally qualified Okonite Okolene/Okoseal 600V instrumentation and control cables is PE.

THERMAL ANALYSIS

The qualified life analysis for Okonite X-Olene/Okoseal 600V power and control cables, Okonite Okonex/Okoseal 600V power cables, and Okonite Okolene/Okoseal 600V instrumentation and control cables shows the cables are qualified for continuous operation for greater than 60 years at temperatures of 60°C (power) and 50°C (I&C) for inside Containment cables and 50°C (power) and 40°C (I&C) for outside Containment cables. The cables have maximum design operating temperatures of 60°C (includes 10°C of ohmic heating) for inside Containment cables and 50°C (includes 10°C of ohmic heating) for outside Containment cables.

RADIATION ANALYSIS

The qualified life analysis for Okonite X-Olene/Okoseal 600V power and control cable shows the cable is qualified for $1x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose for Okonite X-Olene/Okoseal 600V power and control cable at Turkey Point is $4x10^7$ rads.

The qualified life analysis for Okonite Okonex/Okoseal 600V power cable shows the cable is qualified for $5x10^6$ rads. The maximum projected post accident plus 60-year normal operation radiation dose for Okonite Okonex/Okoseal 600V power cable is $2.22x10^6$ rads.

The qualified life analysis for Okonite Okolene/Okoseal 600V instrumentation and control cable shows the cable is qualified for $1x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose for Okonite Okolene/Okoseal 600V instrumentation and control cable is $7.6x10^6$ rads.

CONCLUSION

Okonite cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.21 RAYCHEM HEAT SHRINK SLEEVING

Raychem heat shrink sleeving is installed both inside and outside the Containments for insulation of electrical connections at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for WCSF-N, WCSF-050-N, NMCK, and RNF-100 sleeving materials shows the sleeving is qualified for continuous operation for greater than 60 years at a temperature of 85°C. The WCSF-N and WCSF-050-N sleeving materials have a maximum design operating temperature of 53.2°C, NMCK and RNF-100 sleeving materials have a maximum design operating temperature of 50°C.

The qualified life analysis for NHVT sleeving material shows the sleeving is qualified for 7.5 years of operation at a temperature of 50°C, and at least 148 additional years at 40°C. This exceeds the required 7.5 years at 50°C and 52.5 years at 40°C for residual heat removal pump motor operation.

The qualified life analysis for NMCK-8(L) sleeving material shows the sleeving is qualified for continuous operation for 60 years at a temperature of 78°C. The NMCK-8(L) sleeving material has a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for WCSF-N sleeving material shows the sleeving is qualified for $2x10^8$ to $2.9x10^8$ rads for bolted and butt connections. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

The qualified life analysis for NMCK sleeving material shows the sleeving is qualified for $2.9x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

The qualified life analysis for RNF-100, WCSF-050-N, and NHVT sleeving material shows the sleeving is qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose for RNF-100 is $7.56x10^6$ rads, for WCSF-050-N is $4x10^7$ rads, and for NHVT is $2.4x10^6$ rads.

The qualified life analysis for NMCK-8(L) sleeving material shows the sleeving is qualified for $5x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $2.4x10^6$ rads.

CONCLUSION

Raychem heat shrink sleeving is qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.22 RAYCHEM CABLES

Raychem cables are installed inside the Containments at Turkey Point in 600V control circuits. Cable insulation for environmentally qualified Raychem cables is XLPE.

THERMAL ANALYSIS

The qualified life analysis for Raychem cables shows the cables are qualified for continuous operation for greater than 60 years at a temperature of 60°C. The cables have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Raychem cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

Raychem cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.23 MACWORTH REES PUSHBUTTON STATIONS

MacWorth Rees pushbutton stations are installed outside the Containments for control of various motors at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for MacWorth Rees pushbutton stations shows the pushbutton stations are qualified for continuous operation for greater than 60 years at a temperature of 40°C. The pushbutton stations have a maximum design operating temperature of 40°C.

RADIATION ANALYSIS

The qualified life analysis for MacWorth Rees pushbutton stations shows the pushbutton stations are qualified for $1x10^6$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $7.51x10^5$ rads.

CONCLUSION

MacWorth Rees pushbutton stations are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.24 ROCKBESTOS CABLES

Rockbestos cables are installed as control, instrumentation, and power cables both inside and outside the Containments at Turkey Point. Cable insulation for environmentally qualified Rockbestos cables is XLPE.

THERMAL ANALYSIS

The qualified life analysis for Rockbestos instrumentation and control cables shows the cables are qualified for continuous operation for 60 years at a temperature of 54°C. The instrumentation and control cables have a maximum design operating temperature of 50°C.

The qualified life analysis for Rockbestos power cables shows the cables are qualified for continuous operation for 60 years at a temperature of 87°C. The cables have a maximum design operating temperature of 60°C (includes 10°C of ohmic heating).

RADIATION ANALYSIS

The qualified life analysis for Rockbestos control, instrumentation, and power cables shows the cable and wire are qualified for 1.84×10^8 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 5×10^7 rads.

CONCLUSION

Rockbestos control, instrumentation, and power cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.25 SAMUEL MOORE CABLES

Samuel Moore cables are installed as 600V instrumentation cables inside the Containments at Turkey Point. Cable insulation for environmentally qualified Samuel Moore cables is EPDM.

THERMAL ANALYSIS

The qualified life analysis for Samuel Moore instrumentation cables shows the cables are qualified for continuous operation for 60 years at a temperature of 49.7°C. Although this temperature is below the inside Containment environmental qualification temperature of 50°C, the temperature of 49.7°C is above the Technical Specification Containment temperature limit of 48.9°C. The integrated maximum temperature profile for inside Containment over Turkey Point's history has been and will be below the Technical Specification limit.

RADIATION ANALYSIS

The qualified life analysis for Samuel Moore cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

The analyses associated with the environmental qualification of Samuel Moore cables have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.26 3M INSULATING TAPE AND SCOTCHFIL

3M Insulating tape and Scotchfil are installed outside the Containments for splicing and terminations at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for 3M Insulating tapes and Scotchfil shows the insulating materials are qualified for continuous operation for greater than 60 years at a temperature of 50°C. The 3M Insulating tape and Scotchfil have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for 3M Insulating tapes shows the tapes are qualified for $2x10^8$ rads (Type 130C/33+), $1x10^8$ rads (Types 23, 130C, and 70), and $5x10^7$ rads (Type 33+). The qualified life analysis for 3M Scotchfil shows that Scotchfil is qualified for $6.02x10^6$ rads. The maximum projected post accident plus 60-year normal operation radiation dose for both 3M Insulating tape and Scotchfil is $2.4x10^6$ rads.

CONCLUSION

3M Insulating tape and Scotchfil are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.27 WESTINGHOUSE RESIDUAL HEAT REMOVAL PUMP MOTORS

Westinghouse residual heat removal pump motors are installed outside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Westinghouse residual heat removal pump motors shows the motors are qualified for 7.2 years of operation at 120°C and 53.3 years of standby operation at 94.6°C. The Westinghouse residual heat removal pump motors have maximum design temperatures of 120°C when operating and 93°C when in standby.

RADIATION ANALYSIS

The qualified life analysis for the Westinghouse residual heat removal pump motors shows the motors are qualified for $5x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $2.2x10^5$ rads.

CONCLUSION

Westinghouse residual heat removal pump motors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.28 WESTINGHOUSE CONTAINMENT SPRAY PUMP MOTORS

Westinghouse containment spray pump motors are installed outside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Westinghouse containment spray pump motors shows the motors are qualified for 0.3 years of operation at 130°C and 59.7 years of standby operation at 97.1°C. The Westinghouse containment spray pump motors have maximum design temperatures of 130°C when operating and 93°C when in standby.

RADIATION ANALYSIS

The qualified life analysis for the Westinghouse containment spray pump motors shows the motors are qualified for $5x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $8.1x10^5$ rads.

CONCLUSION

Westinghouse containment spray pump motors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.29 WESTINGHOUSE SAFETY INJECTION PUMP MOTORS

Westinghouse safety injection pump motors are installed outside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Westinghouse safety injection pump motors shows the motors are qualified for 0.3 years of operation at 130°C and 59.7 years of standby operation at 97.1°C. The Westinghouse safety injection pump motors have maximum design temperatures of 130°C when operating and 93°C when in standby.

RADIATION ANALYSIS

The qualified life analysis for the Westinghouse safety injection pump motors shows the motors are qualified for $5x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $8.1x10^5$ rads.

CONCLUSION

Westinghouse safety injection pump motors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.30 COMBUSTION ENGINEERING MINERAL INSULATED CABLES AND CONNECTORS

Combustion Engineering Mineral Insulated cables with ERD twin-pin and multi-pin connectors, Litton connectors with Grafoil seals, G&H connectors, and Litton "B" connectors are installed inside the Containments at Turkey Point for the core exit thermocouples and heated junction thermocouples.

THERMAL ANALYSIS

The qualified life analysis for Combustion Engineering Mineral Insulated cables with ERD twin-pin and multi-pin connectors, Litton connectors with Grafoil seals, G&H connectors, and Litton "B" connectors shows the cables are qualified for continuous operation for greater than 60 years at a temperature of 55°C. The cables have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Combustion Engineering Mineral Insulated cables with ERD twin-pin and multi-pin connectors, Litton connectors with Grafoil seals, and G&H connectors shows the cables are qualified for 2.2×10^8 rads. The qualified life analysis for Combustion Engineering Mineral Insulated cables with Litton "B" connectors shows the cables are qualified for 2.1×10^8 rads. The maximum projected post accident plus 60-year normal operation radiation dose is 4×10^7 rads.

CONCLUSION

Combustion Engineering Mineral Insulated cables with ERD twin-pin and multi-pin connectors, Litton connectors with Grafoil seals, G&H connectors, and Litton "B" connectors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.31 KERITE HTK/FR CABLES

Kerite HTK/FR cables are installed in 600V power applications inside the Containments at Turkey Point. Insulation and jackets for environmentally qualified Kerite HTK/FR cables are HTK and FR, respectively.

THERMAL ANALYSIS

The qualified life analysis for Kerite HTK/FR cables shows the cables are qualified for continuous operation for 60 years at a temperature of 80.1°C. The cables have a maximum design operating temperature of 60°C (includes 10°C of ohmic heating).

RADIATION ANALYSIS

The qualified life analysis for Kerite HTK/FR cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $5x10^7$ rads.

CONCLUSION

Kerite HTK/FR cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.32 KERITE FR2/FR CABLES

Kerite FR2/FR cables are installed both inside and outside the Containments at Turkey Point in 600V instrumentation, control, and power applications. Insulation and jackets for environmentally qualified Kerite FR2/FR cables are FR2 and FR, respectively.

THERMAL ANALYSIS

The qualified life analysis for Kerite FR2/FR cables shows the cables are qualified for continuous operation for 60 years at a temperature of 80°C. The cables have a maximum design operating temperature of 60°C (includes 10°C of ohmic heating).

RADIATION ANALYSIS

The qualified life analysis for Kerite FR2/FR cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $5x10^7$ rads.

CONCLUSION

Kerite FR2/FR cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.33 KERITE FR/FR CABLES

Kerite FR/FR cables are installed both inside and outside the Containments at Turkey Point in 600V control applications. Insulation and jackets for environmentally qualified Kerite FR/FR cables are FR.

THERMAL ANALYSIS

The qualified life analysis for Kerite FR/FR cables shows the cables are qualified for continuous operation for 60 years at a temperature of 80°C. The cables have a maximum design operating temperature of 50°C.

RADIATION ANALYSIS

The qualified life analysis for Kerite FR/FR cables shows the cables are qualified for $5x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $5x10^7$ rads.

CONCLUSION

Kerite FR/FR cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.34 KERITE HTK/FR POWER CABLES

Kerite HTK/FR power cables are installed outside the Containments at Turkey Point in 8000V power applications. Insulation and jackets for environmentally qualified Kerite HTK/FR power cables are HTK and FR, respectively.

THERMAL ANALYSIS

The qualified life analysis for Kerite HTK/FR power cables shows the cables are qualified for continuous operation for 60 years at a temperature of 80°C. The cables have a maximum design operating temperature of 50°C (includes 10°C of ohmic heating).

RADIATION ANALYSIS

The qualified life analysis for Kerite HTK/FR power cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $2.9x10^5$ rads.

CONCLUSION

Kerite HTK/FR power cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.35 TELEDYNE THERMATICS CABLES

Teledyne Thermatics cables are installed both inside and outside the Containments at Turkey Point. Insulation for environmentally qualified Teledyne Thermatics cables is Tefzel 280.

THERMAL ANALYSIS

The qualified life analysis for Teledyne Thermatics cables shows the cables are qualified for continuous operation for 60 years at a temperature of 65.0°C. The cables have a maximum design operating temperature of 61.2°C (includes 11.2ºC for the space heaters inside the Limitorque operators being energized during the first 20 years of operation).

RADIATION ANALYSIS

The qualified life analysis for the Teledyne Thermatics cables shows the cables are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

Teledyne Thermatics cables are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.36 WEED RESISTANCE TEMPERATURE DETECTORS

Weed resistance temperature detectors are installed both inside and outside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Weed resistance temperature detectors shows the resistance temperature detectors are qualified for continuous operation for greater than 60 years at a temperature of 52.7°C. The resistance temperature detectors have a maximum design operating temperature of 51.6°C (includes 1.6ºC for effects of the high process temperature being measured).

RADIATION ANALYSIS

The qualified life analysis for Weed resistance temperature detectors shows the resistance temperature detectors are qualified for $3x10^8$ rads. The maximum projected post accident plus 60 year normal operation radiation dose is $1.06x10^8$ rads inside Containment, and $7.5x10^6$ rads outside Containment.

CONCLUSION

Weed resistance temperature detectors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.37 AMERACE NQB TERMINAL BLOCKS

Amerace NQB terminal blocks are installed outside the Containments at Turkey Point.

THERMAL ANALYSIS

The qualified life analysis for Amerace NQB terminal blocks shows the terminal blocks are qualified for continuous operation for 60 years at a temperature of 44°C. The terminal blocks have a maximum design operating temperature of 40°C.

RADIATION ANALYSIS

The qualified life analysis for Amerace NQB terminal blocks shows the terminal blocks are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $7.56x10^6$ rads.

CONCLUSION

Amerace NQB terminal blocks are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.38 PATEL/EGS CONFORMAL SPLICES

Patel/EGS conformal splices are installed outside the Containments at Turkey Point.

THERMAL ANALYSIS

The Patel/EGS conformal splices were first installed at Turkey Point in 1991. The qualified life analysis for Patel/EGS conformal splices shows the conformal splices are qualified for continuous operation for 42 years (1991-2033) at a temperature of 90°C. The conformal splices have a maximum design operating temperature of 50°C (includes 10°C of ohmic heating).

RADIATION ANALYSIS

The qualified life analysis for Patel/EGS conformal splices shows the conformal splices are qualified for $2x10^7$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $7.5x10^6$ rads.

CONCLUSION

The Patel/EGS conformal splices are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.1.39 PATEL/EGS GRAYBOOT CONNECTORS

Patel/EGS Grayboot connectors are installed both inside and outside the Containments at Turkey Point.

THERMAL ANALYSIS

The Patel/EGS Grayboot connectors were first installed at Turkey Point in 1991. The qualified life analysis for Patel/EGS Grayboot connectors shows the connectors are qualified for continuous operation for 42 years (1991-2033) at a temperature of 55.7°C. The connectors have a maximum design operating temperature of 55.6°C (includes 5.6°C of ohmic heating based on the specific Grayboot connector applications).

RADIATION ANALYSIS

The qualified life analysis for Patel/EGS Grayboot connectors shows the connectors are qualified for $2x10^8$ rads. The maximum projected post accident plus 60-year normal operation radiation dose is $4x10^7$ rads.

CONCLUSION

The Patel/EGS Grayboot connectors are qualified for the period of extended operation based on the determination that the existing analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.4.2 GSI-168, ENVIRONMENTAL QUALIFICATION OF ELECTRICAL COMPONENTS

NRC guidance for addressing GSI-168 for license renewal is contained in the June 2, 1998, NRC letter to NEI [Reference 4.4-5]. In this letter, the NRC states, “With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the Statements of Consideration is to provide a technical rationale demonstrating that the current licensing basis for environmental qualification pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the Statements of Consideration also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time."

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for Turkey Point Units 3 and 4. The evaluations of these time-limited aging analyses are considered the technical rationale that the current licensing basis will be maintained during the period of extended operation. These evaluations are provided in Section 4.4 of the Turkey Point License Renewal Application. Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a renewal application at this time.

4.4.3 REFERENCES

4.4-1 DOR Guidelines, “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors,” U. S. Nuclear Regulatory Commission, June 1979.

4.4-2 NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment,” U. S. Nuclear Regulatory Commission, July 1981.

4.4-3 Regulatory Guide 1.89, Revision 1, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,” U. S. Nuclear Regulatory Commission, June 1984.

4.4-4 EPRI NP-1558, “A Review of Equipment Aging Theory and Technology,” Electric Power Research Institute, September 1980.

4.4-5 C. I. Grimes (NRC) letter to D. Walters (NEI), “Guidance on Addressing GSI 168 for License Renewal,” Project 690, June 2, 1998.

4.5 CONTAINMENT TENDON LOSS OF PRESTRESS

The Turkey Point Units 3 and 4 containment buildings are post-tensioned, reinforced concrete structures composed of vertical cylinder walls and a shallow dome, supported on a conventional reinforced concrete base slab. The cylinder walls are provided with vertical tendons and horizontal hoop tendons. The dome is provided with three groups of tendons oriented 120-degrees apart.

The prestress of containment tendons decreases over time as a result of seating of anchorage losses, elastic shortening of concrete, creep of concrete, shrinkage of concrete, relaxation of prestressing steel, and friction losses. At the time of initial licensing, the magnitude of the prestress losses throughout the life of the plant was predicted and the estimated final effective preload at the end of 40 years was calculated for each tendon type. The final effective preload was then compared with the minimum required preload to confirm the adequacy of the design.

New upper limit curves, lower limit curves, and trend lines of measured prestressing forces have been established for all tendons through the period of extended operation. The predicted final effective preload at the end of 60 years exceeds the minimum required preload for all containment tendons. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation.

The analyses associated with containment tendon loss of prestress have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

As a confirmatory program, the Containment structure post-tensioning system surveillance performed as a part of the ASME Section XI, Subsection IWL Inservice Inspection Program will continue to be performed in accordance with the requirements of Technical Specifications 4.6.1.6.1 and 4.6.1.6.2. The ASME Section XI, Subsection IWL Inservice Inspection Program is described in Appendix B.

4.6 CONTAINMENT LINER PLATE FATIGUE

The interior surface of each Containment is lined with welded steel plate to provide an essentially leak-tight barrier. Design criteria are applied to the liner to assure that the specified allowed leak rate is not exceeded under the design basis accident conditions. The following fatigue loads, as described in UFSAR Appendix 5B, Section B.2.1, were considered in the design of the liner plates and are considered a time-limited aging analyses for the purposes of license renewal:

1. Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 for the plant life of 40 years.
2. Thermal cycling due to Containment interior temperature varying during the heatup and cooldown of the Reactor Coolant System. The number of cycles for this loading is assumed to be 500.
3. Thermal cycling due to the maximum hypothetical accident will be assumed to be one.
4. Thermal load cycles in the piping system are somewhat isolated from the liner plate penetrations by concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, fatigue considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the unit life.

Each of the above items has been evaluated for the period of extended operation.

For item (1.), the number of thermal cycles due to annual outdoor temperature variations was increased from 40 to 60 for the extended period of operation. The effect of this increase is insignificant in comparison to the assumed 500 thermal cycles due to Containment interior temperature varying during heatup and cooldown of the Reactor Coolant System. The 500 thermal cycles includes a margin of 300 thermal cycles above the 200 Reactor Coolant System allowable design heatup and cooldown cycles, which is sufficient margin to accommodate the additional 20 cycles of annual outdoor temperature variation. Therefore, this loading condition is considered valid for the period of extended operation as it is enveloped by item (2.).

For item (2.), the assumed 500 thermal cycles was evaluated based on the more limiting heatup and cooldown design cycles (transients) for the Reactor Coolant System. The Reactor Coolant System was designed to withstand 200 heatup and

cooldown thermal cycles. The evaluation described in Subsection 4.3.1 determined that the originally projected number of maximum Reactor Coolant System design cycles is conservative enough to envelop the projected cycles for the extended period of operation. Therefore, the original containment liner plate fatigue analysis for 500 heatup and cooldown cycles is considered valid for the period of extended operation.

For item (3.), the assumed value for thermal cycling due to the maximum hypothetical accident remains valid. No maximum hypothetical accident has occurred and none is expected, therefore, this assumption is considered valid for the period of extended operation.

For item (4.), the design of the containment penetrations has been reviewed. The design meets the general requirements of the 1965 Edition of ASME Boiler and Pressure Vessel Code, Section III. The main steam piping, feedwater piping, blowdown piping, and letdown piping are the only piping penetrating the containment wall and liner plate that contribute significant thermal loading on the liner plate. The projected number of actual operating cycles for these piping systems through 60 years of operation was determined to be less than the original design limits.

The analyses associated with the containment liner plate and penetrations have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

4.7.1 BOTTOM MOUNTED INSTRUMENTATION THIMBLE TUBE WEAR

As discussed in NRC Information Notice No. 87-44, Supplement 1, "Thimble Tube Thinning in Westinghouse Reactors," thimble tubes have experienced thinning as a result of flow-induced vibration. Thimble tube wear results in degradation of the Reactor Coolant System pressure boundary and could potentially create a non-isolable leak of reactor coolant. Therefore, the NRC staff requested that licensees perform the actions described in NRC Bulletin No. 88-09, “Thimble Tube Thinning in Westinghouse Reactors.” In response to this bulletin, Turkey Point established a program for inspection and assessment of thimble tube thinning. Turkey Point commitments to the NRC for two eddy current inspections of the thimble tubes for each unit were completed in May 1990 for Unit 4, and in December 1992 for Unit 3. The results demonstrated that the thimble tubes were acceptable for operation and that no appreciable thinning had occurred between the two inspections. Based on the results of the inspections and the flaw analyses performed, only the Unit 3 thimble tube N-05 will require further evaluation for the extended period of operation.

In order to ensure thimble tube reliability, an inspection of Unit 3 thimble tube N-05 will be conducted under the Thimble Tube Inspection Program, described in Appendix B. This aging management program will ensure that thimble tube thinning will be adequately managed for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

4.7.2 EMERGENCY CONTAINMENT COOLER TUBE WEAR

The component cooling water flow rate through the emergency containment coolers could exceed the nominal design flow during certain plant conditions. High flow rates can produce increased wear on the inside surface of the emergency containment cooler coils. The effect of increased wear was previously evaluated and the tube wall nominal thickness was determined to exceed the minimum required wall thickness during the existing operating period of 40 years. In order to ensure emergency containment cooler coil reliability, a one-time inspection for minimum tube wall thickness will be conducted prior to the end of the existing operating period to further assess the actual tube wall thinning. The inspection will be conducted in accordance with the Emergency Containment Coolers Inspection, described in Appendix B.

The Emergency Containment Coolers Inspection will ensure that the aging effect of emergency containment cooler tube wear will be adequately managed for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

4.7.3 LEAK-BEFORE-BREAK FOR REACTOR COOLANT SYSTEM PIPING

A plant-specific Leak-Before-Break (LBB) analysis was performed for Turkey Point Units 3 and 4 in 1994. The LBB analysis was performed to show that any potential leaks that develop in the Reactor Coolant System loop piping can be detected by plant monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life. As documented in the June 23, 1995, NRC letter to FPL [Reference 4.7-1], the NRC approved the Turkey Point LBB analysis. The NRC safety evaluation concluded that the LBB analysis was consistent with the criteria in NUREG-1061, Volume 3, and the draft Standard Review Plan, Section 3.6.3; therefore, the analysis complied with 10 CFR 50, Appendix A, General Design Criterion 4.

The aging effects that must be addressed during the period of extended operation include thermal aging of the primary loop piping components and fatigue crack growth. Thermal aging refers to the gradual change in the microstructure and properties of a material due to its exposure to elevated temperatures for an extended period of time. The only significant thermal aging effect on the Reactor Coolant System loop piping is embrittlement of the duplex ferritic cast austenitic stainless steel components. This effect results in a reduction in fracture toughness of the material.

The LBB analysis for Turkey Point Units 3 and 4 was revised to address the extended period of operation utilizing criteria consistent with the requirements of NUREG-1061, Volume 3, and the draft Standard Review Plan, Section 3.6.3, that the NRC had referenced in their approval of the original LBB analysis. Since the primary loop piping includes cast stainless steel fittings, fully aged fracture toughness properties were determined for each heat of material. Based on loading, pipe geometry, and fracture toughness considerations, enveloping critical locations were determined at which LBB crack stability evaluations were made. Through-wall flaw sizes were postulated at the critical locations that would cause leakage at a rate ten times the leakage detection system capability. Including the requirement for margin of applied loads, large margins against flaw instability were demonstrated for the postulated flaw sizes.

Finally, a plant-specific fatigue crack growth analysis for Turkey Point Units 3 and 4 for a 60-year plant life was performed. A design transient set that bounds the Turkey Point design transients was utilized in the fatigue crack growth analysis. Fatigue crack growth for the period of extended operation is negligible.

The Reactor Coolant System primary loop piping Leak-Before-Break analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.7.4 CRANE LOAD CYCLE LIMIT

The load cycle limit for cranes was identified as a time-limited aging analysis. At Turkey Point Units 3 and 4, the following cranes are within the scope of license renewal:

- the spent fuel pool bridge cranes
- the spent fuel cask crane
- the reactor polar cranes
- the reactor cavity manipulator cranes
- the intake structure bridge crane
- the turbine gantry cranes

The spent fuel pool bridge cranes were replaced in 1990. The spent fuel pool bridge cranes are analyzed for up to 200,000 cycles of maximum load. These 200,000 cycles are equivalent to approximately 12.7 cycles-per-day for each spent fuel pool bridge crane through the period of extended operation. Since the actual crane usage factors over their projected lives will be far less than 200,000 cycles, no additional evaluation is required.

The other cranes in the scope of license renewal were analyzed for up to 2,000,000 cycles of maximum load based on the design codes utilized for these cranes. These 2,000,000 cycles are equivalent to approximately 90 cycles-per-day for each crane through the period of extended operation. Since the actual crane usage factors over 60 years will be far less than 2,000,000 cycles, the other cranes in the scope of license renewal will continue to perform their intended function throughout the period of extended operation.

The analyses associated with crane design, including fatigue, remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.7.5 REFERENCES

4.7-1 Richard P. Croteau (NRC) letter to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping," June 23, 1995.

APPENDIX A

UPDATED
FINAL SAFETY ANALYSIS REPORT
SUPPLEMENT

INTRODUCTION

This appendix contains the Updated FSAR (UFSAR) Supplement required by 10 CFR 54.21(d) for the Turkey Point Units 3 and 4 License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Chapter 3 and Appendix B of the Turkey Point LRA provide descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Chapter 4 of the LRA contains the evaluations of the time-limited aging analyses for the period of extended operation. These LRA sections have been used to prepare the program and activity descriptions that are contained in the UFSAR Supplement. The UFSAR Supplement will be incorporated into the Turkey Point Units 3 and 4 UFSAR following issuance of the renewed operating licenses for Turkey Point. Upon inclusion of the UFSAR Supplement in the Turkey Point UFSAR, changes to the descriptions of the programs and activities for their implementation will be made in accordance with 10 CFR 50.59.

UFSAR CHAPTER 4.0
CHANGES

For the combination of normal plus design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal plus no-loss-of-function earthquake loadings, the stresses in the support structures are limited to values necessary to ensure their integrity, and to keep the stresses in the Reactor Coolant System components within the allowable limits as given in Appendix 5A.

4.1.5 CYCLIC LOADS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes. These cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes and their bases are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes ~~(40 year life)~~ and are not intended to be an ~~accurate~~ exact representation of actual transients or actual operating experience. For example the number of cycles for unit heatup and cooldown at 100°F per hour was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number~~, which averages five heatup and cooldown cycles per year,~~ could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Although loss of flow and loss of load transients are not included in Table 4.1-8 since the tabulation is only intended to represent normal design transients, the effect of these transients have been analytically evaluated and are included in the fatigue analysis for primary system components.

Over the range from 15% full power up to but not exceeding 100% of full power, the Reactor Coolant System and its components are designed to accommodate 10% of full power step changes in unit load and 5% of full power per minute ramp changes without reactor trip. The turbine bypass and steam dump system make it possible to accept a step load decrease of 50% of full power without reactor trip.

4.1.6 SERVICE LIFE

The service life of Reactor Coolant System pressure components depends upon the material irradiation, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to any appreciable material irradiation effects. The NDTT shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program which conforms with ASTM-E 185 standards.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as result of operations such as leak testing and heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (part III), Boiler and Pressure Vessel Code for Class "A" Vessels, the unit operating conditions have been established for the initial 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients. The evaluation for extended plant design life concludes that the 40-year design cycles envelope the 60-year extended design life.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

TABLE 4.1-8

DESIGN THERMAL AND LOADING CYCLES - ~~40~~ 60 YEARS

Transient Design Condition	Design Cycles	~~Expected Cycles~~
1. Station heatup at 100°F per hour	200 ~~(5/yr)~~	~~80~~
2. Station cooldown at 100°F per hour	200 ~~(5/yr)~~	~~80~~
3. Station loading at 5% of full power/min	14,500 ~~(1/day)~~	~~2500~~
4. Station unloading at 5% of full power/min	14,500 ~~(1/day)~~	~~2500~~
5. Step load increase of 10% of full power (but not to exceed full power)	2000 ~~(1/week)~~	~~500~~
6. Step load decrease of 10% of full power	2,000 ~~(1/week)~~	~~500~~
7. Step load decrease of 50% of full power	200 ~~(5/year)~~	~~20~~
8. Reactor trip	400 ~~(10/year)~~	~~40~~
9. Hydrostatic test at 3107 psig pressure, 100°F temperature	5 ~~(pre-operational)~~	~~2~~
10. Hydrostatic test at 2435 psig pressure and 400°F temperature	150 ~~(post-operational)~~	~~30~~

11. Steady state fluctuations - the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. It is assumed that an infinite number of such fluctuations will occur.

12. Feedwater Cycling/Hot Standby - This transient assumes that a low steam generation rate is made up by intermittant (slug of water) feeding of 32 °F feedwater into the steam generator. For design purposes, 2000 occurrences are assumed over the life of the plant.

Rev. [LATER] ~~14 2/97~~

The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum neutron (E > 1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectra at the sample can be applied with confidence to the adjacent section of reactor vessel, the maximum vessel exposure will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The maximum integrated fast neutron (E>1 Mev) exposure of the vessel was computed to be 5.1 x 10^{19} n/cm^2 for 40 years of operation at 2300 Mwt at 80 percent load factor. After flux reduction was imposed in 1984 and after thermal uprating performed in 1995, the maximum vessel exposure at the limiting circumferential vessel weld is ~~preducted~~ predicted to be ~~2.74~~ 4.5 x 10^{19} n/cm^2 at the end of the extended license terms (~~29.5~~ 48 EFPY* approximately) (Reference 7). The predicted extended end of life RT(ndt) is less than the 10CFR50.61 screening criteria (Reference 6).

To evaluate the RT(ndt) shift of welds, heat affected zones and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Section 4A.

* This value is approximate and will change from year to year based on the unit availability. Fluence prediction is acceptable in the ±20% range, so this value can easily vary within that limit.

4.2.13 REFERENCES

1. Westinghouse Electric Corporation, Report Number STC-TR-85-003 dated February 8, 1985, "Structural Evaluation - Pressurizer Surge Line and Spray Line for Pressurizer/RCS Differential Temperature of 320°F," PROPRIETARY.

2. Safety Evaluation, JPE-M-85-013, dated June 13, 1985, "Increased ΔT between Pressurizer and Reactor Coolant System to 320°F for PTP Unit 3."

3. NRC Letter, from G.E. Edison (NRC) to W.F. Conway (FPL), "Turkey Point Units 3 and 4 - Generic Letter 84-04, Asymmetric LOCA Loads," dated November 28, 1988.

4. NRC Letter, from R. P. Croteau (NRC) to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping (TAC Nos. M91494 and M91495)," dated June 23, 1995.

5. Westinghouse WCAP-14237, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants," dated December 1994.

6. Westinghouse WCAP-14291, "Turkey Point Units 3 and 4 Uprating Engineering Report Volume 2," dated December, 1995.

7. Westinghouse WCAP-15092, Revision 3, "Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation."

met is the more restrictive of a), the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 90% of the material yield stress at the operating temperature; or b), the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving maximum stress, do not exceed 135% of the material yield stress at operating temperature. This use of these stress criteria for this abnormal operation is consistent with the ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, paragraph N 714.2 hydrotest stress criteria. The stresses and stress factors in the actual design tube sheet, obtained using the above stress criteria, are given in Table 4.3-3.

The tube sheet designed on the above basis meets code allowable stresses for a primary to secondary differential pressure of 1520 psi. The normal operating differential pressure is 1475 psi.

The tubes have been designed to the requirements (including stress limitations) of Section III for normal operation, assuming 2485 psi as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

No significant corrosion of the Inconel tubing is expected during the lifetime of the unit. The corrosion rate reported in Reference (4), (4) shows "worst case" rates of 15.9 mg/dm^2 in the 2000 hour test under steam generator operating conditions. Conversion of this rate to a ~~40~~ 60-year unit life gives a corrosion loss of less than ~~1.5~~ 2.25 x 10^{-3} inches which is insignificant compared to the nominal tube wall thickness of 0.050 inches.

In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 1100 psi. This pressure differential is less than the primary-secondary pressure differential capability

TABLE 4.4-2

SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE
TURKEY POINT UNITS 3 & 4

Capsule(4)	Capsule Location (Degree)	Updated Lead Factor	Removal EFPY(1)	Capsule Fluence (n/cm^2)
T_3(2)	270	2.60	1.15	7.39×10^{18}
T_4(2)	270	2.48	1.17	7.08×10^{18}
S_4(2)	280	1.60	3.41	1.43×10^{19}
S_3(2)	280	1.96	3.46	1.72×10^{19}
V_3(2)	290	0.75	8.06	1.53×10^{19}
X_3(3)	270	2.48	19.4 (29 years)	2.74×10^{19}
X_4(3)	270	2.48	24.0 (34 years)	3.85×10^{19}
Y_3	150	0.49	Standby	--
U_3	30	0.49	Standby	--
W_3	40	0.34	Standby	--
Z_3	230	0.34	Standby	--
V_4	290	0.79	Standby(5)	--
Y_4	150	0.49	Standby	--
U_4	30	0.49	Standby	--
W_4	40	0.34	Standby	--
Z_4	230	0.34	Standby	--

NOTES:

(1) Effective Full Power Years (EFPY) from plant startup.

(2) Plant specific evaluation.

(3) Since the vessel controlling material is the weld metal, and only Capsule V from Unit 4 and Capsules X from Units 3 and 4 contain weld specimens, Capsule X in Units 3 and 4 were moved to the 270° location to increase the lead factor.

(4) Unit designation shown in subscript.

(5) Standby end of life capsule, as needed.

Rev. [LATER] ~~14 2/97~~

UFSAR CHAPTER 5.0
CHANGES

5.1.3 CONTAINMENT DESIGN ANALYSES

This section discusses analytical techniques, references and design philosophy for the containment building design/analyses. The results of the original analyses and the 1994 re-analysis are provided in Section 5.1.4 and Appendix 5H, respectively. The original design criteria, analyses, and construction drawings have been reviewed by Bechtel's consultants, T. Y. Lin, Kulka, Yang & Associate.

Original Analysis

The original containment structure analyses fall into two parts, axisymmetric and non-axisymetric. The axisymmetric analysis is performed through the use of a finite element computer program for the individual loads and is described in Section 5.1.3.1. The axisymmetric finite element approximation of the containment structure shell does not consider the buttresses, penetrations, brackets and anchors. These items of configuration, and lateral loads due to earthquakes or winds, and any concentrated loads, are considered in the non-axisymmetric analysis described in Section 5.1.3.2.

1994 Re-analysis

During the performance of the 20th year tendon surveillance of the Turkey Point Units 3 and 4 containment structure post-tensioning systems, a number of measured normalized tendon lift-off forces were below the predicted lower limit (PLL). Evaluation of the 20th year surveillance results concluded that the probable cause for the low tendon lift-off forces was due to an increased tendon wire steel relaxation loss caused by average tendon temperatures higher than originally considered. The evaluations also concluded that the containment post-tensioning system will provide sufficient prestress force to maintain Turkey Point licensing basis requirements through the 25th year tendon surveillance. The evaluations recommended that a structural re-analysis of the containment structure be performed to determine the minimum required prestress forces, and to establish that the containment structure will continue to meet the licensing basis requirements through the end of the licensed plant 40-year life (see Appendix 5H for additional detail).

A containment structure re-analysis was completed in 1994 and Safety Evaluation JPN-PTN-SECJ-94-027 (Reference 9) has been performed to document the results of this re-analysis.

The containment re-analysis used a three dimensional (3-D) finite element model of the containment structure. The 3-D model consisted of the cylindrical wall (including buttresses), ring girder, dome, base slab, and the major penetrations (equipment hatch and personnel hatch). The containment re-analysis did not include a new evaluation of the base slab since it was not affected by the post-tensioning system. The base slab was included in the 3-D model to provide a realistic boundary condition for the model.

Appendix 5H provides a summary of the containment re-analysis methodology, analytical techniques, references, and results.

The portions of Sections 5.1.3 and 5.1.4 relative to the original analysis of the containment structure which are affected by the 1994 re-analysis (see Appendix 5H) are annotated in the pertinent sections.

License Renewal Analysis

During the License Renewal process, the Turkey Point Units 3 and 4 containment tendons were analyzed for a 60-year life. The analysis concluded that the containment tendons will continue to meet the licensing basis requirements through the licensed plant 60-year life. (Subsection 16.3.4)

5.1.3.1 Axisymmetric Analysis (original analysis)

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Standard conventional analysis of frames and trusses can be considered to be examples of the finite element method. In the application of the method to an axisymmetric solid (e.g., a concrete containment structure) the continuous structure is replaced by a system of rings of triangular cross-section which are interconnected along circumferential joints. Based on energy principles, work equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are the unknowns. The results of the solution of this set of equations is the deformation of the structure under the given loading

Assuming that the jacking stress for the tendons is 0.80 f'_s or 192,000 psi and using the above prestress loss parameters, the following tabulation shows the magnitude of the design losses and the final effective prestress at end of 40 years for a typical dome, hoop, and vertical tendon.(5)

	Dome (Ksi)	Hoop (Ksi)	Vertical (Ksi)	Allowable (Ksi)
Temporary Jacking Stress	192	192	192	192
Friction Loss	19	21.3(1)	21	
Seating Loss	-	0	0	
Elastic Loss (average)	14.7	15.3	6.6	
Creep Loss	19.2	19.2	19.2(4)	
Shrinkage Loss	3.0	3.0	3.0	
Relaxation Loss(3)	12.5	12.5	12.5	
Final Effective Stress (2)	123.6	120.7	129.7	144.0

(1) Average of adjacent tendons

(2) This force does not include the effect of pressurization which increases the prestress force.

(3) See footnote (1) in listing at beginning of Section 5.1.4.4.

(4) To determine tendon surveillance lift-off acceptance criteria, the creep loss for the vertical tendons has been adjusted. For further details, see Reference 11 of safety evaluation JPN-PTN-SECJ-94-027 (Reference 9 on Page 5.1.3-38).

(5) The 40-year prestress losses depicted in the tabulation were utilized to calculate 60-year prestress losses for license renewal.

To provide assurance, of achievement of the desired level of Final Effective Prestress and that ACI 318-63 requirements are met, a written procedure was prepared for guidance of post-tensioning work. The procedures provided nominal values for end anchor forces in terms of pressure gage readings for calibrated jack-gage combinations. Force measurements were made at the end anchor, of course, since that is the only practical location for such measurements.

5.1.7.4 Tendon Surveillance

Provisions are made for an in-service tendon surveillance program, throughout the life of the plant that will maintain confidence in the integrity of the containment structure. ~~This program is supplemented by a corrosion control program.~~ (See Subsection 16.2.1.4 for program description relating to license renewal.)

The following quantity of tendons have been provided over and above the structural requirements:

Horizontal - Three 120 degree tendons comprising one complete hoop system.

Vertical - Three tendons spaced approximately 120 degrees apart.

Dome - Three tendons spaced approximately 120 degrees apart.

Beginning with the twentieth year tendon surveillance, inspections and lift-off readings are performed on five horizontal, four vertical, and three dome tendons. The tendons chosen for surveillance are a random but representative sample.

The surveillance program for structural integrity and corrosion protection consists of the following operations to be performed during each inspection:

(a) Lift-off readings will be taken for all of the twelve tendons.

(b) One tendon of each directional group will be relaxed and one wire from each relaxed tendon will be removed as samples for inspection. Since these tendons are re-tensioned to their original lift-off forces these samples need not be replaced.

(c) After the inspection, the tendons will be retensioned to the stress level measured at the lift-off reading and then checked by a final lift-off reading.

(d) Should the inspection of one of the wires reveal any significant corrosion (pitting, or loss of area), further inspection of the other two sets will be made to determine the extent of the corrosion and its significance to the load-carrying capacity of the structure. Samples of corroded wire will be tested to failure to evaluate the effects of any corrosion on the tensile strength of the wire.

The inspection of the four vertical tendons in the wall is sufficient to indicate any tendon corrosion that could possibly appear longitudinally along the full height of the structure. ~~Furthermore, the vertical tendons extend below the ground water table where corrosion is most likely to occur, if at all.~~ Therefore, the twelve tendons arranged as described will provide adequate corrosion surveillance.

5.1.7-6 Rev. [LATER] ~~16 10/99~~

The anchorage details permit some degree of accessibility for inspection of all tendons in the containment structure. Corrective action will be taken if and when so indicated by the surveillance program, and an adequate containment structure will be maintained throughout the life of the plant.

The following steps are taken to protect the tendons and the reinforcing steel in the containment structure from corrosion due to stray current and moisture environment.

A tendon protection sheathing filler compound encloses the whole length of every tendon. This compound will not deteriorate during the ~~forty-year~~ life of the unit. As its chemical composition is about 98% petroleum jelly, it will possess the normal stability of the linear hydrocarbons subjected to normal ambient temperature levels. The electrical resistivity of the compound is relatively high. This prevents the possibility of galvanic corrosion that would be detrimental to the tendons. Anodic corrosion centers that could develop on the surface of tendons surrounded by a good electrolyte material will not form in the presence of the protective sheathing filler.

All metallic components such as the tendon trumplate, reinforcing bars and liner plate are interconnected to form an electrically continuous cathodic structure, thereby avoiding inherent difficulties associated with isolation and interference of these members. This interconnection of the steel work with the liner plate ensures that cathodic protection currents will not be allowed to flow through any isolated member to cause electrolytic corrosion.

the combination of normal loads and design earthquake loading. Critical equipment needed for this purpose is required to operate within normal design limits.

In the case of the maximum hypothetical earthquake, it is only necessary to ensure that critical components do not lose their capability to perform their safety function, i.e., shut the unit down and maintain it in a safe condition. This capability is ensured by maintaining the stress limits as shown in Table 5A-1. No rupture of a Class I pipe is caused by the occurrence of the maximum hypothetical earthquake.

Careful design and thorough quality control during manufacture and construction and inspection during unit life, ensures that the independent occurrence of a reactor coolant pipe rupture is extremely remote. Leak-Before-Break (LBB) criteria has been applied to the reactor coolant system piping based on fracture mechanics technology and material toughness. That evaluation, together with the leak detection system, demonstrates that the dynamic effects of postulated primary loop pipe ruptures may be eliminated from the design basis (Reference 5A-2). This Leak-Before-Break evaluation was approved by the NRC for use at Turkey Point (Reference 5A-5). This evaluation has been revised for the period of extended operation, as discussed in Subsection 16.3.8.

5A-1.3.2.2 Reactor Vessel Internals

5A-1.3.2.2.1 Reactor Vessel Internals Design Criteria

The internals and core are designed for normal operating conditions and subjected to load of mechanical, hydraulic, and thermal origin. The response of the structure under the design earthquake is included in this category.

The stress criteria established in the ASME Boiler and Pressure Vessel Code, Section III, Article 4, have been adopted as a guide for the design of the internals and core with the exception of those fabrication techniques and materials which are not covered by the Code. Earthquake stresses are combined in the most conservative way and are considered primary stresses.

to accommodate the forces exerted by the restrained liner plate, and that careful attention be paid to details at corners and connections to minimize the effects of discontinuities.

The most appropriate basis for establishing allowable liner plate strains is considered to be that portion of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4. Specifically the following sections are adopted as guides in establishing the allowable strain limits:

Paragraph N 412 (m)	Thermal Stress
Paragraph N414.5	Peak Stress Intensity
	Table N 413
	Figure N 414, N 415 (A)
Paragraph N 412 (n)	
Paragraph N 415.1	

Implementation of the ASME Code requires that the liner material be prevented from experiencing significant distortion due to thermal load and that the stresses be considered from a fatigue standpoint. (Paragraph N412 (m) (2)).

The following fatigue loads are considered in the 60-year design analysis of the liner plate (See Subsection 16.3.5 for additional details):

(a) Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is ~~40~~ 60 cycles for the unit life of ~~40~~ 60 years.

(b) Thermal cycling due to the containment interior temperature variation during the startup and shutdown of the reactor system. The number of cycles for this loading is assumed to be 500 cycles.

(c) Thermal cycling due to the MHA will be assumed to be one cycle.

(d) Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by the concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III fatigue considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the 60-year unit life.

The thermal stresses in the liner plate fall into the categories considered in Article 4, Section III, of the ASME Boiler and Pressure Vessel Code. The allowable stresses in Figure N-415 (A) are for alternating stress intensity for carbon steel and temperatures not exceeding 700°F.

In accordance with ASME Code Paragraph N412 (m) 2, the liner plate is restrained against significant distortion by continuous angle anchors and never exceeds the temperature limitation of 700°F and also satisfies the criteria for limiting strains on the basis of fatigue consideration. Paragraph N412 (n) Figure N-415 (A) of the ASME Code has been developed as a result of research, industry experience, and the proven performance of code vessels, and it is a part of recognized design code. Figure N-415 (A) and its appropriate limitations have been used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415 (A) does not extend below 10 cycles, 10 cycles is being used for MHA instead of one cycle.

The maximum compressive strains are caused by accident pressure, thermal loading prestress, shrinkage and creep. The maximum strains do not exceed .0025 in/in and the liner plate always remains in a stable condition.

5B-17 Rev. [LATER]

UFSAR CHAPTER 6.0
CHANGES

Testing of Operational Sequence of Air Cleanup Systems

Criterion: A capability shall be provided to test initially under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability. (GDC 65)

Means are provided to test initially under conditions as close to design as is practical, the full operational sequence that would bring the Emergency Containment Filtering System into action, including transfer to the emergency diesel-generator power source.

6.3.6 MOTORS FOR EMERGENCY CONTAINMENT FANS

General

These totally enclosed fan cooled motors will have a useful life of ~~forty (40)~~ sixty (60) years under the normal containment service conditions as demonstrated by the appropriate EQ documentation package (See Appendix 8A). Internal heaters will dispel moisture condensation when motor is idle.

Insulation

The insulation will be a special Class B suitable for MHA conditions.
The insulation system is described in Table 6.3-2.

Bearings

The bearings will be specially selected, conservatively rated ball bearings

UFSAR CHAPTER 8.0
CHANGES

Environments in which radiation is the only parameter of concern are considered to be mild if the total radiation dose (includes ~~40~~ 60-year normal dose plus the post accident dose) is 1.0E5 rads or less. This value is the threshold for evaluation and consideration based on EPRI NP-2129. However, certain solid state electronic components and components that utilize teflon are considered to be in a mild environment only if total radiation dose is 1.0E3 rads or less.

For additional detail on the identification of environmental conditions refer to Equipment Qualification Documentation Package (Doc Pac) 1001, "Generic Approach and Treatment of Issues."

8A.5 MAINTENANCE

The purpose of the Turkey Point Equipment Qualification Maintenance Program is the preservation of the qualification of systems, structures and components. In order to accomplish this task, the plants have developed approved Design Control, Procurement and Maintenance Procedures. In addition, the component specific documentation package contains the equipment's qualified life. The qualified life is developed based upon the qualification test report reviewed in conjunction with the environmental parameters associated with the area. After this review is completed a qualified life is established. Maintenance activities to be performed in addition to the vendor recommended maintenance are determined to ensure that qualification of each piece of equipment is maintained throughout its qualified life.

8A.6 RECORDS/QUALITY ASSURANCE

A documentation package is prepared for the qualification of each manufacturer's piece of equipment under the auspices of 10CFR50.49. This package contains the information, analysis and justifications necessary to demonstrate that the equipment is properly and validly qualified as defined in 10CFR50.49 for the environmental effects of ~~40~~ 60 years of service plus a design basis accident.

This documentation package is developed from the criteria stipulated in Doc Pac 1001.

A complete listing of equipment under the auspices of 10CFR50.49 is maintained.

UFSAR CHAPTER 9.0
CHANGES

TABLE 9.2-2

NOMINAL CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE (1)

Unit design life, years	~~40~~ 60
Seal water supply flow rate, gpm (2)	24
Seal water return flow rate, gpm	9
Normal letdown flow rate, gpm	60
Maximum letdown flow rate, gpm	120
Normal charging pump flow (one pump), gpm	69
Normal charging line flow, gpm	45
Maximum rate of boration with one transfer and one charging pump from an initital RCS concentration of 1800 ppm, ppm/min	5.4
Equivalent cooldown rate to above rate of boration, °F/min	1.5
Maximum rate of boron dilution with two charging pumps from an initial RCS concentration of 2500 ppm, ppm/hour	350
Two-pump rate of boration, using refueling water, from initial RCS concentration of 10 ppm, ppm/min	6.2
Equivalent cooldown rate to above rate of boration, °F/min	1.7
Temperature of reactor coolant entering system at full power (design), °F	555.0
Temperature of coolant return to reactor coolant system at full power (design), °F	493.0
Normal coolant discharge temperature to holdup tanks, °F	127.0
Amount of 3.0 weight percent boron solution required to meet cold shutdown requirements, at end of life with peak xenon (including consideration for one stuck rod) gallons	7500

NOTES :

1. Reactor coolant water quality is given in Table 4.2-2.

2. Volumetric flow rates in gpm are based on 130°F and 2350 psig.

Rev. [LATER] ~~10 7/92~~

UFSAR CHAPTER 11.0
CHANGES

TABLE 11.1-1

WASTE DISPOSAL SYSTEM
PERFORMANCE DATA
(Two Units)

Plant Design Life	~~40~~ 60 years
Normal process capacity, liquids	Table 11.1-3
Evaporator load factor	Table 11.1-4
Annual liquid discharge	
Volume	Table 11.1-4
Activity	
Tritium	Table 11.1-5
Other	Table 11.1-5
Annual gaseous discharge	
Activity	Table 11.1-6

Rev. [LATER]

UFSAR CHAPTER 14.0
CHANGES

The neutron absorber rack design includes a poison verification view-hole in the cell wall so that the presence of poison material may be visually confirmed at any time over the life of the racks. Upon completion of rack fabrication, such an inspection was performed. This visual inspection, coupled with the Westinghouse quality assurance program controls and the use of qualified Boraflex neutron absorbing material, satisfies an initial verification test to assure that the proper quantity and placement of material was achieved during fabrication of the racks. This precludes the necessity for on-site poison verification.

As discussed in Section 4.7.2, irradiation tests have been previously performed to test the stability and structural integrity of Boraflex in boric acid solution under irradiation[7]. These tests have concluded that there is no evidence of deterioration of the suitability of the Boraflex poison material through a cumulative irradiation in excess of 1 x 10^{11} rads gamma radiation. As more data on the service life performance of Boraflex becomes available in the nuclear industry in the coming years through both experimentation and operating experience, FPL will evaluate this information and will take action accordingly. (See Subsection 16.2.2 for a program description relating to License Renewal.)

[NEW]
UFSAR CHAPTER 16.0

[NEW CHAPTER 16]

16.0 AGING MANAGEMENT PROGRAMS AND TIME-LIMITED AGING ANALYSES ACTIVITIES

The integrated plant assessment for license renewal 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 chapter describes these programs and their planned implementation.

This chapter also discusses the evaluation results for each of the plant-specific time-limited aging analyses performed for license renewal. The evaluations have demonstrated that the analyses remain valid for the period of extended operation; the analyses have been projected to the end of the period of extended operation; or that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

No 10 CFR 50.12 exemptions involving a time-limited aging analysis as defined in 10 CFR 54.3 were identified for Turkey Point.

16.1 NEW PROGRAMS

16.1.1 AUXILIARY FEEDWATER PUMP OIL COOLERS INSPECTION

The cast iron parts of the auxiliary feedwater pumps lube oil coolers and turbine governor controller oil coolers, which are wetted internally by auxiliary feedwater, are potentially susceptible to graphitic corrosion (i.e., selective leaching) and other types of corrosion. A one-time visual inspection will be performed on one of the cast iron bonnets of the auxiliary feedwater pump lube oil coolers to assess the extent of loss of material due to corrosion. The results of this inspection will be evaluated to determine the need for additional inspections/programmatic corrective actions. This inspection and evaluation will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.2 AUXILIARY FEEDWATER STEAM PIPING INSPECTION PROGRAM

The Auxiliary Feedwater Steam Piping Inspection Program manages the aging effects of loss of material due to general and pitting corrosion on the internal and external surfaces of carbon steel auxiliary feedwater steam supply lines. Periodic volumetric examinations of representative auxiliary feedwater steam supply components will be performed to ensure that minimum required wall thickness is maintained. Examinations will be performed on piping/fittings and other components using volumetric techniques, such as ultrasonic or computed radiography. The inspections will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.3 EMERGENCY CONTAINMENT COOLERS INSPECTION

A one-time volumetric examination of a sample of emergency containment coolers (ECC) tubes will be performed to determine the extent of loss of material due to erosion in the ECC tubes. The results of this inspection will be evaluated to determine the need for additional inspections/programmatic corrective actions. This inspection and evaluation will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.4 FIELD ERECTED TANKS INTERNAL INSPECTION

A one-time visual inspection to determine the extent of corrosion on the internal surfaces of the field erected tanks for both units -- including the Condensate Storage Tanks, the Demineralized Water Storage Tank, and the Refueling Water Storage Tanks -- will be performed. The results of these inspections will be evaluated to determine the need for additional inspections/programmatic corrective actions. These inspections will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.5 GALVANIC CORROSION SUSCEPTIBILITY INSPECTION PROGRAM

The Galvanic Corrosion Susceptibility Inspection Program manages the aging effect of loss of material due to galvanic corrosion on the internal surfaces of susceptible piping and components. The program involves selected, one-time inspections on the internal surfaces of piping and components with the greatest susceptibility to galvanic corrosion. Loss of material is expected mainly in carbon steel components directly coupled to stainless steel components in raw water systems, however, baseline examinations in select systems will be performed and evaluated to

establish if the corrosion mechanism is active. Based on the results of these inspections, the need for followup examinations or programmatic corrective actions will be established. The program will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.6 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The Reactor Vessel Internals Inspection Program consists of two types of examinations, visual and ultrasonic testing. The visual examination manages the aging effect of cracking due to irradiation assisted stress corrosion (IASCC) and reduction in fracture toughness due to irradiation and thermal embrittlement. The ultrasonic testing examination manages the aging effect of loss of mechanical closure integrity of reactor vessel internals bolting. The program, including an evaluation of program scope with regard to dimensional changes due to void swelling, will be in place prior to the end of the initial operating license terms for Turkey Point Units 3 and 4, and the actual visual and ultrasonic examinations, one inspection per unit, will be performed during the period of extended operation.

16.1.7 SMALL BORE CLASS 1 PIPING INSPECTION

A volumetric inspection of a sample of small bore Class 1 piping and nozzles will be performed to determine if cracking is an aging effect requiring management during the period of extended operation. This one-time inspection will address Class 1 piping less than 4 inches in diameter. Based on the results of these inspections, the need for additional inspections or programmatic corrective actions will be established. The inspection will be performed prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2 EXISTING PROGRAMS

16.2.1 ASME SECTION XI INSERVICE INSPECTION PROGRAMS

16.2.1.1 ASME SECTION XI, SUBSECTIONS IWB, IWC, AND IWD INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program inspections identify and correct degradation in Class 1, 2, and 3 components and piping. The program manages the aging effects of loss of material, cracking, and loss of mechanical closure integrity. The program provides inspection and

examination of accessible components, including welds, pump casings, valve bodies, steam generator tubing, and pressure-retaining bolting.

16.2.1.2 ASME SECTION XI, SUBSECTION IWE INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsection IWE Inservice Inspection Program inspections identify and correct degradation of pressure retaining components and their integral attachments and the metallic liner of Class CC pressure-retaining components and their integral attachments. The program manages the aging effects of loss of material and loss of pressure retention. The program provides inspection and examination of containment surfaces, seals, gaskets and moisture barriers, pressure-retaining bolting, and pressure retaining components in accordance with the requirements of ASME Section XI, Subsection IWE.

16.2.1.3 ASME SECTION XI, SUBSECTION IWF INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsection IWF Inservice Inspection Program inspections identify and correct degradation of ASME Class 1, 2, and 3 component supports. This program manages the aging effect of loss of material. The scope of the program provides for inspection and examination of accessible surface areas of the component supports in accordance with the requirements of ASME Section XI, Subsection IWF.

16.2.1.4 ASME SECTION XI, SUBSECTION IWL INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsection IWL Inservice Inspection Program inspections assess the quality and structural performance of the Containment structure post-tensioning system components. The program manages the aging effects of loss of material and confirms the results of the Containment tendon loss of prestress Time-Limited Aging Analysis (see Subsection 16.3.4). The program includes inspection of tendon and anchorage hardware surfaces and measurement of tendon force and elongation.

16.2.2 BORAFLEX SURVEILLANCE PROGRAM

The Boraflex Surveillance Program manages the aging effect of change in material properties for the Boraflex material in the spent fuel storage racks.

The program will be enhanced to provide for density testing (or other approved testing methods if available) of the encapsulated Boraflex material in the spent fuel

storage racks prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.3 BORIC ACID WASTAGE SURVEILLANCE PROGRAM

The Boric Acid Wastage Surveillance Program manages the aging effects of loss of material and mechanical closure integrity due to aggressive chemical attack resulting from borated water leaks. The program addresses the Reactor Coolant System and structures and components containing, or exposed to, borated water. This program utilizes systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of pressure boundary or structural integrity of components, supports, or structures, including electrical equipment in proximity to borated water systems. This program includes commitments to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

Some systems outside Containment (i.e., Spent Fuel Pool Cooling and portions of Waste Disposal associated with containment integrity) are currently inspected under other existing programs. The scope of the Boric Acid Wastage Surveillance Program will be enhanced to include these systems and components prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.4 CHEMISTRY CONTROL PROGRAM

The Chemistry Control Program manages loss of material, cracking, and fouling aging effects for primary and secondary systems, structures, and components. The aging effects are minimized or prevented by controlling the chemical species that cause the underlying mechanism(s) that results in these aging effects. Alternatively, chemical agents, such as corrosion inhibitors and biocides, are introduced to prevent certain aging mechanisms. The program includes sampling activities and analysis. The program provides assurance that an elevated level of contaminants and oxygen does not exist in the systems, structures, and components covered by the program, and thus prevents and minimizes the occurrences of aging effects.

16.2.5 CONTAINMENT SPRAY SYSTEM PIPING INSPECTION PROGRAM

The Containment Spray System Piping Inspection Program manages the aging effect loss of material due to general, crevice, and pitting corrosion on the internal surfaces of carbon steel piping and fittings, and valves wetted by boric acid in the Containment Spray System spray headers. Periodic ultrasonic examinations of

selected locations are used to determine wall thickness and are evaluated to ensure that minimum thickness requirements are maintained.

16.2.6 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification Program evaluations of electrical equipment are identified as Time-Limited Aging Analyses. Equipment covered by the Environmental Qualification Program has been evaluated to determine if the existing Environmental Qualification aging analyses can be projected to the end of the period of extended operation by reanalysis or additional analysis. Qualification into the license renewal period is treated the same as equipment initially qualified for 40 years or less. When analysis cannot justify a qualified life in excess of the license renewal period, then the component parts will be replaced, refurbished, or requalified prior to exceeding the qualified life in accordance with the Environmental Qualification Program.

16.2.7 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program is designed to track design cycles to ensure that Reactor Coolant System components remain within their design fatigue limits. Design cycle limits for Turkey Point Units 3 and 4 are provided in Table 4.1-8. The specific fatigue analyses validated by the Fatigue Monitoring Program are associated with the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines. Administrative procedures provide the methodology for logging design cycles. Guidance is provided in the event design cycle limits are approached.

16.2.8 FIRE PROTECTION PROGRAM

The Fire Protection Program manages the aging effects of loss of material, cracking, and fouling for the components/piping of the Fire Protection System and Fire Rated Assemblies. Additionally, this program manages the aging effects of loss of material, loss of seal, cracking, and erosion for structures and structural components associated with fire protection. Appendix 9.6A contains a detailed discussion of the Fire Protection Program.

The scope of the Fire Protection Program will be enhanced to include inspection of additional components prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.9 FLOW ACCELERATED CORROSION PROGRAM

The Flow Accelerated Corrosion Program manages the aging effect of loss of material due to flow accelerated corrosion. The Flow Accelerated Corrosion Program predicts, detects, monitors, and mitigates flow accelerated corrosion wear in high energy carbon steel piping associated with the Main Steam and Turbine Generators, and Feedwater and Blowdown Systems, and is based on industry guidelines and experience. The program includes analysis and baseline inspections; determination, evaluation, and corrective actions for affected components; and follow-up inspections.

This program will be enhanced to address internal and external loss of material of steam trap lines due to flow accelerated corrosion and general corrosion, respectively, prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.10 INTAKE COOLING WATER SYSTEM INSPECTION PROGRAM

The Intake Cooling Water System Inspection Program manages the aging effects of loss of material due to various corrosion mechanisms, stress corrosion cracking, and biological fouling for Intake Cooling Water System components. The program includes inspections, performance testing, evaluations, and corrective actions that are performed as the result of FPL commitments to NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.”

This program will be enhanced to improve documentation of scope and frequency of the intake cooling water piping crawl-through inspections and component cooling water heat exchanger tube integrity inspections prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.11 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

The Periodic Surveillance and Preventive Maintenance Program manages the aging effects of loss of material, cracking, fouling buildup, loss of seal, and embrittlement for systems, structures, and components. The scope of the program provides for visual inspection and examination of selected surfaces of specific components and structural components. The program also includes leak inspection of limited portions of the Chemical and Volume Control Systems. Additionally, the program provides

for replacement/refurbishment of selected components on a specified frequency, as appropriate.

Specific enhancements to the scope and documentation of some inspections performed under this program will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.12 REACTOR VESSEL HEAD ALLOY 600 PENETRATION INSPECTION PROGRAM

The Reactor Vessel Head Alloy 600 Penetration Inspection Program encompasses the Turkey Point Units 3 and 4 reactor vessel head Alloy 600 penetrations that are part of the Reactor Coolant System pressure boundary. This program manages the aging effect of cracking due to primary water stress corrosion (PWSCC). The program includes a one-time volumetric examination of selected Unit 4 reactor vessel head penetrations to detect crack initiation. Visual examination of the Unit 3 and Unit 4 reactor vessel head external surfaces during outages and the Boric Acid Wastage Surveillance Program are also utilized to manage cracking.

16.2.13 REACTOR VESSEL INTEGRITY PROGRAM

The Reactor Vessel Integrity Program manages reactor vessel irradiation embrittlement and encompasses the following subprograms:

- Reactor Vessel Surveillance Capsule Removal and Evaluation
- Fluence and Uncertainty Calculations
- Monitoring Effective Full Power Years
- Pressure-Temperature Limit Curves

Program documentation will be enhanced to integrate all aspects of the Reactor Vessel Integrity Program prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.13.1 REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL AND EVALUATION

This subprogram manages the aging effect of reduction in fracture toughness of the Turkey Point Units 3 and 4 reactor vessel materials (beltline forgings and circumferential welds) due to neutron irradiation embrittlement by performing Charpy V-notch and tensile tests on the reactor vessel irradiated specimens. The Reactor Vessel Surveillance Capsule Removal and Evaluation subprogram is a

NRC-approved program that meets the requirements of 10 CFR 50, Appendix H. The surveillance capsule withdrawal schedule is specified in Table 4.4-2.

16.2.13.2 FLUENCE AND UNCERTAINTY CALCULATIONS

This subprogram provides an accurate prediction of the Turkey Point Units 3 and 4 reactor vessel accumulated fast neutron fluence values at the reactor vessel beltline forgings and circumferential welds.

16.2.13.3 MONITORING EFFECTIVE FULL POWER YEARS

This subprogram accurately monitors and tabulates the accumulated operating time experienced by the reactor vessels to ensure that the Turkey Point Units 3 and 4 pressure-temperature limit curves and end-of-life reference temperatures are not exceeded.

16.2.13.4 PRESSURE-TEMPERATURE LIMIT CURVES

This subprogram provides pressure-temperature limit curves for the Turkey Point Units 3 and 4 reactor vessels to establish the Reactor Coolant System operating limits. The pressure-temperature limit curves are included in the Technical Specifications.

16.2.14 STEAM GENERATOR INTEGRITY PROGRAM

The Steam Generator Integrity Program ensures steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The program manages the aging effects of cracking and loss of material and includes the following essential elements:

- Inspection of steam generator tubing and tube plugs
- Steam generator secondary-side integrity inspections
- Tube integrity assessments
- Assessment of degradation mechanisms
- Primary-to-secondary leakage monitoring
- Primary and secondary chemistry control
- Sludge lancing
- Maintenance and repairs

- Foreign material exclusion

16.2.15 SYSTEMS AND STRUCTURES MONITORING PROGRAM

The Systems and Structures Monitoring Program manages the aging effects of loss of material, cracking, fouling, loss of seal, and change in material properties. The program provides for periodic visual inspection and examination for degradation of accessible surfaces of specific systems, structures, and components, and corrective actions as required based on these inspections.

This program will be enhanced by restructuring it to address inspection requirements to manage certain aging effects in accordance with 10 CFR 54, modifying the scope of specific inspections, and improving documentation requirements prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.16 THIMBLE TUBE INSPECTION PROGRAM

The Thimble Tube Inspection Program manages the aging effect of material loss due to fretting wear. This program consists of an eddy current test inspection of thimble tube N-05 on Unit 3. Eddy current testing of thimble tubes was initiated in response to NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors,” and inspections have been performed on all in-service thimble tubes for Units 3 and 4. This inspection will be performed prior to the end of the initial operating license term for Turkey Point Unit 3.

16.3 TIME-LIMITED AGING ANALYSIS ACTIVITIES

16.3.1 REACTOR VESSEL IRRADIATION EMBRITTLEMENT

The Turkey Point Units 3 and 4 reactor vessels are described in Chapters 3.0 and 4.0. Time-limited aging analyses (TLAAs) applicable to the reactor vessels are:

- pressurized thermal shock
- upper-shelf energy
- pressure-temperature limits

The Reactor Vessel Integrity Program, described in Subsection 16.2.13, manages reactor vessel irradiation embrittlement utilizing subprograms to monitor, calculate, and evaluate the time-dependent parameters used in the aging analyses for pressurized thermal shock, Charpy upper-shelf energy, and pressure-temperature limit curves to ensure continuing vessel integrity through the period of extended operation.

16.3.1.1 PRESSURIZED THERMAL SHOCK

The requirements in 10 CFR 50.61 provide rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of the maximum nil ductility reference temperature (RT_{PTS}) whenever a significant change occurs in projected values of RT_{PTS}, or upon request for a change in the expiration date for the operation of the facility.

The calculated RT_{PTS} values at the end of the extended period of operation (48 effective full power years) for the Turkey Point Units 3 and 4 reactor vessels are less than the 10 CFR 50.61(b)(2) screening criteria of 270° F for intermediate and lower shells and 300° F for the circumferential welds. Based upon the revised calculations, additional measures will not be required for the Turkey Point reactor vessels during the license renewal period.

The analysis associated with pressurized thermal shock has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.1.2 UPPER-SHELF ENERGY

The requirements on reactor vessel Charpy upper-shelf energy are included in 10 CFR 50, Appendix G. Specifically, 10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any reactor vessel material is predicted to drop below 50 ft-lb., as measured by Charpy V-notch specimen testing.

A fracture mechanics evaluation was performed in accordance with Appendix K of ASME Section XI to demonstrate continued acceptable equivalent margins of safety against fracture through 48 effective full power years.

The analysis associated with upper-shelf energy has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.1.3 PRESSURE-TEMPERATURE LIMITS

The requirements in 10 CFR 50, Appendix G, ensure that heatup and cooldown of the reactor pressure vessel are accomplished within established pressure-temperature limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced. Operation of the Reactor Coolant System is also limited by the net positive suction curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. Therefore, in order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G pressure-temperature limits and the net positive suction curves.

To address the period of extended operation, the 48 effective full power year projected fluences and the Turkey Point-specific reactor vessel material properties were used to determine the limiting material and calculate pressure-temperature limits for heatup and cooldown. The limiting material at all temperatures for the period of extended operation is the circumferential girth weld.

A license amendment to incorporate the pressure-temperature limit curves projected to 48 effective full power years will be submitted to the NRC for review and approval prior to exceeding the licensed operating period for these curves.

The analysis associated with reactor vessel pressure-temperature limit curves has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

16.3.2 METAL FATIGUE

The thermal and mechanical fatigue analyses of plant mechanical components have been identified as time-limited aging analyses for Turkey Point. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the Turkey Point UFSAR.

16.3.2.1 ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, CLASS 1 COMPONENTS

The reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the Turkey Point Units 3 and 4 Nuclear Steam Supply System components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various Nuclear Steam Supply System components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycle set applicable to the Class 1 components was assembled. The actual frequency of occurrence for the design cycles was determined and compared to the design cycle set. The severity of the actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis, the design cycle profiles envelop actual plant operation. In addition, a review of the applicable

administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation.

The analyses associated with verifying the structural integrity of the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

As a confirmatory program, the monitoring of plant transients performed as a part of the Fatigue Monitoring Program, as described in Subsection 16.2.7, will assure that the design cycle limits are not exceeded.

16.3.2.2 REACTOR VESSEL UNDERCLAD CRACKING

In early 1971, an anomaly identified as grain boundary separation, perpendicular to the direction of the cladding weld overlay, was identified in the heat-affected zone of reactor vessel base metal. A generic fracture mechanics evaluation demonstrated that the growth of underclad cracks during a 40-year plant life is insignificant.

The evaluation was extended to 60 years using fracture mechanics evaluations based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service life. The 60-year evaluation shows insignificant growth of the underclad cracks.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

16.3.2.3 REACTOR COOLANT PUMP FLYWHEEL

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to potentially produce high-energy missiles in the unlikely event of failure. Conditions which may result in overspeed of the reactor coolant pump increase both the potential for failure and the kinetic energy. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway. An evaluation of the probability of failure over the extended period of operation was performed. It demonstrates that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack extension over a 60-year service life.

The analysis associated with the structural integrity of the reactor coolant pump flywheel has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

16.3.2.4 ANSI B31.1 PIPING

The Reactor Coolant System primary loop piping and balance-of-plant piping are designed to the requirements of ANSI B31.1, Power Piping. The exceptions are the Units 3 and 4 pressurizer surge lines and the Unit 4 Emergency Diesel Generator safety-related piping.

The pressurizer surge lines have been designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 1.

The Unit 4 Emergency Diesel Generator safety-related piping has been designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 3, which is essentially the same as ANSI B31.1 design requirements. The evaluation of the Unit 4 Emergency Diesel Generator safety-related piping fatigue is, therefore, included in the discussion below.

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was performed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and vary operating temperatures only during plant heatup and cooldown, during plant transients, or during periodic testing. The results of the evaluation for ANSI B31.1 piping systems demonstrate that the number of assumed thermal cycles would not be exceeded in 60 years of plant operation.

The analyses associated with ANSI B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

16.3.2.5 ENVIRONMENTALLY ASSISTED FATIGUE

The Turkey Point approach to address reactor water environmental effects accomplishes two objectives. First, the TLAA on fatigue design has been resolved by confirming that the original transient design limits remain valid for the 60-year operating period. Confirmation by fatigue monitoring will ensure these transient design limits are not exceeded. Second, reactor water environmental effects on fatigue life are examined using the most recent data from laboratory simulation of the reactor coolant environment.

As a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratories (INEL) evaluated, in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995, fatigue-sensitive component locations at plants designed by all four U. S. Nuclear Steam Supply System (NSSS) vendors. The pressurized water reactor (PWR) calculations, especially the early-vintage Westinghouse PWR calculations, are directly relevant to Turkey Point. The description of the “Older Vintage Westinghouse Plant” evaluated in NUREG/CR-6260 matches Turkey Point with respect to design code. In addition, the transient cycles considered in the evaluation match or bound Turkey Point design.

NUREG/CR-6260 calculated fatigue usage factors for critical fatigue-sensitive component locations for the early-vintage Westinghouse plant utilizing the interim fatigue curves provided in NUREG/CR-5999, "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments," August 1993. The results of NUREG/CR-6260 analyses were then utilized to scale up the Turkey Point plant-specific usage factors for the same locations to account for environmental effects. Generic industry studies performed by EPRI and NEI were also considered in this aspect of the evaluation, as well as environmental data that have been collected and published subsequent to the generic industry studies. Based on these adjustments, only the pressurizer surge line piping required further evaluation for the period of extended operation.

In lieu of additional analyses to refine the usage factor for the pressurizer surge lines, Turkey Point has selected aging management to address pressurizer surge line fatigue during the period of extended operation. In particular, the potential for crack initiation and growth, including reactor water environmental effects, will be managed by the Turkey Point ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, as described in Subsection 16.2.1.1.

16.3.3 ENVIRONMENTAL QUALIFICATION

The thermal, radiation, and wear cycle aging analyses of plant electrical and I&C components have been identified as time-limited aging analyses for Turkey Point. In particular, the environmental qualification evaluations of electrical equipment with a 40-year qualified life or greater have been determined to be time-limited aging analyses.

Equipment included in the Turkey Point Environmental Qualification Program has been evaluated to determine if existing environmental qualification aging analyses can be projected to the end of the period of extended operation by reanalysis or additional analysis. Qualification into the license renewal period is treated the same as for equipment currently qualified at Turkey Point for 40 years or less. When aging analysis cannot justify a qualified life into the license renewal period, then the component or parts will be replaced prior to exceeding their qualified lives in accordance with the Environmental Qualification Program, as described in Subsection 16.2.6.

Age-related service conditions that are applicable to the environmentally qualified equipment (i.e., 60 years of exposure versus 40 years) were evaluated for the period of extended operation to verify that the current environmental qualification analyses are bounding. The evaluations considered radiation, thermal, and wear cycle aging effects.

Therefore, the analyses associated with the environmental qualification of electrical equipment remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i), or have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.4 CONTAINMENT TENDON LOSS OF PRESTRESS

The Turkey Point Units 3 and 4 containment buildings are post-tensioned, reinforced concrete structures composed of vertical cylinder walls and a shallow dome, supported on a conventional reinforced concrete base slab. The cylinder walls are provided with vertical tendons and horizontal hoop tendons. The dome is provided with three groups of tendons oriented 120-degrees apart.

The prestress of containment tendons decreases over time as a result of seating of anchorage losses, elastic shortening of concrete, creep of concrete, shrinkage of

concrete, relaxation of prestressing steel, and friction losses. New upper limit curves, lower limit curves, and trend lines of measured prestressing forces have been established for all tendons through the period of extended operation. The predicted final effective preload at the end of 60 years exceeds the minimum required preload for all containment tendons. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation.

The analyses associated with containment tendon loss of prestress have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

As a confirmatory program, the Containment structure post-tensioning system surveillance performed as a part of the ASME Section XI, Subsection IWL Inservice Inspection Program, as described in Subsection 16.2.1.4, will continue to be performed in accordance with the requirements of plant Technical Specifications.

16.3.5 CONTAINMENT LINER PLATE FATIGUE

The interior surface of each Containment is lined with welded steel plate to provide an essentially leak-tight barrier. Design criteria are applied to the liner to assure that the specified allowed leak rate is not exceeded under the design basis accident conditions. The fatigue loads, as described in Appendix 5B, Section B.2.1, were considered in the design of the liner plates and are considered time-limited aging analyses for the purposes of license renewal. Each of these has been evaluated for the period of extended operation.

The number of thermal cycles due to annual outdoor temperature variations was increased from 40 to 60 for the extended period of operation. The effect of this increase is insignificant in comparison to the assumed 500 thermal cycles due to Containment interior temperature varying during heatup and cooldown of the Reactor Coolant System. The 500 thermal cycles includes a margin of 300 thermal cycles above the 200 Reactor Coolant System allowable design heatup and cooldown cycles, which is sufficient margin to accommodate the additional 20 cycles of annual outdoor temperature variation. Therefore, this loading condition is considered valid for the period of extended operation as it is enveloped by the evaluation for 500 thermal cycles.

The assumed 500 thermal cycles was evaluated based on the more limiting heatup and cooldown design cycles (transients) for the Reactor Coolant System. The Reactor Coolant System was designed to withstand 200 heatup and cooldown

thermal cycles. The evaluation determined that the originally projected number of maximum Reactor Coolant System design cycles is conservative enough to envelop the projected cycles for the extended period of operation. Therefore, the original containment liner plate fatigue analysis for 500 heatup and cooldown cycles is considered valid for the period of extended operation.

The assumed value of one for thermal cycling due to the maximum hypothetical accident remains valid. No maximum hypothetical accident has occurred and none is expected, therefore, this assumption is considered valid for the period of extended operation.

The design of the containment penetrations has been reviewed. The design meets the general requirements of the 1965 Edition of ASME Boiler and Pressure Vessel Code, Section III. The main steam piping, feedwater piping, blowdown piping, and letdown piping are the only piping penetrating the containment wall and liner plate that contribute significant thermal loading on the liner plate. The projected number of actual operating cycles for these piping systems through 60 years of operation was determined to be less than the original design limits.

The analyses associated with the containment liner plate and penetrations have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

16.3.6 BOTTOM MOUNTED INSTRUMENTATION THIMBLE TUBE WEAR

As discussed in NRC Information Notice No. 87-44, Supplement 1, "Thimble Tube Thinning in Westinghouse Reactors," thimble tubes have experienced thinning as a result of flow-induced vibration. Thimble tube wear results in degradation of the Reactor Coolant System pressure boundary and could potentially create a non-isolable leak of reactor coolant. Therefore, the NRC staff requested that licensees perform the actions described in NRC Bulletin No. 88-09, “Thimble Tube Thinning in Westinghouse Reactors.” In response to this bulletin, FPL established a program for inspection and assessment of thimble tube thinning. Turkey Point commitments to the NRC for two eddy current inspections of the thimble tubes for each unit were completed in May 1990 for Unit 4, and in December 1992 for Unit 3. The results demonstrated that the thimble tubes were acceptable for operation and that no appreciable thinning had occurred between the two inspections. Based on the results of the inspections and the analyses performed, only the Unit 3 thimble tube N-05 will require further evaluation for the extended period of operation.

In order to ensure thimble tube reliability, an inspection of Unit 3 thimble tube N-05 will be conducted under the Thimble Tube Inspection Program, described in Subsection 16.2.16. This aging management program will ensure that thimble tube thinning will be adequately managed for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

16.3.7 EMERGENCY CONTAINMENT COOLER TUBE WEAR

The component cooling water flow rate through the emergency containment coolers could exceed the nominal design flow during certain plant conditions. High flow rates can produce increased wear on the inside surface of the emergency containment cooler tubes. The effect of increased wear was previously evaluated and the tube wall nominal thickness was determined to exceed the minimum required wall thickness during the existing operating period of 40 years. In order to ensure emergency containment cooler tube reliability, a one-time inspection for minimum tube wall thickness will be conducted on a sample of cooler tubes prior to the end of the existing operating period to further assess the actual tube wall thinning. The inspection will be conducted in accordance with the Emergency Containment Coolers Inspection, described in Subsection 16.1.3.

The Emergency Containment Coolers Inspection will ensure that the aging effect of emergency containment cooler tube wear will be adequately managed for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

16.3.8 LEAK-BEFORE-BREAK FOR REACTOR COOLANT SYSTEM PIPING

A plant-specific Leak-Before-Break (LBB) analysis was performed for Turkey Point Units 3 and 4 in 1994. The LBB analysis was performed to show that any potential leaks that develop in the Reactor Coolant System loop piping can be detected by plant monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life. As documented in the June 23, 1995, NRC letter to FPL (Appendix 5A, Reference 5A-5), the NRC approved the Turkey Point LBB analysis. The NRC safety evaluation concluded that the LBB analysis was consistent with the criteria in NUREG-1061, Volume 3, and the draft Standard Review Plan, Section 3.6.3; therefore, the analysis complied with GDC-4.

The LBB analysis for Turkey Point was revised to address the extended period of operation utilizing criteria consistent with the requirements of NUREG-1061,

Volume 3, and the draft Standard Review Plan, Section 3.6.3, that the NRC had referenced in their approval of the original LBB analysis. Since the primary loop piping includes cast stainless steel fittings, fully aged fracture toughness properties were determined for each heat of material. Based on loading, pipe geometry, and fracture toughness considerations, enveloping critical locations were determined at which LBB crack stability evaluations were made. Through-wall flaw sizes were postulated at the critical locations that would cause leakage at a rate ten times the leakage detection system capability. Including the requirement for margin of applied loads, large margins against flaw instability were demonstrated for the postulated flaw sizes.

Finally, a plant-specific fatigue crack growth analysis for Turkey Point Units 3 and 4 for a 60-year plant life was performed. A design transient set that bounds the Turkey Point design transients was utilized in the fatigue crack growth analysis. Fatigue crack growth for the period of extended operation is negligible.

The Reactor Coolant System primary loop piping Leak-Before-Break analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.9 CRANE LOAD CYCLE LIMIT

The crane load cycle limit was identified as a time-limited aging analysis for the cranes within the scope of license renewal. They include the polar cranes, reactor cavity manipulator cranes, spent fuel pool bridge cranes, spent fuel cask crane, turbine gantry crane, and intake structure bridge crane.

The load cycles for these cranes were evaluated for the period of extended operation. For each crane, the actual usage over the projected life through the period of extended operation will be far less than the analyzed quantity of cycles. All the cranes in the scope of license renewal will continue to perform their intended function throughout the period of extended operation.

Therefore, the analyses associated with crane design, including fatigue, are valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

APPENDIX B

AGING MANAGEMENT PROGRAMS

TABLE OF CONTENTS

1.0 INTRODUCTION

The Turkey Point Integrated Plant Assessment comprises four major activities, consistent with the draft NRC, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" [Reference B-1]. The first two activities, "Identification of Structures and Components that are Subject to Aging Management Review" and "Identification of Aging Effects Requiring Management," have been described in the body of this Application. The remaining major activities, "Identification of Plant-specific Programs That Will Manage the Identified Aging Effects Requiring Management" and "Aging Management Demonstration for Existing Programs," are described herein.

The Turkey Point programs described herein, with the exception of the Environmental Qualification Program and the Fatigue Monitoring Program, are credited for managing the effects of aging. The Environmental Qualification Program is credited for ensuring the qualified life of electrical and I&C components within the scope of 10 CFR 50.49 is maintained. The Fatigue Monitoring Program is credited for confirming that Reactor Coolant System design cycle assumptions remain valid. The programs described include both existing programs and new programs currently not being conducted. Aging management programs provide reasonable assurance that the effects of aging will be adequately managed so that the structures and components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. The demonstrations, along with the program and activity descriptions, meet the requirements of 10 CFR 54.21(a)(3). Along with the technical information contained in the body of this Application, this appendix is intended to allow the NRC to make the finding contained in 10 CFR 54.29(a)(1).

Commitment dates associated with the implementation of new programs and enhancements to existing programs are contained in Appendix A.

2.0 AGING MANAGEMENT PROGRAM ATTRIBUTES

The attributes that are used to describe aging management programs are discussed in this section. NEI 95-10 [Reference B-2], Sections 4.2 and 4.3, served as the primary input to the attribute definitions used in this appendix.

Two attributes common to all programs discussed in this appendix are Corrective Actions and Administrative Controls. They are described as follows:

Corrective Actions

This attribute is a description of the action taken when the established acceptance criterion or standard is not met. This includes timely root cause determination and prevention of recurrence, as appropriate.

Administrative Controls

This attribute is an identification of the plant administrative structure under which the programs are executed.

FPL has established and implemented a Quality Assurance Program to provide assurance that the design, procurement, modification, and operation of nuclear power plants conform to applicable regulatory requirements. The FPL Quality Assurance Program, described in the FPL Topical Quality Assurance Report, is in compliance with the requirements of 10 CFR 50, Appendix B. The FPL Quality Assurance Program meets the requirements provided by the NRC Regulatory Guidance and Industry Standards as listed in Appendix C of the FPL Topical Quality Assurance Report.

Corrective Actions and Administrative Controls apply to aging management programs credited for license renewal and performed, or in the case of new programs to be performed, in accordance with the FPL Quality Assurance Program. Accordingly, discussion of Corrective Actions and Administrative Controls is not included in the summary descriptions of the individual programs in this appendix.

The remaining attribute definitions used to describe new and existing programs are:

Scope

This attribute is a clear statement of the reason why the program exists for license renewal.

Preventive Actions

This attribute is a description of preventive actions taken to mitigate the effects of the susceptible aging mechanisms and basis for the effectiveness of these actions.

Parameters Monitored or Inspected

This attribute is a description of parameters monitored or inspected, and how they relate to the degradation of the particular component or structure and its intended function.

Detection of Aging Effects

This attribute is a description of the type of action or technique used to identify or manage the aging effects or relevant conditions.

Monitoring and Trending

This attribute is a description of the monitoring, inspection, or testing frequency and sample size (if applicable).

Acceptance Criteria

This attribute is identification of the acceptance criteria or standards for the relevant conditions to be monitored or the chosen examination methods.

Confirmation Process

This attribute is a description of the process to ensure that adequate corrective actions have been completed and are effective.

Operating Experience and Demonstration

This attribute is a summary of the operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs. Program demonstration is also included in this summary.

3.0 AGING MANAGEMENT PROGRAMS

The following programs are credited to manage the aging effects for license renewal.

New Aging Management Programs

- Auxiliary Feedwater Pump Oil Coolers Inspection
- Auxiliary Feedwater Steam Piping Inspection Program
- Emergency Containment Coolers Inspection
- Field Erected Tanks Internal Inspection
- Galvanic Corrosion Susceptibility Inspection Program
- Reactor Vessel Internals Inspection Program
  - Visual Examination
  - Ultrasonic Examination
- Small Bore Class 1 Piping Inspection

Existing Aging Management Programs

- ASME Section XI Inservice Inspection Programs
  - ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
  - ASME Section XI, Subsection IWE Inservice Inspection Program
  - ASME Section XI, Subsection IWF Inservice Inspection Program
  - ASME Section XI, Subsection IWL Inservice Inspection Program
- Boraflex Surveillance Program
- Boric Acid Wastage Surveillance Program
- Chemistry Control Program
- Containment Spray System Piping Inspection Program
- Environmental Qualification Program
- Fatigue Monitoring Program
- Fire Protection Program

- Flow Accelerated Corrosion Program
- Intake Cooling Water System Inspection Program
- Periodic Surveillance and Preventive Maintenance Program
- Reactor Vessel Head Alloy 600 Penetration Inspection Program
- Reactor Vessel Integrity Program
  - Reactor Vessel Surveillance Capsule Removal and Evaluation
  - Fluence and Uncertainty Calculations
  - Monitoring Effective Full Power Years
  - Pressure-Temperature Limit Curves
- Steam Generator Integrity Program
- Systems and Structures Monitoring Program
- Thimble Tube Inspection Program

Demonstration that each of the above programs adequately addresses the identified aging effect is in the following sections.

3.1 NEW AGING MANAGEMENT PROGRAMS

3.1.1 AUXILIARY FEEDWATER PUMP OIL COOLERS INSPECTION

As identified in Chapter 3, the Auxiliary Feedwater Pumps Oil Coolers Inspection is credited for aging management of the auxiliary feedwater pumps in Auxiliary Feedwater and Condensate Storage.

Scope

This inspection is intended to be a one-time inspection of an oil cooler of one of the three shared auxiliary feedwater pumps. The Auxiliary Feedwater Pump Oil Coolers Inspection will manage the effects of loss of material due to graphitic corrosion (i.e., selective leaching) and other types of corrosion of the internal surfaces of cast iron parts of the oil coolers wetted internally by treated water – secondary. A visual inspection will be performed to detect loss of material. The inspection will include the cast iron bonnet of one of the auxiliary feedwater pump lube oil coolers and, if necessary, the cast iron parts of an auxiliary feedwater turbine governor controller oil cooler. Commitment dates associated with the implementation of this new program are contained in Appendix A.

Preventive Actions

No preventive actions are applicable to this inspection.

Parameters Monitored or Inspected

The Auxiliary Feedwater Pump Oil Coolers Inspection will identify the presence of graphitic corrosion activity and will quantify the loss of structurally sound wall thickness of cast iron parts. The inspection will consist of two parts, an “as found” inspection of parts and an inspection of parts after light sandblasting to bare metal.

Detection of Aging Effects

Visual inspection will be used to verify whether graphitic corrosion has taken place. The aging effect of concern, loss of material due to graphitic corrosion and other types of corrosion, will be further evident by the reduced wall thickness of structurally sound material in the cast iron parts being examined (following the sandblasting).

Monitoring and Trending

As stated above, this inspection is intended to be a one-time inspection of one cooler. If significant loss of material due to graphitic corrosion or other corrosion is detected, Turkey Point will assess the extent of the corrosion, and determine if inspection of other coolers and additional future monitoring are required.

Acceptance Criteria

If the inspection results in white non-porous metallic surface without major indications, Turkey Point may declare the part as “not affected by graphitic corrosion” and not require further evaluation. If there is evidence of significant effects of graphitic corrosion, an evaluation will be prepared to establish the minimum required wall thickness including a corrosion allowance adequate for a pre-determined inspection interval. Wall thickness measurements greater than minimum wall thickness values will be acceptable.

Confirmation Process

Follow-up examination requirements will be established based on the evaluation of the inspection results and will be entered into the corrective action program.

Operating Experience and Demonstration

Visual inspections and wall thickness measurements of equipment have been performed at Turkey Point for many years. The techniques have proven successful in determining actual material condition of components.

This Auxiliary Feedwater Pump Oil Coolers Inspection is a new program that will use techniques with demonstrated capability and a proven industry record to detect loss of material due to graphitic corrosion. Visual examination has been used in the past to identify graphitic corrosion. This inspection will be performed utilizing approved procedures and qualified personnel.

Based upon the above, the implementation of the Auxiliary Feedwater Pump Oil Coolers Inspection will provide reasonable assurance that loss of material will be managed such that components within the scope of license renewal will continue to perform their intended functions consistent with current licensing basis for the period of extended operation.

3.1.2 AUXILIARY FEEDWATER STEAM PIPING INSPECTION PROGRAM

As identified in Chapter 3, the Auxiliary Feedwater Steam Piping Inspection Program is credited for aging management of steam piping associated with Auxiliary Feedwater and Condensate Storage.

Scope

The Auxiliary Feedwater Steam Piping Inspection Program will manage the effects of loss of material due to general and pitting corrosion on the internal and external surfaces of the auxiliary feedwater steam supply carbon steel piping and fittings. The program will provide for representative volumetric examinations to detect loss of material in the auxiliary feedwater steam piping between the steam supply check valves and each of the three auxiliary feedwater pump turbines. Commitment dates associated with the implementation of this new program are contained in Appendix A.

Preventive Actions

No preventive actions are applicable to this program.

Parameters Monitored or Inspected

The program will monitor the wall thickness of representative piping/fittings in the auxiliary feedwater steam supply headers and the drain lines upstream of the steam traps. The volumetric examination will identify potential effects of inside diameter corrosion due to accumulation of water at the bottom of horizontal run pipes and outside diameter corrosion at areas of contact with the lower section of wet insulation.

Detection of Aging Effects

The aging effect of concern, loss of material due to general and pitting corrosion, will be evident by the reduced wall thickness in the piping/fittings.

Monitoring and Trending

The examination will initially be performed every five years. Piping/fittings thickness measurements will permit calculation of an integrated inside diameter and outside

diameter corrosion rate. Inspection frequency may be adjusted based on corrosion rate to insure that the minimum wall thickness requirements will be maintained.

Acceptance Criteria

Wall thickness measurements greater than minimum values for the component design of record will be acceptable. Wall thickness measurements less than required minimum values will be entered into the corrective action program.

Confirmation Process

Follow-up examinations will be based on the evaluation of the examination results and will be documented in accordance with the corrective action program.

Operating Experience and Demonstration

Ultrasonic and computer aided radiography wall thickness measurement techniques have been performed at Turkey Point for years. These techniques have proven successful in determining wall thickness of piping and other components. Computer aided radiography has been used in the auxiliary feedwater steam supply headers and drain lines. The results of these examinations have detected some areas of localized corrosion in the headers.

This new program will use techniques with demonstrated capability and a proven industry record to measure pipe wall thickness. The examinations will be performed utilizing approved plant procedures and qualified personnel.

Based on the above, the implementation of the Auxiliary Feedwater Steam Piping Inspection Program will provide reasonable assurance that loss of material will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.1.3 EMERGENCY CONTAINMENT COOLERS INSPECTION

As identified in Chapter 3, the Emergency Containment Coolers Inspection is credited for aging management of cooler tubes in Emergency Containment Cooling.

Scope

The Emergency Containment Coolers Inspection is a one-time inspection that will determine the extent of loss of material due to erosion in the emergency containment cooler tubes of Units 3 and 4. A sample of tubes for examination will be selected based on piping geometry and flow conditions that represent those with the greatest susceptibility to erosion. Commitment dates associated with the implementation of this new program are contained in Appendix A.

Preventive Actions

No preventive actions are applicable to this inspection.

Parameters Monitored or Inspected

The inspection will document wall thickness of the emergency containment cooler heat exchanger tubes.

Detection of Aging Effects

The aging effect of concern, loss of material due to erosion, will be detected and sized in accordance with the volumetric technique chosen.

Monitoring and Trending

As stated above, this is a one-time inspection and as such no monitoring and trending is anticipated. The evaluation of the inspection results may result in additional testing, monitoring, and trending.

Acceptance Criteria

The results of the inspection will be evaluated by Turkey Point to verify that the minimum required wall thickness for the emergency containment cooler heat exchanger tubes will be maintained during the period of extended operation.

Confirmation Process

Any follow-up inspection required will be based on the evaluation of the inspection results and will be documented in accordance with the corrective action program.

Operating Experience and Demonstration

This one-time inspection is a new activity that will use techniques with demonstrated capability and a proven industry record to detect wall thickness (loss of material due to erosion). Effective and proven volumetric examination techniques will be selected for use in performing this inspection. This inspection will be performed utilizing approved procedures and qualified personnel.

Based upon the above, the implementation of the Emergency Containment Coolers Inspection will provide reasonable assurance that loss of material due to erosion will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.1.4 FIELD ERECTED TANKS INTERNAL INSPECTION

As discussed in Chapter 3, the Field Erected Tanks Internal Inspection is credited for aging management of field erected tanks in the following systems:

- Auxiliary Feedwater and Condensate Storage
- Feedwater and Blowdown
- Safety Injection

Scope

This is a one-time inspection of the two condensate storage tanks, two refueling water storage tanks, and the shared demineralized water storage tank. The Field Erected Tanks Internal Inspection is credited with managing the aging effect of loss of material due to corrosion of the tanks within the scope. The one-time inspection of selected internal areas, including surface welds, will determine the extent of internal corrosion in the listed tanks. The visual inspection will consist of direct (e.g., divers) or remote (e.g., television cameras, fiber optic scopes, periscopes) means. Commitment dates associated with the implementation of this new program are contained in Appendix A.

Preventive Actions

Internal tank surfaces are coated to reduce corrosion. Coatings minimize corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management.

Parameters Monitored or Inspected

The material condition of the internal surfaces of accessible areas of the tanks will be visually inspected.

Detection of Aging Effects

The presence of corrosion that could lead to loss of material will be determined by visual inspection of the accessible areas of the field erected tanks. Internal surfaces will be examined for evidence of flaking, blistering, peeling, discoloration, pitting, or excessive corrosion.

Monitoring and Trending

As noted above, this is a one-time inspection, therefore, monitoring or trending is not anticipated. Results of the inspection will be evaluated to determine if additional actions are required.

Acceptance Criteria

The results of the one-time inspection will be evaluated. Specific acceptance criteria will be provided in the implementing procedure.

Confirmation Process

Any follow-up inspection required will be based on the evaluation of the inspection results and will be documented in accordance with the corrective action program.

Operating Experience and Demonstration

Visual inspections have been performed at Turkey Point for several years. This technique has proven successful for identifying material defects on the surface of field erected tanks.

This inspection is a new activity that will use techniques with demonstrated capability and a proven industry record to detect corrosion. This inspection will be performed utilizing approved procedures and qualified personnel.

Based upon the above, the implementation of the Field Erected Tanks Internal Inspection will provide reasonable assurance that loss of material due to corrosion will be managed such that the structures and components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.1.5 GALVANIC CORROSION SUSCEPTIBILITY INSPECTION PROGRAM

As identified in Chapter 3, the Galvanic Corrosion Susceptibility Inspection Program is credited for aging management of specific component/commodity groups in the following systems:

Auxiliary Feedwater and Condensate Storage

Chemical and Volume Control

Component Cooling Water

Containment Spray

Control Building Ventilation

Emergency Containment Cooling

Emergency Diesel Generators and Support Systems

Feedwater and Blowdown

Fire Protection

Instrument Air

Normal Containment and Control Rod Drive Mechanism Cooling

Reactor Coolant

Residual Heat Removal

Safety Injection

Spent Fuel Pool Cooling

Turbine Building Ventilation

Waste Disposal

Scope

The Galvanic Corrosion Susceptibility Inspection Program will manage the potential effects of loss of material due to galvanic corrosion on the internal surfaces of susceptible piping and components. Carbon steel components directly coupled to stainless steel components in raw water systems at Turkey Point are the most susceptible to galvanic corrosion. However, baseline examinations will be performed and evaluated to establish if the corrosion mechanism is active in other systems. The program will involve selected one-time inspections (see Monitoring and Trending below) whose results will be utilized to determine the need for additional actions. Commitment dates associated with the implementation of this new program are contained in Appendix A.

Preventive Actions

Components and systems utilize insulating flanges or cathodic protection to minimize galvanic corrosion. The use of insulated flanges and cathodic protection is not credited with the elimination of galvanic corrosion.

Parameters Monitored or Inspected

The program will assess the loss of material due to galvanic corrosion between dissimilar metals in locations determined to represent the most limiting conditions. Selection of the most limiting conditions will be based on high galvanic potential, high cathode/anode area ratio, and high conductivity of the fluid in contact with the materials.

Detection of Aging Effects

Loss of material due to galvanic corrosion will be evident by material loss at the location of the junction between the dissimilar metals. Volumetric examinations or visual inspections will be utilized to address the extent of material loss.

Monitoring and Trending

Inspections will be conducted on a sampling basis. Locations selected for inspection will represent those with the greatest susceptibility for galvanic corrosion (i.e., greatest galvanic potential, high cathode/anode area ratio, and high fluid conductivity). Initial inspection results will be utilized to assess the need for expanded sample locations. Inspection frequency will be determined based on the corrosion rate identified during the initial inspections.

Acceptance Criteria

Wall thickness measurements greater than required minimum wall thickness values for the components will be acceptable. Wall thickness measurements less than required minimum values will be entered into the corrective action program.

Confirmation Process

Any follow-up examination required will be based on the evaluation of the examination results and will be documented in accordance with the corrective action program.

Operating Experience and Demonstration

Visual and volumetric inspection techniques have been used at Turkey Point for years. These techniques have proven successful in determining material condition of components.

This is a new program that will use techniques with demonstrated capability and a proven industry record to monitor material loss due to galvanic corrosion. This examination will be performed utilizing approved procedures and qualified personnel. The inspection techniques used in this program have been previously used to monitor material condition for plant systems.

Based upon the above, the implementation of the Galvanic Corrosion Susceptibility Inspection Program will provide reasonable assurance that loss of material due to galvanic corrosion will be managed such that the systems and components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.1.6 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

As identified in Chapter 3, the Reactor Vessel Internals Inspection Program is credited for aging management of the reactor vessel internals in the Reactor Coolant Systems.

The Reactor Vessel Internals Inspection Program will involve the combination of several activities culminating in the inspection of the Turkey Point Units 3 and 4 reactor vessel internals once for each unit during the 20-year period of extended operation, as described below. This program is intended to supplement the reactor vessel internals inspections required by the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program. Ongoing industry efforts are aimed at characterizing the aging effects associated with the reactor vessel internals. Further understanding of these aging effects will be developed by the industry over time and will provide additional bases for the inspections under this program. Pending results of industry progress with regard to validation of the significance of dimensional changes due to void swelling, the visual examinations described below may be supplemented to incorporate requirements for measurement of critical parts to evaluate potential dimensional changes. Accordingly, an evaluation will be performed to establish the requirements for dimensional verification of critical reactor vessel internals parts as part of the visual examination scope.

The Reactor Vessel Internals Inspection Program consists of two types of examinations, visual and ultrasonic testing. These examinations will manage the aging effects of cracking, reduction in fracture toughness, and loss of mechanical closure integrity. Commitment dates associated with the implementation of this new program are contained in Appendix A.

3.1.6.1 VISUAL EXAMINATION

Scope

This activity will manage the aging effects of cracking due to irradiation assisted stress corrosion (IASCC) and reduction in fracture toughness due to irradiation and thermal embrittlement on accessible parts of the Turkey Point Units 3 and 4 reactor vessel internals. The reactor vessel internals parts susceptible to these aging effects and included in the visual examination scope are accessible areas of the lower core plates and fuel pins, lower support columns, core barrels, baffle/former assemblies, thermal shields, and lower support castings. The program will consist of

VT-1 examinations utilizing remote equipment such as television cameras, fiberoptic scopes, periscopes, etc.

Preventive Actions

There are no practical preventive actions available that will prevent IASCC and reduction in fracture toughness. However, to minimize the potential for IASCC, the concentrations of chlorides, fluorides, and sulfates in the reactor coolant are controlled by implementation of the Chemistry Control Program.

Parameters Monitored or Inspected

This examination monitors the effects of cracking and reduction in fracture toughness on the reactor vessel internals selected parts by the detection and sizing of cracks.

Detection of Aging Effects

IASCC and reduction in fracture toughness of reactor vessel internals selected parts will be detected by performance of VT-1 examinations for the detection of cracks. Cracking is expected to initiate at the surface and, therefore, will be detectable by visual examination.

Monitoring and Trending

The VT-1 examination of selected parts of the reactor vessel internals will be performed one time for each unit during the period of extended operation. Based on the results of each examination, additional examinations and/or repairs will be scheduled.

Acceptance Criteria

Acceptance criteria will be developed prior to the visual examination. Cracks will be evaluated for determination of the need and method of repair.

Confirmation Process

Any follow-up examination will be based on the evaluation of the initial examination results and will be documented in accordance with the corrective action program.

Operating Experience and Demonstration

The remote visual examination proposed by this program utilizing equipment such as television cameras, fiberoptic scopes, periscopes, etc., has been demonstrated previously as an effective method to detect cracking of reactor vessel internals. Similar visual examinations were successfully performed at St. Lucie Unit 1 during the core barrel repair/modification.

3.1.6.2 ULTRASONIC EXAMINATION

Scope

This activity manages the aging effect of loss of mechanical closure integrity on reactor vessel internals baffle/former bolts, barrel/former bolts, and lower support column bolts. The volumetric examination will involve ultrasonic testing on the baffle/former bolts in each unit to supplement the current examination techniques. The results of this examination will be utilized to determine the need for similar examinations of the barrel/former bolts, lower support column bolts, and other reactor vessel internals bolting.

Preventive Actions

There are no practical preventive actions available that will prevent loss of mechanical closure integrity of reactor vessel internals bolting. However, to minimize the potential for loss of mechanical closure integrity due to IASCC, the concentrations of chlorides, fluorides, and sulfates in the reactor coolant are controlled by implementation of the Chemistry Control Program.

Parameters Monitored or Inspected

This examination monitors loss of mechanical closure integrity of the reactor vessel internals bolts by the detection and sizing of cracks.

Detection of Aging Effects

The aging effect of loss of mechanical closure integrity of reactor vessel internals bolting will be detected by performance of ultrasonic examinations.

Monitoring and Trending

The ultrasonic examination of the reactor vessel internals baffle/former bolts will be performed one time during the period of extended operation. Based on the results of the examination, additional examinations and/or repairs will be scheduled.

Acceptance Criteria

The quantity of cracked baffle/former bolts shall be less than the number of bolts that can be damaged without affecting the intended function of the reactor vessel internals. This quantity will be established by evaluation.

Confirmation Process

Any follow-up examination will be based on the evaluation of the initial examination results and will be documented in accordance with the corrective action program.

Operating Experience and Demonstration

The ultrasonic examination methods are proven techniques that have been used in other programs to successfully detect cracking. Ultrasonic examinations have been demonstrated as an effective method of detecting cracking in baffle/former bolting at other Westinghouse plants.

The ultrasonic examinations utilize techniques with a demonstrated capability and a proven industry record to detect cracking. These examinations are performed utilizing approved procedures and qualified personnel.

Based upon the above, the implementation of the Reactor Vessel Internals Inspection Program will provide reasonable assurance that cracking, reduction in fracture toughness, and loss of mechanical closure integrity will be managed such that reactor vessel internals components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.1.7 SMALL BORE CLASS 1 PIPING INSPECTION

As identified in Chapter 3, the Small Bore Class 1 Piping Inspection is credited for aging management of small bore Class 1 piping in the Reactor Coolant Systems.

Scope

The Small Bore Class 1 Piping Inspection will be a one-time inspection of a sample of Class 1 piping less than 4 inches in diameter. Commitment dates associated with the implementation of this new program are contained in Appendix A.

Preventive Actions

No preventive actions are applicable to this inspection.

Parameters Monitored or Inspected

The volumetric technique chosen will permit detection and sizing of significant cracking of small bore Class 1 piping.

Detection of Aging Effects

The aging effect requiring management, cracking, will be detected and sized in accordance with the volumetric technique chosen.

Monitoring and Trending

As noted above, this is a one-time inspection and as such, no monitoring and trending is anticipated. The evaluation of the inspection results may result in additional examinations consistent with ASME Section XI, Subsection IWB. A small sample of the affected welds will be selected for examination based on piping geometry, piping size, and flow conditions.

Acceptance Criteria

Any cracks identified will be evaluated and, if appropriate, entered into the corrective action program.

Confirmation Process

Any follow-up inspection required will be based on the evaluation of the inspection results and will be documented in accordance with the corrective action program.

Operating Experience and Demonstration

This one-time inspection is a new activity, which will use techniques with demonstrated capability and a proven industry record to detect piping weld and base material flaws. Effective and proven volumetric examination techniques will be selected for use in performing this inspection. This inspection will be performed utilizing approved procedures and qualified personnel. Results and recommendations from industry initiatives will be incorporated into the inspection.

Based upon the above, the Small Bore Class 1 Piping Inspection will provide reasonable assurance that cracking in small bore Class 1 piping welds will be managed such that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2 EXISTING AGING MANAGEMENT PROGRAMS

3.2.1 ASME SECTION XI INSERVICE INSPECTION PROGRAMS

The ASME Section XI Inservice Inspection Programs within the scope of license renewal include the following:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program
- ASME Section XI, Subsection IWE Inservice Inspection Program
- ASME Section XI, Subsection IWF Inservice Inspection Program
- ASME Section XI, Subsection IWL Inservice Inspection Program

3.2.1.1 ASME SECTION XI, SUBSECTIONS IWB, IWC, AND IWD INSERVICE INSPECTION PROGRAM

As identified in Chapter 3, the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program is credited for aging management of specific component/commodity groups in the Reactor Coolant Systems.

A request to revise the Turkey Point Unit 3 inservice inspection scope for Class 1 piping to risk informed inservice inspection (RI-ISI) has been submitted to the NRC [Reference B-3]. The revision affects the nondestructive examination (NDE) scope of Class 1 piping currently required by ASME Section XI. Examinations performed are based upon the postulated failure mechanism associated with the piping being inspected. A similar revision request will be submitted for Turkey Point Unit 4 at a later date.

Scope

This program, as defined by the Third Interval Inservice Inspection Program for Turkey Point Nuclear Units 3 and 4 [Reference B-4], is credited with managing the aging effects of cracking, loss of mechanical closure integrity, and loss of material for piping and components. This program provides inspection and examination of accessible components, including welds, pump casings, valve bodies, steam generator tubing, and pressure-retaining bolting.

Inservice inspection requirements may be modified by applicable relief requests and code cases that are approved specifically for each unit. A particular code edition is

applicable for a 120-month interval. Prior to the end of each interval, the program is revised to reflect the update requirements of 10 CFR 50.55a.

Although ASME Section XI, Subsection IWD is included in the scope of this program, this Application does not credit Subsection IWD for managing the effects of aging.

Preventive Actions

There are no specific preventive actions under this program to prevent the effects of aging. Specific actions that serve to limit the effects of aging for Class 1, 2, and 3 piping and components are conservative design, fabrication, construction, inservice inspections, and strict control of chemistry.

Parameters Monitored or Inspected

Inservice examinations include visual inspections, surface examinations, and volumetric examinations in accordance with the requirements of ASME Section XI.

Detection of Aging Effects

The degradation of piping and components is determined by visual, surface, or volumetric examination in accordance with the requirements of ASME Section XI as modified by the Third Interval Inservice Inspection Program for Turkey Point Nuclear Units 3 and 4 [Reference B-4]. Piping and components are examined for evidence of operation-induced flaws using volumetric and surface techniques. The VT-1 visual examination is used to detect cracks, symptoms of wear, corrosion, erosion, or physical damage. VT-2 examinations are conducted to detect evidence of leakage from pressure-retaining components. VT-3 examinations are conducted to determine the general mechanical and structural condition of components and to detect discontinuities and imperfections such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. The extent and frequency of inspections is specified in ASME Section XI as modified in accordance with the Third Interval Inservice Inspection Program for Turkey Point Nuclear Units 3 and 4 [Reference B-4]. The frequency and scope of examinations are sufficient to ensure that the aging effects are detected prior to impacting the component intended functions. The inspection intervals are not restricted by the Code to the current term of operation and are valid for the period of extended operation.

Monitoring and Trending

The frequency and scope of examinations are sufficient to ensure that the aging effects are detected before impacting the component intended functions. Inspections are performed in accordance with the inspection intervals specified by ASME Section XI as modified by the Third Interval Inservice Inspection Program for Turkey Point Nuclear Units 3 and 4 [Reference B-4].

Examinations performed during any inspection interval that reveal flaws or areas of degradation exceeding the acceptance criteria are to be extended to include additional examinations within the same category. When examination results require evaluation of flaws or areas of degradation, the areas are reexamined during subsequent inspection intervals in accordance with the requirements of ASME Section XI.

Records of the inspection program, examination and test procedures, results of activities, examination/test data, and corrective actions taken or recommended are maintained in accordance with the requirements of ASME Section XI, Subsection IWA.

Acceptance Criteria

Acceptance standards for the inservice inspections are identified in ASME Section XI. Relevant indications that are revealed by the inservice inspections may require additional inspections of similar components in accordance with ASME Section XI. Examinations that reveal indications exceeding the acceptance standards are made acceptable by repair, replacement, or evaluation.

Confirmation Process

When areas of degradation are identified, an evaluation is performed to determine if repair or replacement is required. The results are documented in accordance with the corrective action program. Reexaminations are conducted for repaired flaws or areas of degradation to demonstrate that the repairs meet the acceptance standards.

Operating Experience and Demonstration

ASME Section XI provides the rules and requirements for inservice inspection, testing, repair, and replacement of Class 1, 2, and 3 components. Components are

chosen for inspection in accordance with the requirements of Subsections IWB, IWC, and IWD and are inspected using the volumetric, surface, or visual examination methods.

The ASME Section XI inspections are conducted as part of the inservice inspections typically performed during plant refueling outages. The inservice inspection of Class 1, 2, and 3 components and piping has been conducted since initial plant start-up as required by the plant Technical Specifications and 10 CFR 50.55a. These inspections have documented, evaluated, and corrected degraded conditions associated with piping and components inspected under the program.

Implementation of the ASME Section XI program at Turkey Point currently includes over 480 Class 1, Class 2, and Class 3 examinations per unit per ten-year interval. For Class 1 piping, the examinations have yielded only indications of surface anomalies and surface geometry with no indication of fatigue cracking. For Class 2 piping, the only indications have been surface anomalies, acceptable slag inclusions, surface geometry, and fatigue cracking of steam generator feedwater nozzle reducers. The feedwater reducers were replaced and subsequent inspections are being performed in accordance with the requirements of ASME Section XI (see Section 3.2).

Based on the above, the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides reasonable assurance that the aging effects of cracking, loss of mechanical closure integrity, and loss of material will be managed such that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the extended period of operation.

3.2.1.2 ASME SECTION XI, SUBSECTION IWE INSERVICE INSPECTION PROGRAM

As identified in Chapter 3, the ASME Section XI, Subsection IWE Inservice Inspection Program is credited for aging management of specific structural component/commodity groups in the Containments.

Scope

The ASME Section XI, Subsection IWE Inservice Inspection Program is credited with managing the effects of loss of material for containment steel components and change in material properties for elastomers (seals, gaskets, and moisture barriers) associated with containment steel components. The program provides inspection

and examination of accessible surface areas, including surfaces of welds, pressure-retaining bolting, and moisture barriers intended to prevent intrusion of moisture against inaccessible Containment metallic surfaces.

Preventive Actions

Visual inspections are performed to detect loss of material due to general corrosion and change in material properties of elastomers due to embrittlement and permanent set. Carbon steel surfaces are typically coated, in accordance with plant procedures, to reduce the effects of loss of material due to corrosion. In addition, cathodic protection and moisture barriers are used where appropriate to minimize corrosion. However, coatings, cathodic protection, and moisture barriers are not credited in the determination of the aging effects requiring management.

Parameters Monitored or Inspected

The ASME Section XI, Subsection IWE Inservice Inspection Program provides for examination of the following categories as defined by ASME Section XI, Subsection IWE and the Turkey Point Inservice Inspection Program:

- Examination Category E-A, Containment Surfaces
- Examination Category E-C, Containment Surfaces Requiring Augmented Examination
- Examination Category E-D, Seals, Gaskets, and Moisture Barriers
- Examination Category E-G, Pressure-retaining Bolting
- Examination Category E-P, All Pressure-retaining Components

Surface conditions are monitored through visual examinations to determine the existence of corrosion. Surfaces that are inaccessible require an evaluation of the acceptability when conditions in accessible areas exist that could indicate the presence of or result in degradation to such inaccessible areas. Moisture barriers are visually inspected for degradation per Category E-D. Seals and gaskets are pressure tested in accordance with 10 CFR 50, Appendix J, per Category E-P.

Detection of Aging Effects

The presence of corrosion that could lead to loss of material is determined by visual inspection of the steel components. Surfaces are examined for evidence of flaking,

blistering, peeling, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, or other signs of surface irregularities.

All pressure-retaining components, per Category E-P, require leakage testing in accordance with 10 CFR 50, Appendix J, to evaluate the change in material properties for airlocks, seals, and gaskets. Under the inspection plan, 100% of the accessible surfaces are inspected during the inspection intervals as established by the ASME Section XI, Subsection IWE, Inservice Inspection Program.

Monitoring and Trending

In accordance with the requirements of 10 CFR 50.55a, the first-period inspections for Turkey Point are scheduled for completion prior to September 9, 2001. Subsequent inspections are performed in accordance with the inspection intervals specified by ASME Section XI, Subsection IWE, and the ASME Section XI, Subsection IWE Inservice Inspection Program.

Surface areas likely to experience accelerated degradation and aging require augmented examinations and include areas as determined by the ASME Section XI, Subsection IWE Inservice Inspection Program. Identification of the areas subject to augmented examinations will be accomplished in accordance with the corrective action program and monitoring of industry events.

Examinations performed during any inspection interval that reveal flaws or areas of degradation exceeding the acceptance criteria are expanded to include additional examinations within the same category. When examination results require evaluation of flaws or areas of degradation, the area(s) are reexamined during the next inspection interval. Flaws or areas of degradation are documented and evaluated in accordance with the corrective action program and the requirements of the ASME Section XI, Subsection IWE Inservice Inspection Program.

Acceptance Criteria

Examinations and evaluations are performed under the direction of the Responsible Engineer in accordance with the requirements of ASME Section XI, Subsection IWE, and the ASME Section XI, Subsection IWE Inservice Inspection Program. Inspection results are evaluated against the acceptance standards of the program. Inspections that reveal evidence of degradation exceeding the acceptance standards may be subject to additional inspections to determine the nature and extent of the condition.

Confirmation Process

When areas of degradation are identified, an evaluation is performed to determine if repair or replacement is required. The results are documented in accordance with the corrective action program. Reexaminations are conducted for repaired flaws or areas of degradation to demonstrate that the repairs meet the acceptance standards.

Operating Experience and Demonstration

ASME Section XI, Subsection IWE was recently incorporated by reference in 10 CFR 50.55a and, accordingly, the Turkey Point program was developed. Full implementation is scheduled for September 2001. The current inspections of the containment liner are conducted in accordance with the Containment Leak Rate Testing Program and the Maintenance Rule Implementation Program. The inspections performed under these programs were previously documented and evaluated for any degraded conditions associated with the containment liner.

Containment leak-tight verification and visual examination of the steel components that are part of the leak-tight barrier have been conducted at Turkey Point since initial unit startup. Prior to the development of the ASME Section XI, Subsection IWE Inservice Inspection Program, examinations were performed in accordance with 10 CFR 50, Appendix J. Appendix J requires that licensees provide for preoperational and periodic verification, by performing tests of the leak-tight integrity of the Containment, and systems and components that penetrate the Containment.

Approved plant procedures provide the requirements, precautions/limitations, and acceptance criteria for the visual inspection of Unit 3 and Unit 4 accessible interior and exterior Containment surfaces, including the liner plate. Detailed inspections and evaluations are performed as warranted if gross discrepancies are detected. All conditions noted during the inspection of the Containment are documented on inspection reports. The inspection procedures provide general guidelines for inspection in accordance with NEI 94-01, "Industry Guidelines for Implementing Performance-Based Options of 10 CFR Part 50, Appendix J" [Reference B-6].

Material properties for non-metallic components, such as gaskets and seals, change over time and are replaced in accordance with approved plant procedures. The Appendix J tests performed at the Turkey Point units during the years of operation have not shown any loss of intended function of the containment steel components that were attributed to loss of material or other aging effects.

The ASME Section XI, Subsection IWE Inservice Inspection Program incorporates the inspection criteria from the current inspection programs, which are similar to the Subsection IWE requirements.

FPL's Nuclear Quality Assurance Department performed an audit on the program and concluded that the Turkey Point ASME Section XI, Subsection IWE Inservice Inspection Program met the requirements of 10 CFR 50.55a and ASME Section XI, Subsection IWE, for inspection of Class CC metallic liners and pressure retention components.

The NRC Safety Evaluation Report for the Turkey Point Inservice Inspection Program [Reference B- 5] concluded there were no deviations from the regulatory requirements or commitments.

Based on the above, the continued examinations performed under the guidance of the ASME Section XI, Subsection IWE Inservice Inspection Program provide reasonable assurance that the aging effects loss of material and change of material properties for the containment steel components within the scope of license renewal will be managed for the period of extended operation.

3.2.1.3 ASME SECTION XI, SUBSECTION IWF INSERVICE INSPECTION PROGRAM

As identified in Chapter 3, the ASME Section XI, Subsection IWF Inservice Inspection Program is credited for aging management of Class 1, 2, and 3 component supports in the following structures:

- Auxiliary Building
- Containments
- Emergency Diesel Generator Buildings
- Yard Structures

Scope

The ASME Section XI, Subsection IWF Inservice Inspection Program is credited with managing the aging effect of loss of material for Class 1, 2, and 3 component supports. The scope of the Turkey Point program provides inspection and examination of accessible surface areas of these component supports.

Preventive Actions

Carbon steel surfaces are typically coated, in accordance with plant procedures, to reduce the effects of loss of material due to corrosion. Coatings minimize corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management.

Parameters Monitored or Inspected

Class 1, 2, and 3 component supports are examined in accordance with ASME Section XI, Subsection IWF. The ASME Section XI, Subsection IWF Inservice Inspection Program provides for visual examination for general corrosion that could reduce the structural capacity of the component supports.

Detection of Aging Effects

The presence of corrosion that could lead to loss of material is determined by visual inspection of component supports. Surfaces are examined for evidence of flaking, blistering, peeling, discoloration, wear, pitting, corrosion, arc strikes, gouges, surface discontinuities, dents, or other signs of surface irregularities. The extent and frequency of the inspections is in accordance with ASME Section XI, Subsection IWF.

Monitoring and Trending

The selected supports are monitored each inspection period. The program inspects 25% of non-exempt Class 1 piping supports, 15% of Class 2 piping supports, and 10% of Class 3 piping supports, including exposed surfaces of structural bolting. For those component supports within a system that have similar design, function, and service, only one support is examined. Unacceptable supports are subject to corrective measures or evaluation, and are re-examined during the next inspection period.

Acceptance Criteria

Acceptance standards for the examination and evaluation of supports are provided in ASME Section XI, Subsection IWF. A condition observed during a visual examination that requires supplemental examination, corrective measures, repair, replacement, or analytical evaluation is categorized as a relevant condition and is not considered acceptable.

Confirmation Process

Component supports subject to corrective measures in accordance with ASME Section XI, Subsection IWF, are re-examined during the next inspection period and documented in accordance with the corrective action program. If additional corrective measures are not required, the examinations revert back to the original schedule of successive inspection intervals.

Operating Experience and Demonstration

The ASME Section XI, Subsection IWF, inspections are conducted as part of the inservice inspections typically during plant refueling outages. The inspection of Class 1, 2, and 3 component supports has been conducted since initial plant startup as required by Technical Specifications.

ASME Section XI provides the rules and requirements for inservice inspection testing, repair, and replacement of Class 1, 2, and 3 component supports. The ASME Section XI, Subsection IWF Inservice Inspection Program applies to Class 1, 2, and 3 component supports (piping supports and supports other than piping supports). These supports are chosen for inspection in accordance with the requirements of ASME Section XI, Subsection IWF, and shall be inspected using visual examination methods.

The visual examinations of Class 1, 2, and 3 component supports look for deformations or structural degradations, corrosion, and other conditions that could affect the intended function of the support. All conditions noted during the inspection of component supports, whether or not they are considered to require further review, are documented on inspection reports.

The NRC Safety Evaluation Report for the Turkey Point Inservice Inspection Program [Reference B-5] concluded that there were no deviations from regulatory requirements or commitments. The FPL Nuclear Division Quality Assurance Department performed an audit of the Inservice Inspection Program. This audit concluded that the program was complete and in compliance with the requirements of the ASME Code, Section XI, and applicable commitments.

Based on the above, the continued examinations performed under the ASME Section XI, Subsection IWF Inservice Inspection Program provide reasonable assurance that the aging effect loss of material for the Class 1, 2, and 3 components

and piping supports within the scope of license renewal will be managed for the period of extended operation.

3.2.1.4 ASME SECTION XI, SUBSECTION IWL INSERVICE INSPECTION PROGRAM

As identified in Chapter 3, the ASME Section XI, Subsection IWL Inservice Inspection Program is credited for aging management of the post-tensioning system structural components in the Containments.

Scope

The ASME Section XI, Subsection IWL Inservice Inspection Program is credited for managing the aging effect of loss of material for the containment post-tensioning system structural components. The scope of the program provides for inspection of tendon wires and tendon anchorage hardware surfaces for loss of material, as well as a confirmatory program for measurement of tendons for loss of prestress.

Preventive Actions

A layer of non-structural low strength concrete protects the top filler caps of the containment vertical tendons. This layer of concrete prevents the intrusion of rainwater under the caps and any subsequent corrosion of the tendon assemblies. This concrete also prevents the accumulation of rainwater behind the ring girder. Additionally, metal caps are installed over tendon anchorages.

Additionally, all the metallic components (such as reinforcing bars, liner plate, and tendon anchorages) are interconnected to an impressed current cathodic protection system to prevent galvanic corrosion. The cathodic protection system is a non-safety related system that supports the protection of the steel components from corrosion. However, this protective system is not credited in the determination of the aging effects requiring management.

Parameters Monitored or Inspected

In accordance with ASME Section XI, Subsection IWL, unbonded post-tensioning system components are examined. These components consist of tendons, wires or strand, anchorage hardware and surrounding concrete, corrosion protection medium, and free water. Surface conditions are monitored through visual examinations to determine the extent of corrosion or concrete degradation around anchorage locations. Prestress forces are measured for sample tendons to

determine loss of prestressing force. Tension tests are performed on removed wire or strand samples to examine for corrosion and mechanical damage.

Detection of Aging Effects

The presence of age-related degradation is determined by visual inspection or by measurement. Tendon anchorage hardware is examined for corrosion. A select number of tendons are completely detensioned and a sample wire from each group of tendons is examined for the presence of corrosion and tested to verify ultimate strength. Tendon anchorage hardware and concrete surfaces are examined for corrosion protection medium leakage and the tendon caps are examined for deformation.

Monitoring and Trending

In accordance with 10 CFR 50.55a, the first-period inspections for Turkey Point are scheduled for completion by September 9, 2001. Subsequent tendon inspections are performed in accordance with the inspection intervals specified by ASME Section XI, Subsection IWL.

Tendons examined are selected on a random basis among the tendons that have not been examined during previous inspections. A sample of each tendon types, in accordance with ASME Section XI, Subsection IWL, is examined during each inspection interval. Reduced sample size may be used if the applicable criteria of ASME Section XI, Subsection IWL, are met during each of the earlier inspections.

The tendons inspected under the ASME Section XI, Subsection IWL Inservice Inspection Program provide results to evaluate for loss of material and loss of prestress. These previous tendon surveillance records are maintained and the results documented for future use. Prestress limits are calculated and tabulated for each group of tendons examined. All surveillance results are reviewed in order to ensure that any recommendations are considered for incorporation into the surveillance program.

Acceptance Criteria

Examinations and evaluations are performed under the direction of the Responsible Engineer in accordance with the requirements of ASME Section XI, Subsection IWL, and the ASME Section XI, Subsection IWL Inservice Inspection Program. The Responsible Engineer evaluates the inspection results and determines whether

analysis, repair, or additional inspections or testing are required. Inspection results are evaluated against the acceptance standards in accordance with ASME Section XI, Subsection IWL, and the ASME Section XI, Subsection IWL Inservice Inspection Program.

Confirmation Process

When areas of degradation are identified, an evaluation is performed to determine if repair or replacement is required. The results are documented in accordance with the corrective action program. Following repair or replacement activities, a containment pressure test would be conducted, if required, in accordance with the ASME Section XI, Subsection IWL Inservice Inspection Program, to confirm the adequacy of the corrective action. In addition, repaired areas are reexamined.

Operating Experience and Demonstration

ASME Section XI, Subsection IWL, was recently adopted by 10 CFR 50.55a and, accordingly, the Turkey Point program was developed. Implementation is scheduled prior to September 2001. Previous inspections of the tendons and tendon anchorages were conducted in accordance with Technical Specifications, the UFSAR, and plant procedures. The inspections performed under these programs were documented and evaluated any degraded conditions associated with the post-tensioning system. The ASME Section XI, Subsection IWL Inservice Inspection Program incorporates the inspection criteria from the current post-tensioning system inspection programs, which are similar to Subsection IWL.

The containment tendon examination program has been conducted since initial unit startups at 5-year intervals. The containment tendon surveillance examination requirements incorporated the general criteria and requirements of Regulatory Guide 1.35, “Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments” [Reference B-7].

This tendon surveillance program included provisions for determining that a tendon retains a lift-off force equal to or greater than its predicted lower limit. The surveillance program requires visual inspections to detect the presence of water; examination of all anchorage components for indications of corrosion, pitting, cracking, distortion, or damage; examination of surrounding concrete; and examination of removed tendon wire for signs of gross corrosion or damage.

An example of the overall effectiveness of the tendon inspection program is supported by the results of the Turkey Point 20th year tendon surveillance. The measured lift-off forces for a number of randomly selected surveillance tendons were below the predicted lower limit. Condition Reports and a Licensee Event Report were issued. In accordance with the Technical Specifications, engineering evaluations were prepared and concluded that the lower than expected tendon lift-off forces were caused by greater than expected tendon wire relaxation losses due to average tendon temperatures higher than originally considered. To accommodate the increased prestress losses, a license amendment was submitted and approved to reduce the Containment design pressure from 59 psig to 55 psig, and a Containment re-analysis was performed to determine the new minimum required prestress forces to maintain Turkey Point licensing basis requirements. The results of the re-analysis are provided in the UFSAR, Section 5.1.3.

The ASME Section XI, Subsection IWL Inservice Inspection Program incorporates all of the inspection criteria and guidelines of the previous tendon inspection program attributes and is implemented using existing plant procedures.

Based upon the above, the continued implementation of the tendon inspections under the ASME Section XI, Subsection IWL Inservice Inspection Program provides reasonable assurance that the aging effects (loss of material and loss of prestress) will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.2 BORAFLEX SURVEILLANCE PROGRAM

As identified in Chapter 3, the Boraflex Surveillance Program is credited for aging management of the spent fuel racks associated with Spent Fuel Storage and Handling.

Scope

The Boraflex Surveillance Program is credited with managing the aging effect of change in material properties for the Boraflex material in the spent fuel storage racks for Units 3 and 4.

The program will be enhanced to provide for density testing (or other approved testing method) of the encapsulated Boraflex material in the spent fuel storage racks. Commitment dates associated with the enhancement of this program are contained in Appendix A.

Preventive Actions

Turkey Point is not aware of any preventive actions that can be taken to mitigate loss of boron carbide and silica release since the polymer matrix tends to break down over time due to the convective aqueous environment of the spent fuel pool. Continued monitoring of the Boraflex will assure that the 5% subcriticality margin will be maintained.

Parameters Monitored or Inspected

The Boraflex Surveillance Program seeks to determine the amount of degradation of the Boraflex material. The current program, consisting of blackness testing, confirms the in-service Boraflex panel performance data in terms of gap formation, gap distribution, and gap size. In addition, tracking of the spent fuel pool silica levels provides a qualitative indication of boron carbide loss from the panels. The enhanced Boraflex Surveillance Program will include checking the density (or other approved methods) of the Boraflex to ascertain the physical loss of boron carbide.

Detection of Aging Effects

The presence of silica, which is periodically monitored, in the spent fuel pool water is a physical sign of the aging effect occurring in the Boraflex material. The enhanced

Boraflex Surveillance Program will determine the amount of degradation of the Boraflex material.

Monitoring and Trending

Shrinkage, gaps, and density will be monitored during scheduled Boraflex surveillance testing. Subsequent Boraflex tests will be scheduled following evaluation of the measured results. Trends will be established following implementation of the enhanced Boraflex Surveillance Program.

Acceptance Criteria

Acceptance standards for Boraflex degradation are controlled by the assumptions in the criticality analysis. The results of each surveillance are used to evaluate the impact on the assumptions in the criticality analysis to assure that the 5% subcriticality margin will be maintained.

Confirmation Process

When areas of degradation are identified, an evaluation is performed to determine if corrective action is required. Subsequent testing will ensure the effectiveness of any corrective actions.

Operating Experience and Demonstration

The current Boraflex Surveillance Program was initiated following installation of high density spent fuel storage racks at Turkey Point. It is known that degradation of the Boraflex is occurring, as evidenced by silica accumulation in the spent fuel pool water.

Boraflex neutron attenuation (blackness) testing has been performed on a test frequency of once every five years on specific Boraflex panels in either Unit 3 or Unit 4 spent fuel pools. Evaluations of the test results have demonstrated that the required Technical Specification subcritical margin for storage of fuel in the spent fuel pools has been met. Turkey Point's response to NRC Generic Letter 96-04 [Reference B-9] assessed the capability of the Boraflex to maintain the five percent subcriticality margin and provided remedies for long-term Boraflex degradation.

Based upon the above, the continued implementation of the Boraflex Surveillance Program provides reasonable assurance that the effects of aging will be adequately managed such that the components within the scope of license renewal will perform

their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.3 BORIC ACID WASTAGE SURVEILLANCE PROGRAM

As identified in Chapter 3, the Boric Acid Wastage Surveillance Program is credited for aging management of specific cast iron, carbon steel, and low alloy steel component/commodity groups in the following systems and structures:

SYSTEMS

Auxiliary Building Ventilation

Chemical and Volume Control

Component Cooling Water

Containment Isolation

Containment Post Accident Monitoring and Control

Containment Spray

Electrical/ I&C Components

Emergency Containment Cooling

Emergency Containment Filtration

Feedwater and Blowdown

Fire Protection

Instrument Air

Intake Cooling Water

Main Steam and Turbine Generators

Normal Containment and Control Rod Drive Mechanism Cooling

Primary Water Makeup

Reactor Coolant

Residual Heat Removal

Safety Injection

Sample

Spent Fuel Pool Cooling

Waste Disposal

STRUCTURES

Auxiliary Building

Containments

Spent Fuel Storage and Handling

Yard Structures

Scope

The Boric Acid Wastage Surveillance Program manages the effects of loss of material and loss of mechanical closure integrity due to aggressive chemical attack of cast iron, carbon steel, and low alloy steel components and structural components including bolting. The program encompasses mechanical closures (e.g., bolted connections, valve packing, pump seals) in the above systems and structures and components containing, or exposed to, borated water. The program utilizes

systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of the pressure boundary or structural integrity of components, supports, or structures. This program includes electrical structural components (enclosures, cable trays, conduit, etc.) in proximity to borated water systems.

Some systems outside Containment (i.e., Spent Fuel Pool Cooling and portions of Waste Disposal associated with containment integrity) are currently inspected under other existing programs. The scope of the Boric Acid Wastage Surveillance Program will be enhanced to include these systems and components. Commitment dates associated with the enhancement of this program are contained in Appendix A.

Preventive Actions

The removal of concentrated boric acid and boric acid residue and the elimination of boric acid leakage mitigates corrosion by minimizing the exposure of the susceptible material to the corrosive environment.

Parameters Monitored or Inspected

Boric acid residue and borated water leakage are directly related to the degradation of components. The program monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage as required by NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants” [Reference B-8], including guidelines for locating small leaks, conducting examinations, and performing evaluations.

Detection of Aging Effects

Degradation of the component due to boric acid corrosion cannot occur without leakage of coolant containing boric acid. Conditions leading to boric acid corrosion, such as crystal buildup, and evidence of moisture are readily detectable by visual inspections. Visual inspections are performed on external surfaces in accordance with plant procedures.

Monitoring and Trending

Leakage calculations are performed each shift. Visual inspection of systems inside Containment is conducted if unidentified leakage rates exceed 0.5 gpm. During

each refueling, inspections of systems inside Containment are performed. Every 18 months inspections of borated water systems outside Containment are performed.

Acceptance Criteria

All identified cases of boric acid leakage are either corrected or evaluated.

Confirmation Process

Follow-up testing is performed to confirm satisfactory completion of corrective actions. For system leakage, this includes a visual inspection of the repaired components.

Operating Experience and Demonstration

The Boric Acid Wastage Surveillance Program was originally implemented as a result of boric acid leaks experienced at Turkey Point and NRC Generic Letter 88-05. This program addresses the Generic Letter requirements, including: 1) detection of the principal locations where coolant leaks smaller than allowable Technical Specification limits could cause degradation of the pressure boundary; 2) methods for conducting examinations that are integrated into ASME Code VT-2 inspections conducted during system pressure tests; and 3) corrective actions to prevent recurrences of this type of leakage. The conservative philosophy established within the program has been successful in managing loss of material due to boric acid wastage. It has provided for timely identification of leakage and implementation of corrective actions. Since establishing the program, there have been no instances of boric acid corrosion that have impacted license renewal system intended functions.

Based upon the above, the continued implementation of the Boric Acid Wastage Surveillance Program provides reasonable assurance that the aging effects (loss of material and loss of mechanical closure integrity) will be managed such that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.4 CHEMISTRY CONTROL PROGRAM

As identified in Chapter 3, the Chemistry Control Program is credited for aging management of specific component/commodity groups in the following systems and structures:

SYSTEMS

Auxiliary Feedwater and Condensate Storage

Chemical and Volume Control

Component Cooling Water

Containment Post Accident Monitoring and Control

Containment Spray

Control Building Ventilation

Emergency Containment Cooling

Emergency Containment Filtration

Feedwater and Blowdown

Emergency Diesel Generators and Support Systems

Main Steam and Turbine Generators

Normal Containment and Control Rod Drive Mechanism Cooling

Primary Water Makeup

Reactor Coolant

Residual Heat Removal

Safety Injection

Sample

Spent Fuel Pool Cooling

Turbine Building Ventilation

Waste Disposal

STRUCTURES

Spent Fuel Storage and Handling (structural components exposed to fluid)

Scope

The Chemistry Control Program is credited for managing the aging effects of loss of material, cracking, and fouling buildup for the internal surfaces of primary and secondary systems and structures. The program includes sampling activities and

analysis for treated water - primary, treated water - borated, treated water - secondary, treated water, and fuel oil.

Preventive Actions

No preventive actions are applicable to this program.

Parameters Monitored or Inspected

The parameters monitored by the Chemistry Control Program for the purposes of aging management are chloride, fluoride, sulfate, oxygen, biocide, corrosion inhibitor, and water content.

Detection of Aging Effects

The aging effects of concern (i.e., loss of material, cracking, and fouling) are minimized or prevented by controlling the chemical species that cause the underlying aging mechanisms that result in the aging effects. Alternatively, chemical agents, such as corrosion inhibitors and biocides, are introduced to prevent certain aging mechanisms. The chemistry parameters are measured utilizing standard proven industry techniques. The aging mechanisms that can be minimized or prevented by the Chemistry Control Program include general corrosion, pitting corrosion, crevice corrosion, microbiologically influenced corrosion, graphitic corrosion, stress corrosion cracking, intergranular attack, corrosion fouling, and fouling caused by microbiologically influenced corrosion.

Monitoring and Trending

Monitoring and trending requirements for all parameters controlled by the Chemistry Control Program are included in appropriate plant procedures.

Acceptance Criteria

The acceptance criteria for the chemistry parameters required to be monitored and controlled are in accordance with the Nuclear Chemistry Parameters Manual, Technical Specifications, and appropriate plant procedures.

Confirmation Process

Follow-up testing is performed to confirm satisfactory completion of corrective action.

Operating Experience and Demonstration

The Chemistry Control Program is an ongoing program at Turkey Point that considers the best practices of industry organizations, vendors, utilities, and water treatment experts. The program provides assurance that the fluid environments to which piping and associated components are exposed will minimize corrosion. This is accomplished through effective monitoring of key parameters at established frequencies with well-defined acceptance criteria. The chemistry analyses are governed by the plant Quality Control Program to assure accurate results. Chemistry data are monitored for trends that might be indicative of an underlying operational problem.

The overall effectiveness of the program is supported by the excellent operating experience for systems, structures, and components that are influenced by the program. No chemistry-related degradation has resulted in loss of component intended functions on any systems for which the fluid chemistry is actively controlled. A review of plant condition reports indicates that Turkey Point Units 3 and 4 have not experienced any Level 3 excursions as defined by the Electric Power Research Institute's water chemistry guidelines. The program has been subject to periodic internal and external assessment activities that help to maintain highly effective chemistry control, and facilitate continuous improvement.

Based on the above, the continued implementation of the Chemistry Control Program provides reasonable assurance that the aging effects (loss of material, cracking, and fouling) will be managed such that components and systems within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.5 CONTAINMENT SPRAY SYSTEM PIPING INSPECTION PROGRAM

As identified in Chapter 3, the Containment Spray System Piping Inspection Program is credited for aging management of selected piping and fittings in Containment Spray.

Scope

This Containment Spray System Piping Inspection Program manages the aging effect of loss of material due to general, crevice, and pitting corrosion on the internal surfaces of carbon steel piping/fittings and valves wetted by boric acid in the containment spray headers.

Preventive Actions

Surveillance procedures require the closure of a second isolation valve in the containment spray headers when the pumps are started for testing.

Parameters Monitored or Inspected

The program monitors the wall thickness of selected piping/fittings in the spray headers within the Containments.

Detection of Aging Effects

Ultrasonic thickness measurement is utilized for this examination. The aging effect of concern, loss of material, is evident by the reduced wall thickness in the piping/fittings being examined.

Monitoring and Trending

The examination initially is performed each refueling outage. The piping/fittings thickness measurements permit calculation of a corrosion rate. Inspection frequency may be adjusted based on corrosion rate to insure that minimum wall thickness requirements for the pipe are maintained. If evaluation of the inspection results indicates that loss of material due to corrosion is not occurring, the Containment Spray System Piping Inspection Program may be discontinued.

Acceptance Criteria

Wall thickness measurements greater than minimum wall thickness values for the component design of record are acceptable. Wall thickness measurements less than the minimum required values are entered into the corrective action program.

Confirmation Process

Any follow-up examination required is based on the evaluation of the examination results and is documented in accordance with the corrective action program.

Operating Experience and Demonstration

Ultrasonic thickness measurements have been performed for several years. The technique has proven successful at determining the wall thickness of piping/fittings and other components.

This is an existing program at Turkey Point that uses a technique with demonstrated capability and a proven industry record to measure wall thickness. This examination is performed utilizing approved procedures and qualified personnel. The ultrasonic thickness measurement technique has been previously used to measure the wall thickness in the containment spray system spray headers and other plant systems. The results of these examinations have detected some areas of localized corrosion in the headers, and the results are documented.

Based on the above, the continued implementation of the Containment Spray System Piping Inspection Program provides reasonable assurance that loss of material will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.6 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification Program is credited for ensuring the qualified life of electrical and I&C components within the scope of 10 CFR 50.49 (see Sections 2.5, 3.7, and 4.4 of this Application).

Scope

The program includes the environmentally qualified devices that are within the scope of 10 CFR 50.49.

Preventive Actions

The program includes preventive actions required to maintain the qualification time period for the environmentally qualified devices.

Parameters Monitored or Inspected

The program establishes an aging limit (qualified life) for each installed device. The installed life of each device is monitored and appropriate actions (replacement, refurbishment, or requalification) are taken before the aging limit is exceeded.

Detection of Aging Effects

The program does not require the detection of aging effects for equipment while in service since effects are maintained within established acceptable limits by the Environmental Qualification Program actions. When the qualified life is less than the plant license period, the program requires replacement, refurbishment, or requalification of the component prior to the end of its qualified life. When unexpected adverse effects are identified during operational or maintenance activities, they are evaluated to determine the root cause and significance in accordance with approved procedures.

Monitoring and Trending

The installed life of each environmentally qualified device is monitored and appropriate actions (replacement, refurbishment, or requalification) are taken before the aging limit is exceeded. The program does not require monitoring or trending of condition or performance parameters of equipment while in service to manage the effects of aging.

Acceptance Criteria

The program requires replacement, refurbishment, or requalification prior to exceeding the life limit (qualified life) of each installed device.

Confirmation Process

Administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B, that will insure adequacy of corrective actions.

Operating Experience and Demonstration

The Environmental Qualification Program is an ongoing program at Turkey Point that considers the best practices of industry organizations, vendors, and utilities. The program provides assurance that the environments to which installed devices are exposed will not exceed the qualified lives associated with the devices. This is accomplished through effective monitoring of key parameters (temperature, radiation) at established frequencies with well-defined acceptance criteria. The Environmental Qualification Program is governed by the Quality Control Program to assure accurate results.

The overall effectiveness of the Environmental Qualification Program is supported by the excellent operating experience for systems, structures, and components that are influenced by the program. No environmental qualification related degradation has resulted in loss of component intended functions on any systems. The program has been subject to periodic internal and external assessment activities that help to maintain highly effective control and facilitate continuous improvement.

Based on the above, the Environmental Qualification Program is an effective program for managing the effects of aging to ensure that the components within the scope of license renewal will continue to perform their intended function consistent with the current licensing basis for the period of extended operation.

3.2.7 FATIGUE MONITORING PROGRAM

As identified in Subsection 4.3.1, the Fatigue Monitoring Program is a confirmatory program for fatigue of Class 1 components in the Reactor Coolant Systems.

Scope

The Fatigue Monitoring Program tracks design cycles to ensure that Units 3 and 4 Reactor Coolant System components remain within their design fatigue limits. The specific fatigue analyses validated by this monitoring program include those for the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines.

Preventive Actions

The program utilizes the systematic counting of design cycles to ensure that component design fatigue usage limits are not reached.

Parameters Monitored or Inspected

The parameters monitored by the program for confirmation purposes are the design cycles consistent with the Reactor Coolant System Class 1 component design analyses.

Detection of Aging Effects

The Fatigue Monitoring Program assures that the component design fatigue usage limits are not reached.

Monitoring and Trending

Administrative procedures provide the methodology for logging design cycles. Guidance is provided as design cycle limits are approached.

Acceptance Criteria

The maximum allowable design cycles are specified in the plant administrative procedures and are listed in Chapter 4 of the UFSAR Supplement provided in Appendix A of this Application.

Confirmation Process

In order to stay within fatigue design limits, plant procedures require administrative action should the actual cycle count reach 80% of any design cycle limit.

Operating Experience and Demonstration

The Fatigue Monitoring Program has been an ongoing program at Turkey Point since initial unit startups and has evolved over many years of plant operation. As demonstrated in Subsection 4.3.1 of this Application, the number of design cycles considered in the current licensing basis fatigue analyses is sufficiently conservative to account for not only the current licensing basis term, but the extended period of operation as well. Confirmation will be accomplished by continuation of the Fatigue Monitoring Program.

The FPL Quality Assurance Department performed a surveillance/audit of this program to assess the implementation of activities associated with maintaining the Reactor Coolant System plant components within design cycle limits. This audit identified no deficiencies.

In addition, an independent assessment of the Fatigue Monitoring Program concluded the administrative procedure accurately identifies and classifies plant design cycles, and provides an effective and consistent method for categorizing, counting, and tracking design cycles. The assessment concluded that the program maintains in-depth information for each design cycle counted. The assessment also concluded that design cycle severity bounds actual plant operation.

Based upon the above, the continued implementation of the Fatigue Monitoring Program provides reasonable assurance that Reactor Coolant Systems components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.8 FIRE PROTECTION PROGRAM

As identified in Chapter 3, the Fire Protection Program is credited for aging management of specific component/commodity groups associated with Fire Protection and Fire Rated Assemblies.

Scope

The Fire Protection Program manages the aging effects of loss of material, cracking, and fouling for the components/piping of the Fire Protection System and Fire Rated Assemblies. Additionally, this program manages the aging effects of loss of material, loss of seal, cracking, and erosion for structures and structural components associated with fire protection.

The scope of the Fire Protection Program will be enhanced to include inspection of additional components. Commitment dates associated with the enhancement of this program are contained in Appendix A.

Preventive Actions

Many fire protection components are provided with a protective coating to minimize the potential for external corrosion. Coatings minimize corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management.

Parameters Monitored or Inspected

Surface conditions are monitored visually to determine the extent of external material degradation. Visual examination will detect loss of material due to general, crevice, and pitting corrosion; and loss of seal or cracking due to embrittlement. Internal conditions are monitored via leakage, flow, and pressure testing. Internal loss of material (due to general, crevice, and pitting corrosion; microbiologically influenced corrosion; and selective leaching) and blockage due to fouling can be detected by changes in flow or pressure, leakage, or by evidence of excessive corrosion products during flushing of the system.

Detection of Aging Effects

The detection of degradation on external surfaces is determined by visual examination. Surfaces of components and structures are examined for damage, deterioration, leakage, or other forms of corrosion.

Functional testing and flushing of the systems clears away internal scale, debris and other foreign material that could lead to blockage/obstruction of the system. Flow and pressure tests verify system integrity. Visual examinations of breached portions of the system also verify unobstructed flow and integrity of the piping/components.

Monitoring and Trending

The degradation found as a result of inspection/testing of the systems/components is addressed by the Fire Protection Program procedures. The evaluation of the inspection/testing results may result in additional testing, monitoring, and trending.

Acceptance Criteria

The results of the inspection/testing will be evaluated in accordance with the acceptance criteria in the appropriate fire protection procedure(s). Parameters required to be monitored and controlled are listed in the applicable documents.

Confirmation Process

Administrative procedures require verification that the affected fire protection feature be restored to normal configuration and that post maintenance testing, if required, be performed prior to returning the equipment to service.

Operating Experience and Demonstration

The Fire Protection Program has been an ongoing program at Turkey Point. The program was enhanced by implementation of 10 CFR 50, Appendix R, and has evolved over many years of plant operation. The program incorporates the best practices recommended by National Fire Protection Association (NFPA) and Nuclear Electric Insurance Limited (NEIL) and is approved by the NRC.

The overall effectiveness of the program is demonstrated by the excellent operating experience of systems, structures, and components that are included in the Fire Protection Program. The program has been subjected to periodic internal assessment activities. These activities, as well as other external assessments, help

to maintain highly effective fire protection control and facilitate continuous improvement through monitoring industry initiatives and trends in the area of aging control.

Based upon the above, the continued implementation of the Fire Protection Program provides reasonable assurance that the aging effects (loss of seal, loss of material, cracking, and fouling) will be managed such that components/commodity groups within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.9 FLOW ACCELERATED CORROSION PROGRAM

As identified in Chapter 3, the Flow Accelerated Corrosion Program is credited for aging management of selected piping and fittings in the following systems:

- Feedwater and Blowdown
- Main Steam and Turbine Generators

Scope

The Flow Accelerated Corrosion Program manages the aging effect of loss of material due to flow accelerated corrosion. The program predicts, detects, monitors, and mitigates flow accelerated corrosion wear in high energy carbon steel piping associated with Main Steam and Turbine Generators and Feedwater and Blowdown. The program includes analysis and limited baseline inspection, determination of the extent of thinning, repair/replacement as appropriate, and performance of follow-up inspections.

This program will be enhanced to address internal and external loss of material of steam trap lines due to flow accelerated corrosion and general corrosion, respectively. Commitment dates associated with the enhancement of this program are contained in Appendix A.

Preventive Actions

The rate of flow accelerated corrosion is affected by piping material, geometry and hydrodynamic conditions, and operating conditions, such as temperature, pH, steam quality, operating hours, and dissolved oxygen content. The susceptibility to flow accelerated corrosion is reduced by maintaining high water quality.

Parameters Monitored or Inspected

The program monitors the effects of flow accelerated corrosion in piping by measuring wall thickness using nondestructive examination, and by performing analytical evaluations. The inspection program stipulates visual, ultrasonic, or radiographic testing of susceptible locations based on operating conditions. For each location outside the acceptance guidelines, the inspection sample is expanded based on approved guidelines and engineering judgment. Analytical models are used to predict flow accelerated corrosion in piping systems based on specific plant data, including material and hydrodynamic and operating conditions. External piping

wall loss of steam trap lines due to general corrosion will also be maintained by this program. Based on this, volumetric NDE techniques will be used on these lines.

Detection of Aging Effects

Aging degradation of piping and fittings occurs by wall thinning. The methods of inspections are visual, ultrasonic, and computer aided radiography. The extent and schedules of inspections ensure detection of wall thinning before the loss of intended function of this piping.

Monitoring and Trending

Inspections and analytical evaluations monitor and trend wall thinning. The inspection schedule is determined based on the remaining service life that is recalculated after each inspection. If degradation is detected such that the wall thickness is less than the minimum allowed wall thickness, additional examinations are performed in adjacent areas to bound the thinning.

External piping wall loss of steam trap lines due to general corrosion will also be monitored by this program based on the volumetric nondestructive examination techniques utilized for these lines. Inspection frequency for external corrosion monitoring will be developed based on the initial inspection results.

Acceptance Criteria

Inspection results are used to calculate the number of refueling or operating cycles remaining before the component reaches its minimum allowable wall thickness. If calculations indicate that an area will reach its minimum allowable wall thickness before the next inspection interval, the component is repaired, replaced, or reevaluated.

Confirmation Process

Follow-up inspections are scheduled based on the remaining service life that is recalculated after each inspection. Additionally, post maintenance testing is conducted after repairs.

Operating Experience and Demonstration

Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems, two-phase piping in extraction steam lines, and moisture

separation reheater and feedwater heater drains, as identified in NRC generic communications. The Flow Accelerated Corrosion Program was originally implemented in response to NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning" [Reference B-10]. The conservative philosophy established within the program has been successful in managing the loss of material due to erosion/corrosion. Various sections of the Main Steam and Turbine Generators and Feedwater and Blowdown piping are periodically examined using nondestructive examination to determine the effects of flow accelerated corrosion. Results are evaluated and piping is either repaired or replaced as required. Branch connections are examined as plant/industry experience warrants.

Ultrasonic examinations have identified piping wall thickness below the established screening criteria. These degradations were documented in accordance with the corrective action program and resulted in repair, replacement, or subsequent inspection of the piping.

This program has been reviewed by the NRC during several inspections with no deviations or violations identified. FPL Quality Assurance surveillances and reviews have been performed of the program with no deficiencies identified.

Based on the above, the continued implementation of the Flow Accelerated Corrosion Program provides reasonable assurance that the aging effects of flow accelerated corrosion will be managed such that Main Steam and Turbine Generators and Feedwater and Blowdown components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.10 INTAKE COOLING WATER SYSTEM INSPECTION PROGRAM

As identified in Chapter 3, the Intake Cooling Water System Inspection Program is credited for aging management of specific component/commodity groups in the following systems:

- Component Cooling Water
- Intake Cooling Water

Scope

NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment" [Reference B-11], recommended the implementation of an on-going program of surveillance and control techniques to significantly reduce flow blockage caused by biofouling, corrosion, erosion, protective coating failures, stress corrosion cracking, and silting problems in systems and components supplied by the Intake Cooling Water System. The Intake Cooling Water System Inspection Program was developed in response to this generic letter and addresses the aging effects of loss of material due to various corrosion mechanisms, stress corrosion cracking, and fouling due to macro-organisms for those components subject to raw water (i.e., salt water) conditions. The program utilizes performance testing and evaluations, systematic inspections, leakage evaluations, and corrective actions to ensure that loss of material, cracking, or biological fouling does not lead to loss of component intended functions.

This program will be enhanced to improve documentation of scope and frequency of the intake cooling water piping crawl-through inspections and component cooling water heat exchanger tube integrity inspections. Commitment dates associated with the enhancement of this program are contained in Appendix A.

Preventive Actions

The Intake Cooling Water System Inspection Program is preventive in nature since it provides for the periodic inspection and maintenance of internal linings protecting the intake cooling water piping and components, and also employs component cooling water heat exchanger performance monitoring, testing, and periodic tube inspections. Maintenance of the internal piping/component linings minimizes the potential loss of material due to corrosion that could impact the pressure boundary intended function. Performance monitoring and testing; channel head, tube sheet,

and anode inspections; and tube examinations of component cooling water heat exchangers provide for early identification of internal fouling and tube degradation that can impact heat transfer and pressure boundary intended functions. External coatings are applied to portions of the Intake Cooling Water System to minimize corrosion. Coatings minimize corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management.

Parameters Monitored or Inspected

Internal Piping/Component Inspections – Surface conditions of piping/components and their internal linings are visually inspected for degradation. Wall thickness measurements are taken when deemed necessary.

Performance monitoring, testing, and tube inspections of component cooling water heat exchangers - Pressures, temperatures, and flows are measured as part of periodic performance testing of the component cooling water heat exchangers to verify heat transfer capability. This testing is supplemented by routine monitoring of differential temperatures across the heat exchanger during operation. Tube integrity of the component cooling water heat exchangers is monitored by periodic nondestructive examination (e.g., eddy current testing) to ensure early detection of aging effects.

Detection of Aging Effects

Internal Piping/Component Inspections – Visual examination of the piping/ components and their internal linings is performed. Additional nondestructive testing may be utilized to measure surface condition and the extent of wall thinning based on the evaluation of the examination results and as documented in accordance with the corrective action program.

Performance monitoring, testing, and tube inspections of component cooling water heat exchangers – Periodic performance testing and monitoring are conducted to provide for early identification of fouling or degraded conditions that could impact the ability of the component cooling water heat exchangers to perform their intended function. Periodic tube inspections and cleaning are performed to assure heat exchanger performance and integrity.

Monitoring and Trending

Internal Piping/Component Inspections – Inspections and frequencies are in accordance with commitments under Generic Letter 89-13. Internal piping/component inspections are performed periodically during refueling outages. Inspection frequencies are adjusted based upon experience and assure the timely detection of aging effects.

Performance monitoring, testing, and tube inspections of component cooling water heat exchangers – Online monitoring of system parameters is used to provide an indication of flow blockage. Heat transfer testing results are documented and reviewed in plant procedures. The heat transfer capability is trended to ensure that the component cooling water heat exchangers satisfy safety analysis requirements. component cooling water heat exchanger tube condition is determined by eddy current testing and documented accordingly. Heat exchanger tube cleanings, tube replacements, or other corrective actions are implemented as required.

This program will be enhanced to improve documentation of scope and frequency of the intake cooling water piping crawl-through inspections and component cooling water heat exchanger tube integrity inspections.

Acceptance Criteria

Internal Piping/Component Inspections – Biological fouling is considered undesirable and is removed or reduced during the inspection process. When required by procedure, wall thickness values are determined and evaluated.

Performance monitoring, testing, and tube inspections of component cooling water heat exchangers – Acceptance criteria are provided to ensure that the design basis heat transfer capability is maintained and to direct when component cooling water heat exchanger cleaning and inspection are required. Differential pressure criteria guidelines are provided to ensure that the intake cooling water design basis flow rate is maintained and to identify when backflushing or cleaning of the intake cooling water basket strainers is required.

Confirmation Process

Follow-up monitoring is performed to confirm satisfactory completion of corrective actions. This monitoring activity is verified as part of the system surveillances for determining minimum Intake Cooling Water System flows for cleaning basket

strainers and for evaluating component cooling water heat exchanger performance, or following piping/tube inspections.

Operating Experience and Demonstration

The existing Intake Cooling Water System Inspection Program has been an ongoing formalized inspection program at Turkey Point. The program was formally implemented as a result of Generic Letter 89-13, which recommended monitoring of service water systems to ensure that they would perform their safety-related function and based on experiences of biological fouling and corrosion throughout the industry. The conservative philosophy established within the program has been successful in managing the loss of material due to corrosion and fouling of the component cooling water heat exchangers. This program has been effective in maintaining acceptable component cooling water heat exchanger performance and addressing biological fouling of strainers and heat exchangers. Various sections of the intake cooling water piping, basket strainers, and heat exchangers are periodically examined using nondestructive examination to determine the effects of corrosion and biological fouling. Results are evaluated and components are either repaired or replaced as required.

The program has been reviewed by the NRC during several inspections with no significant deviations or violations identified. FPL Quality Assurance surveillances and reviews have been performed with no significant deficiencies identified. Procedures and practices were enhanced as a result of the recommendations provided from these inspections.

Metallurgical analysis of removed component cooling water heat exchanger tubes, in 1991 and 1994, indicated that stress corrosion cracking was a potential root cause and, as a result, zinc anodes were installed and are inspected during tube cleaning. Analysis in 1996 of additional component cooling water heat exchanger tubes indicated that inside pitting was a potential failure mechanism and, as a result, a less abrasive cleaning tool was recommended. Both of these corrective actions have proven to be effective in minimizing repetitive failures.

A review of the Maintenance Rule database for the Intake Cooling Water and the Component Cooling Water Systems shows that the current aging management programs have supported system availability above the required performance criteria for the period from May 1996 through March 2000. There have been no

functional failures attributed to aging or biological fouling of pressure-retaining components during that period.

Based on the above, the continued implementation of the Intake Cooling Water System Inspection Program provides reasonable assurance that the aging effects of corrosion and biological fouling will be managed, such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.11 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

As identified in Chapter 3, the Periodic Surveillance and Preventive Maintenance Program is credited for aging management of specific component/commodity groups in the following systems and structures:

SYSTEMS

Chemical and Volume Control

Control Building Ventilation

Emergency Containment Filtration

Emergency Diesel Generators and Support Systems

Fire Protection

Instrument Air

Intake Cooling Water

Residual Heat Removal

Turbine Building Ventilation

Waste Disposal

STRUCTURES

Auxiliary Building

Emergency Diesel Generator Buildings

Turbine Building

Yard Structures

Scope

The Periodic Surveillance and Preventive Maintenance Program is credited for managing the aging effects of loss of material, cracking, fouling, loss of seal, and embrittlement for structures, systems, and components within the scope of license renewal. This program provides for visual inspection of selected surfaces of specific components and structural components, or alternatively their replacement/ refurbishment during the performance of periodic surveillance and preventive maintenance activities. The program also includes leak inspections of limited portions of the Chemical and Volume Control Systems.

This program will be enhanced to address the scope of specific inspections and their documentation. Commitment dates associated with the enhancement of this program are contained in Appendix A.

Preventive Actions

No preventive actions are applicable to the aging effects being managed by this program.

Parameters Monitored or Inspected

Surface conditions of systems, structures, and components are monitored, through visual inspections, for corrosion, fouling, or in some cases leakage, during the performance of periodic maintenance. Based on inspection results, refurbishment is performed as required. For some equipment, periodic replacement is performed on a specified frequency.

This program will be enhanced with regard to the scope of specific inspections and their documentation.

Detection of Aging Effects

The aging effects of concern, loss of material, cracking, fouling, loss of seal, and embrittlement, will be detected by visual inspection of external surfaces for evidence of corrosion, cracking, leakage, or coating damage. For some equipment, aging effects are addressed by periodic replacement in lieu of visual inspection and refurbishment.

Monitoring and Trending

System, structure, and component inspections are performed periodically during preventive maintenance or surveillance activities. Alternatively, some components are replaced on a specified frequency. Inspection and replacement frequencies are adjusted as necessary based on the results of these activities and industry experience.

Acceptance Criteria

Acceptance criteria and guidelines are provided in the implementing procedures for the inspections, refurbishments, and replacements, as applicable.

Confirmation Process

Periodic inspection, refurbishment, and replacement activities performed under this program ensure that component and structure degradation is corrected in accordance with the corrective action program.

Operating Experience and Demonstration

The Periodic Surveillance and Preventive Maintenance Program is an established program at Turkey Point and has proven effective at maintaining the material condition of systems, structures, and components and detecting unsatisfactory conditions. The effectiveness of the program is supported by improved system, structure, and component material conditions and reliability, documented by internal and external industry assessments. The Periodic Surveillance and Preventive Maintenance Program is subject to periodic assessments to ensure effectiveness and continuous improvement.

Based on the above, the continued implementation of the Periodic Surveillance and Preventive Maintenance Program provides reasonable assurance that the aging effects of loss of material, cracking, fouling, loss of seal, and embrittlement will be managed, such that the components and structural components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.12 REACTOR VESSEL HEAD ALLOY 600 PENETRATION INSPECTION PROGRAM

As identified in Chapter 3, the Reactor Vessel Head Alloy 600 Penetration Inspection Program is credited for aging management of the reactor vessels in the Reactor Coolant Systems.

Scope

The Reactor Vessel Head Alloy 600 Penetration Inspection Program encompasses the Turkey Point Units 3 and 4 reactor vessel head Alloy 600 penetrations that are part of the Reactor Coolant System pressure boundary. This program manages the aging effect of cracking due to primary water stress corrosion. The program includes a one-time volumetric examination of selected Unit 4 reactor vessel head penetrations to detect crack initiation of reactor vessel head penetrations. Visual examination of the Units 3 and 4 reactor vessel head external surfaces during outages and the Boric Acid Wastage Surveillance Program are also utilized to manage cracking.

Preventive Actions

No actions are presently known that would prevent primary water stress corrosion cracking.

Parameters Monitored or Inspected

The program monitors the aging effect of primary water stress corrosion cracking on the intended function of the reactor vessel head penetrations by the detection of cracks and reactor coolant leakage.

Detection of Aging Effects

As noted above, a one-time volumetric examination of the inside diameter of selected penetrations of the Turkey Point Unit 4 reactor vessel head will be performed. Penetrations selected have been identified as being most susceptible to primary water stress corrosion cracking based on default values for yield strength. The results of this examination will be utilized to determine if additional examinations are required. At this time, it is anticipated that the volumetric examination would be an eddy current examination.

Visual inspection of the reactor vessel head to penetration area in accordance with the existing Boric Acid Wastage Surveillance Program is performed to detect any through wall coolant leakage from the reactor vessel head as evidenced by steam, boric acid crystals, or other indirect symptoms of fluid escape.

Monitoring and Trending

The volumetric examination will be performed prior to a 75% through wall flaw that is conservatively predicted to occur using the reference probability equivalent to D. C. Cook Unit 2 based on the Dominion Engineering CIRSE model. Follow up examinations will be determined based on the results of the initial examination and available industry data. The visual inspections of the reactor vessel head to penetration area are performed in accordance with the Boric Acid Wastage Surveillance Program.

Acceptance Criteria

The acceptance criteria for the volumetric examinations for identified flaws will be developed using approved fracture mechanics methods and industry or plant-specific data. Evaluations would consider the stresses at the flaw location and industry-developed crack propagation rates, if the flaw is to be left in service, prior to implementing any corrective action. The acceptance criterion for visual inspections is no reactor vessel head pressure boundary leakage.

Confirmation Process

For the reactor vessel head penetrations, confirmation will include inspection of the repaired/replaced nozzle and pressure boundary verification in accordance with the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Boric Acid Wastage Surveillance Program. Additional testing/examinations will be performed if required by the corrective action program.

Operating Experience and Demonstration

Turkey Point has been an active participant in Westinghouse Owners Group, Electric Power Research Institute, and Nuclear Energy Institute initiatives regarding cracking of Alloy 600 reactor vessel head penetrations. The Reactor Vessel Head Alloy 600 Penetration Inspection Program was created in response to NRC Generic Letter 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations” [Reference B-12]. This program has proven experience

in addressing the concerns and requirements of the generic letter. The reactor vessel heads are currently subject to visual inspection for leakage. To date, Turkey Point has performed visual inspections for leakage on the top of the Turkey Point Units 3 and 4 reactor vessel heads as part of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Boric Acid Wastage Surveillance Program. No evidence of leakage from the Alloy 600 reactor vessel head penetrations has been identified. Volumetric examinations have been performed at several plants, as identified in NEI Letter, "Response to NRC Request for Additional Information on Generic Letter 97-01, Project Number 689" [Reference B-13], and have been effective at detecting primary water stress corrosion cracking.

The NRC has concluded, in their letter dated January 27, 2000 [Reference B-14], that the Turkey Point Reactor Vessel Head Alloy 600 Penetration Inspection Program provides an acceptable basis for evaluating reactor vessel head penetrations.

Based on the above, the continued implementation of the Reactor Vessel Head Alloy 600 Penetration Inspection Program provides reasonable assurance that Reactor Coolant Systems components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.13 REACTOR VESSEL INTEGRITY PROGRAM

As identified in Chapter 3, the Reactor Vessel Integrity Program is credited for aging management of the reactor vessels in the Reactor Coolant Systems.

The Reactor Vessel Integrity Program, which manages reactor vessel irradiation embrittlement, encompasses the following subprograms:

- Reactor Vessel Surveillance Capsule Removal and Evaluation
- Fluence and Uncertainty Calculations
- Monitoring Effective Full Power Years
- Pressure-Temperature Limit Curves

The program documentation will be enhanced to integrate all aspects of the Reactor Vessel Integrity Program. Commitment dates associated with the enhancement of this program are contained in Appendix A.

3.2.13.1 REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL AND EVALUATION

Scope

This subprogram manages the aging effect of reduction in fracture toughness on the reactor vessel materials (beltline forgings and circumferential welds) due to neutron irradiation embrittlement by performing Charpy V-notch and tensile tests on the reactor vessel irradiated specimens.

Preventive Actions

This is a monitoring program, as such, preventive actions are not required.

Parameters Monitored or Inspected

Monitored parameters include fracture toughness and tensile strength as measured by Charpy V-notch and tensile tests for irradiated specimens of reactor vessel forging and weld materials. Additionally, accumulated neutron fluence is monitored utilizing surveillance capsule dosimetry.

Detection of Aging Effects

Lower fracture toughness values than those predicted by extrapolation of historical empirical data provide indications of unexpected accelerated aging of the reactor vessel materials. Fracture toughness values are determined using calculations of vessel fluence and empirical results from Charpy V-notch testing of irradiated specimens.

Monitoring and Trending

Empirical material fracture toughness and accumulated neutron fluence data are obtained from the vessel irradiated specimen surveillance. This data and the trend curves from NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [Reference B-15], provide the basis for the value for reference temperature for nil-ductility transition (RT_{NDT}) and for determining reactor vessel heatup and cooldown limits. These data are monitored and trended to ensure continuing reactor vessel integrity. The surveillance capsule withdrawal schedule is specified in Chapter 4 of the UFSAR Supplement provided in Appendix A of this Application. Turkey Point has sufficient surveillance capsules for the extended period of operation. Future decisions concerning the frequency of withdrawal of surveillance capsules will be based on changes in fuel type or fuel loading pattern.

Acceptance Criteria

Values of RT_{NDT} are calculated based on test results and compared with Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [Reference B-15], trend curves. Data that fall outside of the ±20% at the 1-sigma level require further evaluation. The reference temperature for pressurized thermal shock (RT_{PTS}) values must also be within the screening criteria of 10 CFR 50.61.

Confirmation Process

Periodic testing of the vessel irradiated specimens provides advance indication of future material deterioration. Present testing can be used to validate the accuracy of previous predictions.

Operating Experience and Demonstration

The Reactor Vessel Surveillance Capsule Removal and Evaluation Subprogram is NRC approved, meets the requirements of 10 CFR 50, Appendix H, and has been in

effect since the initial plant startup. This subprogram has been updated over the years and has provided experience in addressing reduction in fracture toughness. Turkey Point Units 3 and 4 pressure-temperature limit curves have been updated using results from the vessel surveillance capsule specimen evaluations. Turkey Point Units 3 and 4 have been evaluated to have values for RT_{PTS} that are within the acceptance criteria of 10 CFR 50.61.

3.2.13.2 FLUENCE AND UNCERTAINTY CALCULATIONS

Scope

This subprogram provides an accurate prediction of the reactor vessel accumulated fast neutron fluence values at the reactor vessel beltline forgings and circumferential welds.

Preventive Actions

This is a monitoring program, as such, preventive actions are not required.

Parameters Monitored or Inspected

The monitored parameters are the reactor vessel accumulated neutron fluence values, which are predicted based on analytical models meeting the requirements of Draft NRC Regulatory Guide DG-1053, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” [Reference B-16], and are benchmarked using dosimetry results that are available from the Reactor Vessel Surveillance Capsule Removal and Evaluation Subprogram. Note that in the past, benchmarking has been supplemented by Draft NRC Regulatory Guide DG-1053 cavity (ex-vessel) dosimetry.

Detection of Aging Effects

Accumulated fluence values in excess of predicted values can result in lower fracture toughness values in reactor vessel materials due to irradiation embrittlement. The potential for these effects is determined using neutron calculations of vessel fluence, empirical results from Charpy V-notch tests of irradiated specimens, and capsule dosimetry in accordance with the Reactor Vessel Surveillance Capsule Removal and Evaluation Subprogram.

Monitoring and Trending

Neutron fluence and uncertainty calculations are performed to predict the accumulated fast neutron fluence. These calculations are verified using dosimetry results that are available from the Reactor Vessel Surveillance Capsule Removal and Evaluation Subprogram, as supplemented by the cavity (ex-vessel) dosimetry. The frequency of updating fluence and uncertainty calculations may change as additional data are obtained. Changes in fuel type or fuel loading pattern may also change the frequency of surveillance capsule withdrawal and the performance of neutron fluence and uncertainty calculations.

Acceptance Criteria

The results of the fluence uncertainty calculations are to be within the NRC suggested limit of ±20%. Calculated fluence values for fast neutrons (above 1.0 MeV) are compared with measured values. This methodology represents a continuous validation process to ensure that no biases have been introduced and that the uncertainties remain comparable to the reference benchmarks.

Confirmation Process

The analytical predictions of reactor vessel fast neutron fluence are validated using dosimeter data from the irradiated specimens. Cavity (ex-vessel) dosimetry may also be used to supplement surveillance capsule data. Present data of neutron fluence can be used to evaluate the accuracy of previous predictions.

Operating Experience and Demonstration

The neutron fluence and uncertainty calculations for Turkey Point Units 3 and 4 have been performed in accordance with the guidelines of the Draft Regulatory Guide DG-1053 and validated using data obtained from the capsule dosimetry. The results of the fluence uncertainty values are to be within the NRC-suggested limit of ±20%. This has been validated by the comparison of the calculated fluence values with measurement values. This methodology represents a continuous validation process to ensure that no biases have been introduced, and that the uncertainties remain comparable to the reference benchmarks.

3.2.13.3 MONITORING EFFECTIVE FULL POWER YEARS

Scope

This subprogram accurately monitors and tabulates the accumulated operating time experienced by the reactor vessels to ensure that the pressure-temperature limit curves and end-of-life reference temperatures are not exceeded.

Preventive Actions

This is a monitoring program, as such, preventive actions are not required.

Parameters Monitored or Inspected

The monitored parameters are the reactor vessels' equivalent time at full power in effective full power years.

Detection of Aging Effects

Effective full power year calculations are utilized for the prediction of total accumulated fast neutron fluence and the determination of the reduction in fracture toughness of reactor vessel critical materials.

Monitoring and Trending

This subprogram monitors the accumulated reactor vessel effective full power years to be used in predicting the accumulated fast neutron fluence. Each Turkey Point unit is monitored to determine the effective full power years of operation. These data are used to validate the applicability of the pressure-temperature limit curves for the next operating cycle.

Acceptance Criteria

Calculated effective full power years shall not exceed the Technical Specification limit for the validity of the pressure-temperature limit curves.

Confirmation Process

The effective full power years of plant operation are based on reactor vessel incore power readings. Effective full power year values are determined by comparing the burnup to the thermal power calculated burnup. Data are collected for both Turkey

Point reactor vessels. The effective full power year calculations are verified each fuel cycle in accordance with the FPL Quality Assurance Program.

Operating Experience and Demonstration

The effective full power year values are determined by comparing the fuel burnup to the thermal power calculated burnup. The fuel burnup comparisons have been found to be within the expected accuracy.

3.2.13.4 PRESSURE-TEMPERATURE LIMIT CURVES

Scope

This subprogram provides pressure-temperature limit curves for the Turkey Point Units 3 and 4 reactor vessels to establish the Reactor Coolant System operating limits.

Preventive Actions

Pressure-temperature limit curves are provided to prevent or minimize the potential of damaging the reactor vessel materials. The curves are included in the Technical Specifications and applicable operating procedures.

Parameters Monitored or Inspected

The pressure-temperature limit curves specify maximum allowable pressure as a function of Reactor Coolant System temperature. Reactor Coolant System pressures and temperatures at Turkey Point are maintained within these limits.

Detection of Aging Effects

The pressure-temperature limit curves are not provided for the detection of aging effects but rather to prevent or minimize potential for damage to the reactor vessel materials.

Monitoring and Trending

The pressure-temperature limit curves are valid for a period expressed in effective full power years. These curves shall be updated prior to exceeding the effective full power years for which they are valid. The time period for updating pressure-temperature limit curves may change if conditions such as changes in fuel type or fuel loading pattern occur.

Acceptance Criteria

NRC-approved pressure-temperature limit curves must be in place for continued plant operation.

Confirmation Process

The pressure-temperature limit curves are verified in accordance with the FPL Quality Assurance Program. These pressure-temperature limit curves are NRC approved prior to use and validated using data obtained from the surveillance capsule specimens.

Operating Experience and Demonstration

Turkey Point Units 3 and 4 utilize pressure-temperature limit curves that have been updated using the results of data obtained from the surveillance capsule specimens. The pressure-temperature limit curves have been developed utilizing an industry methodology that has been approved by the NRC. The pressure-temperature limit curves provide sufficient operating margin while preventing or minimizing the potential for damage to the reactor vessel materials.

Based on the above, the continued implementation of the Reactor Vessel Integrity Program provides an effective program for managing the aging effect of reduction in fracture toughness on the reactor vessel such that the components within the scope of license renewal will continue to perform their intended functions in accordance with the current licensing basis for the period of extended operation.

3.2.14 STEAM GENERATOR INTEGRITY PROGRAM

As specified in Chapter 3, the Steam Generator Integrity Program is credited for aging management of the steam generators in the Reactor Coolant Systems.

Scope

The Steam Generator Integrity Program ensures that steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The program is structured to meet NEI 97-06, "Steam Generator Program Guidelines" [Reference B-17]. The program manages the aging effects of cracking and loss of material and includes the following essential elements:

- Inspection of steam generator tubing and tube plugs
- Steam generator secondary-side integrity inspections
- Tube integrity assessment
- Assessment of degradation mechanisms
- Primary-to-secondary leakage monitoring
- Primary and secondary chemistry control
- Sludge lancing
- Maintenance and repairs
- Foreign material exclusion

Preventive Actions

Preventive measures include primary and secondary chemistry control and sludge lancing. Primary and secondary chemistry control prevents cracking of steam generator tubes, as described in the steam generator aging management review. Sludge lancing is performed to prevent outside diameter pitting of steam generator tubing, which is associated with oxidizing conditions in the sludge piles.

Parameters Monitored or Inspected

The steam generator tube volumetric inspection technique detects flaw size and depth, or alternatively, remaining sound tube wall thickness. Primary-to-secondary leakage is monitored to verify tube integrity during plant operation. Steam generator tube integrity is assessed in accordance with the performance criteria provided in

NEI 97-06 [Reference B-17]. The performance criteria include structural, accident-induced leakage, and operational leakage limits. Inspection activities also monitor for leakage from the tube plugs.

Detection of Aging Effects

The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws do not exceed established performance criteria. Problems with tube inspection -- e.g., failure to detect some flaws, uncertainty in flaw sizing, inaccuracy in flaw location, and inability to detect some cracks at locations with dents -- are considered in the Steam Generator Integrity Program. The extent and schedule of the inspections prescribed by the program are designed to ensure timely detection and replacement of leaking plugs. Detection of primary-to-secondary leakage during plant operation will identify flaw propagation caused by the aging mechanisms.

Monitoring and Trending

Required inspection intervals based on Technical Specification requirements are expected to provide timely detection of cracking, pitting, and wear. Required inspection intervals are intended to provide for timely detection of tube plug leakage. Daily monitoring of primary-to-secondary leakage will identify degradation of steam generator tubing.

Acceptance Criteria

Tubes are removed from service in accordance with the requirements of the Technical Specifications and the Steam Generator Integrity Program. Any tube plug leakage detected requires tube plug replacement. Identified primary-to-secondary leakage is compared with the limits allowed by the Technical Specifications.

Confirmation Process

Administrative procedures require follow-up testing and examinations to verify steam generator integrity prior to returning the steam generator to service.

Operating Experience and Demonstration

The Steam Generator Integrity Program has been effective in ensuring the timely detection and correction of the aging effects of cracking and loss of material in

steam generator tubes. Tube plug cracking appears to have been related to susceptible heats of material and improper heat treatment.

The Steam Generator Integrity Program considers the guidance provided in NEI 97-06 [Reference B-17], which has undergone extensive industry and NRC review. This program is all-inclusive in managing steam generator tube bundle and internals degradation.

The Steam Generator Integrity Program has been reviewed by the NRC during several inspections and no deviations or violations have been identified. Quality Assurance surveillances and reviews have been performed with no deficiencies identified.

The current steam generator inspection activities have been evaluated against industry recommendations provided by EPRI and Westinghouse. The overall effectiveness of the program is supported by the excellent steam generator operating experience and favorable inspection results.

Based on the above, the continued implementation of the Steam Generator Integrity Program provides reasonable assurance that the aging effects will be managed such that the steam generator components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.15 SYSTEMS AND STRUCTURES MONITORING PROGRAM

As identified in Chapter 3, the Systems and Structures Monitoring Program is credited for aging management of specific component/commodity groups in the following systems and structures:

SYSTEMS

Auxiliary Building Ventilation

Auxiliary Feedwater and Condensate Storage

Chemical and Volume Control

Component Cooling Water

Containment Isolation

Containment Post Accident Monitoring and Control

Containment Spray

Control Building Ventilation

Emergency Containment Cooling

Emergency Diesel Generators and Support Systems

Feedwater and Blowdown

Instrument Air

Intake Cooling Water

Main Steam and Turbine Generators

Reactor Coolant

Residual Heat Removal

Safety Injection

Sample

Spent Fuel Pool Cooling

Turbine Building Ventilation

Waste Disposal

STRUCTURES

Auxiliary Building

Containments

Control Building

Diesel Driven Fire Pump Enclosure

Discharge Structure

Electrical Penetration Rooms

Emergency Diesel Generator Buildings

Fire Protection Monitoring Station

Intake Structure

Main Steam and Feedwater Platforms

Plant Vent Stack

Spent Fuel Storage and Handling

Turbine Building

Turbine Gantry Cranes

STRUCTURES (continued)

Yard Structures

Scope

The Systems and Structures Monitoring Program manages the aging effects of loss of material, cracking, fouling, loss of seal, and change in material properties for selected systems, structures, and components within the scope of license renewal. The program provides for visual inspection and examination of accessible surfaces of specific systems, structures, and components, including welds and bolting.

Aging management of structural components that are inaccessible for inspection is accomplished by inspecting accessible structural components with similar materials and environments for aging effects that may be indicative of aging effects for inaccessible structural components. For example, rust bleeding on an accessible surface of a concrete structure may be indicative of corrosion of inaccessible reinforcing steel embedded in the concrete.

This program will be enhanced by restructuring it to address inspection requirements to manage certain aging effects in accordance with 10 CFR 54, modifying the scope of specific inspections, and improving documentation requirements. Commitment dates associated with the enhancement of this program are contained in Appendix A.

Preventive Actions

External surfaces of carbon steel and cast iron valves, piping, and fittings, and specific stainless steel piping welds are coated to minimize corrosion and surfaces of steel structures and supports are also coated to minimize corrosion. Coatings minimize corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management.

Parameters Monitored or Inspected

Surface conditions of structures, system components/piping (including those exposed to a wetted environment), and supports are monitored through visual examinations to determine the existence of external corrosion and internal corrosion of certain ventilation equipment. Flexible connections are monitored for cracking due to embrittlement and ventilation heat exchangers are monitored for fouling. External surfaces of concrete are monitored through visual examination for exposed

rebar, extensive rust bleeding, cracks that exhibit rust bleeding, and cracking of block walls and building roof seals. Leakage inspections of valves, piping, and fittings at limited locations of the Intake Cooling Water and Waste Disposal Systems are utilized to detect the presence of internal corrosion. Additionally, visual inspection of external surfaces of certain ventilation systems is used to assess internal system conditions. Inspection of protective coatings on specific stainless steel piping welds in outdoor locations will be performed to determine coating degradation. Inspection of weatherproofing material for deterioration is performed.

This program will be restructured and some enhancements to the scope of specific inspections and their documentation will be implemented.

Detection of Aging Effects

The aging effects of loss of material, crack initiation, fouling, loss of seal, and change in material properties are detected by visual inspection of external surfaces (including internal surfaces of certain ventilation equipment) for evidence of corrosion, cracking, leakage, fouling, or coatings damage.

Monitoring and Trending

Detailed structural and system/equipment material condition inspections are performed in accordance with approved plant procedures. The results of the visual inspections for systems, structures, and components are documented. The frequency of the inspections may be adjusted, as necessary, based on inspection results and industry experience. For insulated piping, a small sample of the sections of systems exposed to a wetted environment will be selected for inspection on the basis of piping geometry and potential exposure to rain or other conditions that could result in wetting of the insulation (e.g., chilled water systems).

Acceptance Criteria

The results of the inspections and testing are evaluated in accordance with the acceptance criteria in the appropriate corrective action and administrative procedures.

Confirmation Process

Degradations are evaluated and entered into the corrective action program.

Operating Experience and Demonstration

Systems and piping/component support material condition inspections have been successfully performed at Turkey Point since the mid 1980s. The inspection requirements in support of the Maintenance Rule have been in effect since 1996, and have proven effective at maintaining systems/structures material condition and detecting unsatisfactory conditions that have resulted in effective corrective actions being taken.

The Systems and Structures Monitoring Program has been an ongoing program at Turkey Point and has been enhanced over the years to include the best practices recommended by the Institute of Nuclear Power Operations and other industry guidance. Additionally, the Systems and Structures Monitoring Program will continue to support implementation of the NRC Maintenance Rule (10 CFR 50.65).

The effectiveness of the Systems and Structures Monitoring Program is supported by the improved system and structure material conditions, documented by internal as well as external assessments of the last several years. Additionally, the Systems and Structures Monitoring Program is the subject of periodic internal and external assessments to insure effectiveness and continued improvement.

Based upon the above, the continued implementation of the Systems and Structures Monitoring Program provides reasonable assurance that the aging effects (loss of material, crack initiation, fouling, loss of seal, and change in material properties) will be managed such that systems and structures within the scope of License Renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.2.16 THIMBLE TUBE INSPECTION PROGRAM

As identified in Chapter 3, the Thimble Tube Inspection Program is credited for aging management of Unit 3 reactor vessel thimble tube N-05 in the Reactor Coolant Systems. The program was originally performed on all in-service thimble tubes for Units 3 and 4.

Scope

Based on previous inspections and the Time-Limited Aging Analysis results presented in Subsection 4.7.1, only thimble tube N-05 on Unit 3 will require inspection for the period of extended operation. The Thimble Tube Inspection Program manages the aging effect of loss of material due to fretting wear. The program utilizes eddy current test inspections to determine thimble tube wall thickness and predict wear rates for the early identification of potential thimble tube failure.

Preventive Actions

The thimble tube wall thickness obtained from eddy current test inspections will be used to predict thimble tube wear rates and enable corrective action before the thimble tube fails.

Parameters Monitored or Inspected

Thimble tube wall thickness.

Detection of Aging Effects

Thimble tube wall thinning will be determined utilizing eddy current testing inspections.

Monitoring and Trending

An eddy current test inspection will be performed on Unit 3 thimble tube N-05 in accordance with plant procedures. The data obtained from this inspection will be evaluated and the need for additional inspections will be determined.

Acceptance Criteria

The acceptance criteria for thimble tube wall thinning is less than 70% wall loss at the end of life.

Confirmation Process

Follow-up testing is performed to confirm satisfactory completion of corrective actions, if needed.

Operating Experience and Demonstration

The Thimble Tube Inspection Program was created and implemented in response to NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors" [Reference B-18]. This program has proven experience in addressing the requirements of the bulletin.

Wear of the thimble tubes is detected by the eddy current inspection. This technique is well known in the industry and has been used to detect imperfections in thimble tubes and other component tubing, such as steam generators, heat exchangers, etc. The eddy current inspection technique has been used to detect wall thinning on the thimble tubes with satisfactory results. The eddy current test inspection is performed in accordance with approved plant procedures by qualified personnel.

Based on the above, the implementation of the Thimble Tube Inspection Program provides reasonable assurance that the aging effects will be managed such that the thimble tubes will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.0 REFERENCES

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B-2 NEI 95-10, Revision 1, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," Nuclear Energy Institute, January 2000.

B-3 FPL Letter, L-2000-010 to NRC, Turkey Point Unit 3, Docket No. 50-250, "Risk Informed Inservice Inspection Program," January 19, 2000.

B-4 Third Interval Inservice Inspection Program for Turkey Point Nuclear Units 3 and 4, ISI-PTN-3/4 Program, Revision 3, Turkey Point Nuclear Power Plants, Florida Power & Light Company, March 31, 1995.

B-5 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to the Third Ten-Year Interval Inservice Inspection Program Plan and Associated Requests for Relief," U. S. Nuclear Regulatory Commission, March 31, 1995.

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B-10 Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," U.S. Nuclear Regulatory Commission, May 1989.

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B-12 Generic Letter 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," U.S. Nuclear Regulatory Commission, April 1997.

B-13 David J. Modeen, (NEI) letter to Gus C. Lainas (NRC), “Response to NRC Request for Additional Information on Generic Letter 97-01, Project Number 689,” December 11, 1998.

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B-18 Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors,” U.S. Nuclear Regulatory Commission, July 26, 1988.

APPENDIX C

PROCESS FOR IDENTIFYING AGING EFFECTS REQUIRING MANAGEMENT FOR NON-CLASS 1 COMPONENTS

TABLE OF CONTENTS

1.0 INTRODUCTION

FPL utilized industry guidance developed by the B&W Owners Group for determining aging effects requiring management for the non-Class 1 components and steel in fluids associated with structural components. The guidance was reviewed for applicability, and tailored to address Turkey Point Units 3 and 4 materials and environments and to incorporate specific aging mechanisms/effects based upon plant experience (i.e., lessons learned). This appendix summarizes the process for identification of aging effects requiring management for non-Class 1 components.

The potential aging effects evaluated include the following:

- loss of material
- cracking
- fouling
- loss of mechanical closure integrity
- reduction in fracture toughness
- distortion/plastic deformation

Internal operating environments evaluated are:

- treated water – primary
- treated water – secondary
- treated water – borated
- treated water
- raw water – cooling canals
- raw water – city water
- raw water – floor drainage
- air/gas
- fuel oil
- lubricating oil

External operating environments evaluated are:

- outdoor
- indoor – not air conditioned
- indoor – air conditioned
- containment air
- borated water leaks
- buried (above ground water elevation)
- embedded/encased (in concrete)

For components that are submerged, the applicable internal environment listed above is specified.

Where wetted conditions exist (e.g., due to condensation), the environment is annotated accordingly.

Note: Other than borated water leaks, fluid leakage is not considered in the aging management review process. Fluid leakage is considered an event-driven condition and not a normal operating condition. As noted in Christopher I. Grimes (NRC) letter to Douglas J. Walters (NEI) dated June 5, 1998, aging effects from abnormal events need not be postulated for license renewal [Reference C-1].

2.0 COMPONENTS SUBJECT TO AN AGING MANAGEMENT REVIEW

In accordance with 10 CFR 54 and NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," Appendix B [Reference C-2], only passive components are in the scope of review. Within the systems that are in the scope of license renewal, the following are typical components subject to aging management review:

ductwork	heat exchangers	pump casings
expansion joints	mechanical closure bolting	steam traps
flexible hoses	orifices and flow elements	tanks/vessels
filters, strainers and housings	piping, tubing, and fittings	valve bodies and bonnets
fuel handling equipment	fuel storage racks	

Many of the components within the scope of license renewal contain gaskets, packing, and seals. However, these items, defined as consumables [Reference C-3], are not subject to aging management review since they do not support the component intended functions as established by design codes and are not long-lived.

3.0 MATERIALS USED IN NON-CLASS 1 COMPONENTS

The following materials are present in non-Class 1 systems within the scope of license renewal:

aluminum	coated canvas	Monel
aluminum alloys	copper	neoprene
carbon steel (plain and galvanized)	copper alloys (admiralty brass, red brass, copper nickel, bronze, aluminum brass, aluminum bronze)	rubber
cast iron (ductile, white, malleable)	Inconel (alloy 600)	stainless steel (wrought and cast)
gray cast iron	low alloy steel	Worthite (nickel based alloy)

Note that some components contain internal and external coatings or linings. For example, Intake Cooling Water piping is cement lined and some valves are rubber lined. These features perform a preventive function, but are not credited for the elimination of aging effects requiring management.

4.0 ENVIRONMENTS

4.1 INTERNAL ENVIRONMENTS

4.1.1 TREATED WATER

Treated water is demineralized water and is the base water for all clean systems. Depending on the system, treated water may involve additional processing. Treated water can be deaerated, and can include corrosion inhibitors, biocides, boric acid, or a combination of these treatments. Within this Application, treated water has been subdivided into groups based on the chemistry of the water.

- Treated water – primary – Normal operating Reactor Coolant System chemistry
- Treated water – secondary – Normal operating secondary chemistry, including Main Steam, Feedwater, and Blowdown Systems
- Treated water – borated – Systems that contain borated water except those included in treated water – primary, including Chemical and Volume Control, Spent Fuel Cooling, and Emergency Core Cooling Systems
- Treated water – All other treated water systems, including Component Cooling Water, Emergency Diesel Generator Cooling, and Chilled Water Systems. These systems utilize corrosion inhibitors and, in some cases, biocides.

Aging effects for materials typically found in treated water environments are summarized in Section 6.1 of this appendix.

4.1.2 RAW WATER

For Turkey Point, raw water constitutes the salt water that comes from a closed cooling water canal system and is used for the main condensers and Intake Cooling Water System, and the city water used for the Fire Protection System. In general, the water has been rough filtered to remove large particles and may contain a biocide additive for control of micro-organisms and macro-organisms. Although city water is purified, it is conservatively classified as raw water for the purposes of aging

management review. Within this Application, raw water has been subdivided into groups based on the chemistry of the water.

- Raw water – cooling canals – Salt water used as the ultimate heat sink
- Raw water – city water – Potable water supplied to the water treatment plant and the Fire Protection System
- Raw water – floor drainage – Fluids collected in building drains. The fluids can be treated water (primary, secondary, borated, or other), raw water (cooling canals or city water), fuel oil, or lubricating oil

Aging effects for materials typically found in raw water environments are summarized in Section 6.2 of this appendix.

4.1.3 AIR/GAS

This includes atmospheric air, dry/filtered instrument air, nitrogen, carbon dioxide, hydrogen, helium, and Halon. Aging effects for materials typically found in air/gas environments are summarized in Section 6.3 of this appendix. Where wetted conditions are determined to exist (e.g., due to condensation), the environment description is amended accordingly, and potential aging effects are addressed.

4.1.4 FUEL OIL

This includes fuel oil for the emergency diesel generators, diesel fire pump, and standby steam generator feedwater pump. Aging effects for materials typically found in fuel oil environments are summarized in Section 6.4 of this appendix.

4.1.5 LUBRICATING OIL

This environment is the lubricating oil for emergency diesel generators, pumps, and other components. Aging effects for materials typically found in lubricating oil environments are summarized in Section 6.5 of this appendix.

4.2 EXTERNAL ENVIRONMENTS

4.2.1 OUTDOOR

The outdoor environment is characterized by moist, salt laden atmospheric air, temperature 30ºF-95ºF, humidity 5%-95%, and exposure to weather, including precipitation and wind. Aging effects for materials typically found in outdoor environments are summarized in Section 7.1 of this appendix.

4.2.2 INDOOR – NOT AIR CONDITIONED

This includes atmospheric air, a temperature of 104ºF maximum , humidity 5%-95%, no exposure to weather. Aging effects for materials typically found in indoor – not air conditioned environments are summarized in Section 7.2 of this appendix.

4.2.3 INDOOR – AIR CONDITIONED

This includes atmospheric air with a specific temperature/humidity range dependent upon the building/room, and involves no exposure to weather. Typically, the temperature is 70ºF, and the humidity is 60%-80%. Aging effects for materials typically found in indoor – air conditioned environments are summarized in Section 7.3 of this appendix.

4.2.4 CONTAINMENT AIR

This includes atmospheric air, a temperature of 120ºF maximum, humidity 5%-95%, radiation – total integrated dose rate 1 rad per hour (excluding equipment located inside the reactor cavity), and no exposure to weather. Aging effects for materials typically found in containment air environments are summarized in Section 7.4 of this appendix.

Note: Safety-related equipment in the Containment has been analyzed to 122ºF continuous and 125ºF for 2 weeks/year.

4.2.5 BORATED WATER LEAKS

This environment includes exposure to leakage from borated water systems. Aging effects for materials exposed to borated water leak environments are summarized in Section 7.5 of this appendix.

4.2.6 BURIED

Above groundwater elevation, this environment involves exposure to soil/fill. Below groundwater elevation, this environment involves exposure to soil/fill and groundwater. Groundwater contains aggressive chemicals that can attack susceptible materials. Although all buried piping and mechanical components are above groundwater elevation, buried components are assumed to be susceptible to corrosion. Aging effects for materials typically found in buried environments are summarized in Section 7.6 of this appendix.

4.2.7 EMBEDDED/ENCASED

This environment is associated with reinforcing or embedded steel or piping in concrete. Aging effects for materials typically found in embedded/encased environments are summarized in Section 7.7 of this appendix.

5.0 POTENTIAL AGING EFFECTS

Potential aging effects were determined based on materials and environments. Aging effects are considered to require management if the effects could cause loss of component intended function during the period of extended operation. The potential aging effects and associated mechanisms evaluated for non-Class 1 components are as follows.

5.1 LOSS OF MATERIAL

Loss of material may be due to general corrosion, pitting corrosion, galvanic corrosion, crevice corrosion, erosion/corrosion, microbiologically influenced corrosion, selective leaching, wear, and aggressive chemical attack.

General corrosion is the result of a chemical or electrochemical reaction between the material and the environment when both oxygen and moisture are present. General corrosion is characterized by uniform attack resulting in material dissolution and, sometimes corrosion product buildup. General corrosion on components exposed to air tends to form a protective oxide film on the component that prevents further significant corrosion. This is typically true for components not exposed to other sources of moisture such as rain, condensation, or frequent leakage. Wrought austenitic stainless steel, copper, copper alloys, cast austenitic stainless steel (CASS), and nickel-based alloys are not susceptible to general corrosion except when subjected to aggressive environments. Carbon and low alloy steels are susceptible to external general corrosion in all areas with the exception of those exposed to a controlled, air-conditioned environment, and those applications where the metal temperature is greater than 212ºF. Additionally, galvanized carbon steel is not considered susceptible to general corrosion except where buried, submerged in fluid, or subject to wetting, such as salt spray, other than humidity.

Pitting corrosion is a form of localized attack that results in depressions in the metal. For treated water systems, oxygen is required for initiation of pitting corrosion with contaminants, such as halogens or sulfates, required for continued metal dissolution. Pitting corrosion occurs when passive films in local areas attack passive materials. Once a pit penetrates the passive film, galvanic conditions occur because the metal in this pit is anodic relative to the passive film. Maintaining adequate flow rate over this exposed surface of a component can inhibit pitting corrosion. However,

stagnant or low flow conditions are assumed to exist in all systems where dead legs of piping, such as vents or drains, exist. Pitting corrosion is more common with passive materials, such as austenitic stainless steels, than with non-passive materials. Most materials of interest are susceptible to pitting corrosion under certain conditions. For treated water environments, stainless and carbon steels are assumed susceptible to pitting in the presence of halogens in excess of 150 ppb or sulfates in excess of 100 ppb when dissolved oxygen is in excess of 100 ppb. However, like general corrosion, moisture must be present, and those metals exposed to a controlled, air-conditioned environment or an operating temperature greater than 212ºF are not susceptible to external pitting corrosion. Because pitting of stainless steel material in an outdoor environment at Turkey Point is dependent on its location within the plant site, these materials were evaluated based upon experience and visual inspections. Typically, stainless steel materials located near the plant discharge canal are more susceptible due to the salt spray environment. Additionally, bronze and brass are considered susceptible to pitting when the zinc content is greater than 15%, and aluminum bronze is considered susceptible to pitting when the aluminum content is greater than 8%.

Loss of material due to galvanic corrosion can occur only when materials with different electrochemical potentials are in contact within an aqueous environment. Generally, the effects of galvanic corrosion are precluded by design (e.g., isolation to prevent electrolytic connection or using similar materials). In galvanic couples involving brass, carbon steel, cast iron, copper, and stainless steel materials, the lower potential (more anodic) carbon steel, cast iron, and low-alloy steel would be preferentially attacked.

Crevice corrosion occurs when a crevice or area of stagnant or low flow exists that allows a corrosive environment to develop in a component . It occurs most frequently in joints and connections, or points of contact between metals and non-metals, such as gasket surfaces, lap joints, and under bolt heads. Crevice corrosion is strongly dependent on the presence of dissolved oxygen. For environments with extremely low oxygen content (less than 0.1 ppm), crevice corrosion is considered insignificant. Carbon steel, cast iron, low alloy steels, stainless steel, copper, and nickel base alloys are all susceptible to crevice corrosion. However, experience at Turkey Point shows that crevice corrosion is not a significant aging mechanism for components subjected to treated water.

Erosion is a mechanical action of fluid and/or particulate matter on a metal surface, without the influence of corrosion. Equipment exposed to moving fluids is vulnerable to erosion. These include piping, valves, vanes, impellers, etc. General erosion occurs under high velocity conditions, turbulence, and impingement. Geometric factors are extremely important. Typical forms of erosion include liquid impingement, flashing, and cavitation. Systems and components are designed to preclude these mechanisms. Additionally, these mechanisms are quite severe and would typically be discovered early in a component's life. In general, erosion mechanisms are not considered aging effects requiring management during the period of extended operation.

Erosion/corrosion occurs when fluid or particulate is also corrosive. Erosion/ corrosion is influenced by 1) fluid flow velocity, 2) geometry, 3) environmental characteristics (temperature and fluid chemistry), and 4) material susceptibility. Carbon and low alloy steels are most susceptible to erosion/corrosion. Higher alloy steels, nickel based alloys, and stainless steels are considered resistant to both erosion and erosion/corrosion. Most of the treated water systems are immune from erosion/corrosion because of their non-corrosive service fluids. One exception to the above involves high-energy piping systems that are susceptible to a form of erosion/corrosion called flow accelerated corrosion (FAC). FAC involves the dissolution of protective oxides on carbon and low alloy steel components, and continual removal of these dissolved oxides by flowing fluid.

Microbiologically influenced corrosion (MIC) is a form of localized, corrosive attack accelerated by the influence of microbiological activity due to the presence of certain organisms. Microbiological organisms can produce corrosive substances, as a byproduct of their biological processes, that disrupt the protective oxide layer on the component materials and lead to a material depression similar to pitting corrosion. Microscopic organisms have been observed in mediums over a wide range of temperatures and pH values. However, for the purpose of aging management review, loss of material due to MIC is not considered significant at temperatures greater than 120ºF or pH greater than 10.

Selective leaching (also known as dealloying) is the dissolution of one element from a solid alloy by corrosion processes. The most common forms of selective leaching are graphitic corrosion with the loss of the iron matrix in gray cast iron under harsh conditions, and dezincification with the removal of zinc from susceptible brass or

bronze components. The addition of small amounts of alloying elements such as tin, phosphorus, arsenic, and antimony is effective in inhibiting this attack in copper-zinc alloys. Therefore, selective leaching of brass and other alloys applies only to "uninhibited" materials.

Mechanical wear is defined as damage to a solid surface by removal of parts of its material via mechanical action of a contacting solid, liquid, or gas. There are three primary types of wear: abrasive, adhesive, and erosive. Abrasive wear (scouring and gouging) is the removal of material from a surface when hard particles slide or roll across the surface under pressure. Scouring and gouging are often due to loose particles entrapped between the surfaces that are in relative motion. Adhesive wear (galling, scoring, seizing, fretting, and scuffing) involves the transference of material from one surface to another during relative motion or sliding due to a process called solid state welding (i.e., particles that are removed from one surface are either temporarily or permanently attached to the other surfaces). Erosive wear is the mechanical wear action of a fluid and/or particulate matter on a solid surface. Erosive wear is also known as erosion, and has been discussed above.

Aggressive chemical attack is corrosion that may be localized or general, and is caused by a corrodent that is particularly active on a specified material. Boric acid is used in pressurized water reactors as a reactivity agent. The concentrations of boric acid in the Reactor Coolant Systems and other borated water systems are lower than the concentration necessary to cause corrosion. However, borated water that leaks out of systems loses substantial volume due to evaporation, resulting in highly concentrated boric acid solutions or deposits of boric acid crystals. These concentrated solutions may be very corrosive for carbon steel. Most carbon steel components located inside the radiation control area were considered susceptible to boric acid corrosion. Other metals, such as copper, copper alloys, nickel, nickel alloys, and aluminum, are resistant to boric acid corrosion.

5.2 CRACKING

Cracking is non-ductile failure of a component due to stress corrosion, fatigue, or embrittlement. The analysis of the potential for cracking due to metal fatigue is a Time-Limited Aging Analysis and is addressed in Section 4.3 of this Application.

Stress corrosion cracking (SCC) requires a combination of a susceptible material, a corrosive environment, and tensile stress. SCC can be categorized as either transgranular stress corrosion cracking (TGSCC) or intergranular stress corrosion cracking (IGSCC), depending on the cracking morphology. For austenitic stainless steels, TGSCC is the normal cracking mode unless the material is in a sensitized condition. As such, SCC of such materials is assumed transgranular in nature unless specified as IGSCC.

The tensile stresses necessary to induce SCC may be either applied (external) or residual (internal), but must be at or near the material's yield point. The corrosive environments that induce SCC are highly material dependent. For austenitic stainless steels and nickel-based alloys in treated water, the relevant conditions required for stress corrosion cracking are the presence of halogens in excess of 150 ppb or sulfates in excess of 100 ppb, and elevated temperature. For Turkey Point treated and raw water environments, a temperature criterion of greater than 140ºF is utilized for susceptibility of austenitic stainless steels and nickel based alloys to SCC. However, stress corrosion cracking has been observed elsewhere in high purity water (i.e., halogens and sulfates less than 150 ppb and 100 ppb, respectively) at temperatures greater than 200^0F with dissolved oxygen levels greater than 100 ppb. IGSCC of stainless steels is generally associated with sensitized material. Sensitization of unstabilized austenitic stainless steel is characterized by a depletion of chromium at the grain boundaries with accompanying precipitation of a network of chromium carbides occurring at elevated temperatures. Generally, exposure periods of one hour to temperatures between 800ºF and 1500ºF are required to fully develop the network of intergranular carbides. However, studies have shown that the thermal effects of welding followed by prolonged exposure to elevated temperatures below the normal sensitization range can also fully develop the intergranular carbide network, thereby rendering the alloys susceptible to intergranular attack (IGA) and IGSCC. Sensitization to IGSCC can occur as low as 480ºF over a long period of service. Because the depletion of chromium at or near grain boundaries is caused by the formation of carbides, the carbon content of the austenitic stainless steel is critical as to the susceptibility of the material to sensitization.

For stainless steels exposed to atmospheric conditions, IGSCC is considered when exposed to high levels of contaminants (e.g., salt water) and only if the material is in a sensitized condition. Additionally, experience at Turkey Point has revealed

susceptibility to TGSCC in the non-stress relieved heat affected zone regions of weld joints in schedule 10 large bore stainless steel pipe exposed to marine atmospheric conditions. However, most austenitic stainless steel and nickel base alloys are resistant to SCC at temperatures less than 140ºF.

For carbon steels, stress corrosion cracking occurs most commonly in the presence of aqueous chlorides. Industry data do not indicate a significant problem of stress corrosion cracking of low strength carbon steels. For these reasons, stress corrosion cracking of carbon steels is not an aging effect requiring management.

Material fatigue resulting from vibration has been observed in the nuclear industry and can result in crack initiation/growth. Vibration induced fatigue is fast acting and typically detected early in a component's life, and corrective actions are initiated to prevent recurrence. Corrective actions typically involve modifications to the plant, such as addition of supplemental restraints to a piping system, replacement of tubing with flexible hose, etc. Based upon these considerations, cracking due to vibration induced fatigue is not considered an aging effect for the period of extended operation.

Embrittlement is an aging mechanism that could cause cracking of rubber, neoprene, or coated canvas materials at Turkey Point. Embrittlement can occur due to age, temperature, or irradiation.

5.3 FOULING

Fouling may be due to accumulation of particulates or macro-organisms (biological). Fouling is an aging effect that could cause loss of heat transfer intended function at Turkey Point. Biological fouling can also lead to environmental conditions conducive to MIC. Fouling evaluated for Turkey Point includes macrofouling (macro-organisms, grass, etc.) and particulate fouling due to precipitation or corrosion products. Fouling is not considered an aging effect for components with an intended function of filtration (e.g., a strainer). In these cases, the component is designed to foul, and this short-term effect is addressed by normal system operating practices.

5.4 LOSS OF MECHANICAL CLOSURE INTEGRITY

The loss of mechanical closure integrity is an aging effect associated with bolted mechanical closures that can result from the loss of pre-load due to cyclic loading, gasket creep, thermal or other effects, cracking, or loss of bolting material.

Loss of pre-load of mechanical closures can occur due to settling of mating surfaces, relaxation after cyclic loading, gasket creep, and loss of gasket compression due to differential thermal expansion. The effects of these mechanisms are the same as that of a degraded gasket; that is, the potential for external leakage of the internal fluid at the mechanical joint. Since the ASME Code does not consider gaskets, packing, seals, and O-rings to perform a pressure-retaining function, these components are typically not considered to support an intended function and not within the scope of license renewal. Thus, the aging mechanisms associated with loss of pre-load, described above, are not considered to require management for non-Class 1 components during the period of extended operation. An exception to this would be a situation where a gasket/seal is utilized to provide a radiological boundary/barrier and thus may support an intended function. Based on the aging management review of the non-Class 1 systems at Turkey Point, there were no cases where gasket/seals are relied on to support component intended functions.

Loss of bolting material, on the other hand, can result in loss of a component's pressure boundary integrity and, thus, must be addressed. Most carbon steel bolting is in a dry environment and coated with a lubricant, thus general corrosion of bolting has not been a major concern in the industry. Additionally, stainless steel fasteners are immune to loss of material due to general corrosion. Corrosion of fasteners has only been a concern where leakage of a joint occurs, specifically, when bolting is exposed to aggressive chemical attack such as that resulting from borated water leaks.

Aggressive chemical attack is corrosion that may be localized or general, and is caused by a corrodent that is particularly active on a specified material. Boric acid is used in pressurized water reactors as a reactivity agent. The concentrations in the Reactor Coolant System and other borated water systems are lower than the concentration necessary to cause corrosion. However, borated water that leaks out of systems loses substantial volume due to evaporation, resulting in highly concentrated boric acid solutions or deposits of boric acid crystals. These

concentrated solutions may be very corrosive for carbon steel. Loss of mechanical closure integrity due to boric acid corrosion was considered as a potential aging effect for components in proximity to borated water systems.

Although there have been a few instances of cracking of bolting in the industry due to SCC, these have been attributed to high yield stress materials and contaminants, such as the use of lubricants containing MoS_2. For quenched and tempered low alloy steels (e.g., SA193 Grade B7) used for closure bolting material, susceptibility to SCC is controlled by yield strength. Additionally, operating experience and existing data indicate that SCC failure should not be a significant issue for the bolting materials of SA193 Grade B7.

5.5 REDUCTION IN FRACTURE TOUGHNESS

Thermal embrittlement is a mechanism by which the mechanical property fracture toughness is affected as a result of exposure to elevated temperature. CASS materials are susceptible to thermal embrittlement dependent upon material composition and the time at temperature. CASS materials subjected to temperatures >482ºF are considered susceptible. Low alloy steels may be subject to embrittlement from exposure to temperatures in the range of 570ºF – 1100ºF. The loss of fracture toughness may not be accompanied by significant changes in other material properties.

Neutron embrittlement is the loss of fracture toughness resulting from the bombardment of neutrons. The loss of fracture toughness may be accompanied by detectable increases in material hardness. The overall effects of neutron embrittlement on steel are to increase yield strength, decrease the ultimate tensile ductility, and increase the ductile to brittle transition temperature. Neutron embrittlement is considered a potential aging mechanism requiring management only inside the reactor cavity.

5.6 DISTORTION/PLASTIC DEFORMATION

Creep is defined as time-dependent strain, or gradual elastic and plastic deformation, of metal that is under constant stress at a value lower than its normal yield strength. Creep is a concern when the component operating temperature approaches or exceeds the crystallization temperature for the metal. Austenitic

stainless steel with temperatures <800ºF, and carbon steel and low alloy steels with temperatures <700ºF are not susceptible to creep. All Turkey Point plant systems operate <700ºF and, thus, are not susceptible to creep.

Stress relaxation is the time-dependent decrease in stress in a solid under constant constraint at constant temperature. The rate of stress relaxation is temperature dependent. As a rule, stress relaxation is not a significant problem at temperatures less than one-half of the melting point. All Turkey Point plant systems operate below the temperature at which stress relaxation occurs and, thus, are not susceptible to stress relaxation.

6.0 AGING EFFECTS REQUIRING MANAGEMENT FOR INTERNAL ENVIRONMENTS

6.1 TREATED WATER

6.1.1 ATTRIBUTES OF TREATED WATER ENVIRONMENTS

Treated water is demineralized water and is the base water for all clean systems. Depending on the system, treated water may involve additional processing. Treated water could be deaerated and can include corrosion inhibitors, biocides, boric acid, or some combination of these treatments. In the determination of aging effects, steam is considered treated water. Although treated water was evaluated as a single environment, for the purposes of clarity in the Application, treated water has been divided into the following groups:

- Treated water – primary – Normal operating Reactor Coolant System chemistry
- Treated water – secondary – Normal operating secondary chemistry, including Main Steam and Feedwater and Blowdown Systems
- Treated water – borated – Systems that contain borated water except those included in treated water – primary, including Chemical and Volume Control, Spent Fuel Cooling, and Emergency Core Cooling Systems
- Treated water – All other treated water systems, including Component Cooling Water, Emergency Diesel Generator Cooling, and Chilled Water Systems

Chemistry requirements for the treated water are stringent since treated water is used for the Reactor Coolant System, and various secondary and auxiliary systems in which high quality is required.

The chemistry for all treated water that is used in cooling water systems includes a corrosion inhibitor, with the exception of the water associated with steam generator makeup, such as condensate storage and demineralized water storage.

6.1.2 MATERIALS USED IN TREATED WATER ENVIRONMENTS

The following materials are exposed to an internal treated water environment:

admiralty brass	cast iron	low alloy steel
aluminum brass	copper nickel	red brass
brass	copper	rubber
bronze	gray cast iron	stainless steel
carbon steel	Inconel	Worthite (nickel based alloy)

6.1.3 AGING EFFECTS IN TREATED WATER ENVIRONMENTS

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with treated water systems.

6.1.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for low alloy steel, cast iron, carbon steel, and gray cast iron in treated water environments.

Loss of material due to pitting corrosion is an aging effect requiring management for admiralty brass, low alloy steel, aluminum brass, brass, carbon steel, cast iron, gray cast iron, Inconel, stainless steel, and Worthite in treated water environments.

Loss of material due to galvanic corrosion is an aging effect requiring management for admiralty brass, brass, carbon steel, cast iron, and copper in treated water environments when coupled with material having higher electrical potential.

Loss of material due to erosion/corrosion is an aging effect requiring management for carbon steel in treated water under certain conditions. Fluid conditions in the Main Steam, Main Feedwater, and Steam Generator Blowdown Systems can lead to erosion/corrosion.

Loss of material due to microbiologically influenced corrosion is an aging effect requiring management for admiralty brass, aluminum brass, brass, carbon steel,

cast iron, copper, copper nickel, gray cast iron, Inconel, red brass, stainless steel, and Worthite (nickel based alloy).

Loss of material due to selective leaching is an aging effect requiring management for admiralty brass, brass, and gray cast iron in treated water environments.

6.1.3.2 CRACKING

Cracking due to stress corrosion, intergranular stress corrosion, embrittlement (rubber products), and high-cycle fatigue of stainless steel materials is an aging effect requiring management in treated water environments. Cracking resulting from fatigue is precluded by design. However, an exception identified is the charging pump fluid blocks that are susceptible to high-cycle fatigue.

A review of NRC Bulletin 79-17, “Pipe Cracks in Stagnant Borated Water Systems at PWR Plants,” identified stress corrosion cracking as a potential aging mechanism in stainless steel piping of safety-related systems containing stagnant borated water. Turkey Point operating experience, in the early 1990s, identified through wall leakage in heat traced boric acid piping and components. Therefore, SCC is considered an aging mechanism requiring management for stainless steel piping and components associated with the boric acid supply to the charging pump suction header. Note that these piping and components are no longer heat traced or insulated.

6.1.3.3 FOULING

Biological and particulate fouling of admiralty brass, aluminum brass, Inconel, red brass, gray cast iron, and stainless steel heat exchanger tubes is an aging effect requiring management in treated water environments.

6.1.4 INDUSTRY EXPERIENCE WITH TREATED WATER ENVIRONMENTS

To validate the aging effects requiring management for components exposed to a treated water internal operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

6.2 RAW WATER

6.2.1 ATTRIBUTES OF RAW WATER ENVIRONMENTS

The majority of the raw water for Turkey Point is from the city water source, the Cooling Water Canal System, or building floor drains. In general, the water has been rough filtered. The city water system is a potable water system and the Cooling Water Canal System is saltwater system.

6.2.2 MATERIALS USED IN RAW WATER ENVIRONMENTS

Materials within the scope of license renewal at Turkey Point that are exposed to raw water include the following:

aluminum brass	carbon steel - galvanized	rubber
aluminum bronze	cast iron	stainless steel
bronze	gray cast iron	Monel
carbon steel	copper nickel	

6.2.3 AGING EFFECTS IN RAW WATER ENVIRONMENTS

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with raw water systems.

6.2.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for cast iron and carbon steel in raw water environments.

Loss of material due to pitting corrosion is an aging effect requiring management for aluminum bronze, carbon steel, cast iron, Monel, and stainless steel in raw water environments.

Loss of material due to galvanic corrosion is an aging effect requiring management for carbon steel and cast iron in raw water environments when coupled with materials having higher electrical potential.

Loss of material due to crevice corrosion is an aging effect requiring management for aluminum bronze, bronze, carbon steel, cast iron, copper nickel, Monel, and stainless steel in raw water environments.

Loss of material due to microbiologically influenced corrosion is an aging effect requiring management for aluminum bronze, bronze, carbon steel, cast iron, copper nickel, Monel, and stainless steel in raw water environments.

Loss of material due to selective leaching is an aging effect requiring management for aluminum bronze and gray cast iron in raw water environments.

6.2.3.2 CRACKING

Cracking due to embrittlement is an aging effect requiring management for rubber in raw water environments.

6.2.3.3 FOULING

Biological and particulate fouling of copper alloy heat exchanger tubes is an aging effect requiring management in raw water environments. Additionally, particulate fouling (clogging) of stainless steel floor drains is an aging effect requiring management.

6.2.4 INDUSTRY EXPERIENCE WITH RAW WATER ENVIRONMENTS

To validate the aging effects requiring management for components exposed to a raw water internal operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

6.3 AIR/GAS

6.3.1 ATTRIBUTES OF AN AIR/GAS ENVIRONMENT

The environments included in this category include atmospheric air, dry/filtered instrument air, nitrogen, carbon dioxide, hydrogen, and Halon. In some cases, a wetted environment may exist due to condensation (e.g., at an air conditioning air handler housing). For these cases, the environment description is amended and aging effects addressed accordingly. The various gaseous environments addressed in this discussion are described below.

6.3.1.1 AIR

Air is composed of mostly nitrogen and oxygen with smaller fractions of various other constituents. The internal surfaces of a majority of components are, at some time, exposed to air. Where air is the intended internal environment, it is supplied in either its natural state or in a dry condition.

6.3.1.2 NITROGEN

Nitrogen is an inert gas used in many nuclear power plant applications to place components in a dry lay-up condition or to provide a cover gas to prevent exposure to oxygen.

6.3.1.3 CARBON DIOXIDE

Carbon dioxide is a colorless, odorless incombustible gas. The carbon dioxide system contains dry carbon dioxide in gaseous form. Without the presence of moisture, the gaseous carbon dioxide is not a significant contributor to corrosion or other aging effects.

6.3.1.4 HYDROGEN

Hydrogen is a colorless, odorless gas.

6.3.1.5 HALON

Halon is a halogenated extinguishing agent used in the Fire Protection System for its ability to chemically react with fire and smother flames. Halon is a non-corrosive gas.

6.3.2 MATERIALS USED IN AN AIR/GAS ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to air/gas contain the following materials:

aluminum	carbon steel	copper
aluminum alloys	carbon steel - galvanized	rubber
brass	cast iron	stainless steel
bronze	coated canvas	

6.3.3 AGING EFFECTS IN AN AIR/GAS ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with air/gas systems.

6.3.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for carbon steel and cast iron in atmospheric air/gas environments. Loss of material due to general corrosion is an aging effect requiring management for galvanized carbon steel in wetted air/gas environments, such as upstream of the air dryers in the Instrument Air System.

Loss of material due to pitting corrosion is an aging effect requiring management for carbon steel and stainless steel in atmospheric air/gas environments. Loss of material due to general corrosion is an aging effect requiring management for galvanized carbon steel in wetted air/gas environments, such as upstream of the air dryers in the Instrument Air System.

Loss of material due to galvanic corrosion is an aging effect requiring management for carbon steel, galvanized carbon steel, and copper alloys in wetted air/gas environments, such as upstream of the air dryers in the Instrument Air System or some heat exchangers in the ventilation systems.

Loss of material due to crevice corrosion is an aging effect requiring management for carbon steel in wetted air/gas environments.

6.3.3.2 CRACKING

Cracking due to fatigue is an aging effect requiring management for stainless steel in air/gas environments only for the emergency diesel generator exhaust expansion joints.

Cracking due to embrittlement is an aging effect requiring management for coated canvas and rubber in air/gas environments.

6.3.4 INDUSTRY EXPERIENCE WITH AIR/GAS ENVIRONMENT

To validate the aging effects requiring management for components exposed to an air/gas internal operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

6.4 FUEL OIL

6.4.1 ATTRIBUTES OF A FUEL OIL ENVIRONMENT

Fuel oil within the scope of license renewal is No. 2 diesel oil. Diesel fuel oil is delivered to Turkey Point in tanker trucks and is stored in large tanks to provide an on-site supply of diesel fuel for a specified period of emergency diesel generator operating time. The fuel oil is supplied to the emergency diesel engines through pumps, valves, and piping. Strainers, filters, and other equipment assure that the diesel fuel supplied to the engines is clean and free of contaminants. Fuel oil is also supplied to the diesel driven fire pump and the standby steam generator feedwater pump.

6.4.2 MATERIALS USED IN A FUEL OIL ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to fuel oil contain the following materials:

carbon steel	copper
cast iron	stainless steel

6.4.3 AGING EFFECTS IN A FUEL OIL ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with fuel oil systems.

6.4.3.1 LOSS OF MATERIAL

Loss of material due to microbiologically influenced corrosion is an aging effect requiring management for carbon steel, cast iron, copper, and stainless steel in fuel oil environments.

6.4.4 INDUSTRY EXPERIENCE WITH A FUEL OIL ENVIRONMENT

To validate the aging effects requiring management for components exposed to a fuel oil internal operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

6.5 LUBRICATING OIL

6.5.1 ATTRIBUTES OF A LUBRICATING OIL ENVIRONMENT

Lubricating oils within the scope of license renewal are low to medium viscosity hydrocarbons used for bearing, gear, and engine lubrication:

6.5.2 MATERIALS USED IN A LUBRICATING OIL ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to lubricating oil contain the following materials:

admiralty brass	copper	stainless steel
carbon steel	gray cast iron	
cast iron	red brass	

6.5.3 AGING EFFECTS IN A LUBRICATING OIL ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. No aging effects requiring management for non-Class 1 components in a lubricating oil environment have been identified.

6.5.4 INDUSTRY EXPERIENCE WITH A LUBRICATING OIL ENVIRONMENT

To validate the aging effects requiring management for components exposed to a lubricating oil internal operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

7.0 AGING EFFECTS REQUIRING MANAGEMENT FOR EXTERNAL ENVIRONMENTS

7.1 OUTDOOR

7.1.1 ATTRIBUTES OF AN OUTDOOR ENVIRONMENT

The outdoor environment consists of moist, salt laden atmospheric air, temperature 30ºF-95ºF, humidity 5%-95%, and exposure to weather, including precipitation and wind.

Note that a component is considered susceptible to a wetted environment when it is submerged, has the potential to pool water, or is subject to external condensation.

7.1.2 MATERIALS USED IN AN OUTDOOR ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to an outdoor environment contain the following materials:

aluminum	carbon steel	low alloy steel
aluminum bronze	carbon steel - galvanized	Monel
brass	cast iron	rubber
bronze	copper nickel	stainless steel

7.1.3 AGING EFFECTS IN AN OUTDOOR ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with materials in an outdoor environment.

7.1.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for low alloy steel, carbon steel, and cast iron in outdoor environments.

Loss of material due to pitting corrosion is an aging effect requiring management for low alloy steel, carbon steel, cast iron, and, in some locations, stainless steel in outdoor environments.

Loss of material due to galvanic corrosion is an aging effect requiring management for carbon steel and cast iron in outdoor wetted environments.

Loss of material due to crevice corrosion is an aging effect requiring management for carbon steel and cast iron in outdoor wetted environments.

Loss of material due to microbiologically influenced corrosion is an aging effect requiring management for carbon steel and cast iron in outdoor wetted environments.

7.1.3.2 CRACKING

Cracking due to stress corrosion is an aging effect requiring management for stainless steel in outdoor environments for large bore thin wall pipe, as discussed in Section 5.2. Cracking of rubber due to embrittlement is an aging effect requiring management.

7.1.3.3 FOULING

Fouling due to particulates is an aging effect requiring management for aluminum heat exchanger fins of the Turbine Building Ventilation System.

7.1.4 INDUSTRY EXPERIENCE WITH AN OUTDOOR ENVIRONMENT

To validate the aging effects requiring management for components exposed to an outdoor external operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

7.2 INDOOR – NOT AIR CONDITIONED

7.2.1 ATTRIBUTES OF AN INDOOR – NOT AIR CONDITIONED ENVIRONMENT

The indoor – not air conditioned environment consists of atmospheric air, a temperature of 104ºF maximum, humidity 5%-95%, and no exposure to weather.

7.2.2 MATERIALS USED IN AN INDOOR – NOT AIR CONDITIONED ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to an indoor – not air conditioned environment contain the following materials:

aluminum	carbon steel - galvanized	rubber
aluminum alloys	cast iron	stainless steel
brass	copper	Worthite (nickel based alloy)
carbon steel	gray cast iron	

7.2.3 AGING EFFECTS IN AN INDOOR – NOT AIR CONDITIONED ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with materials in an indoor-not air conditioned environment.

7.2.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for carbon steel and cast iron in indoor – not air conditioned environments.

Loss of material due to pitting corrosion is an aging effect requiring management for carbon steel and cast iron in indoor – not air conditioned environments.

7.2.3.2 CRACKING

Cracking due to stress corrosion is an aging effect requiring management for stainless steel piping and components associated with the boric acid supply to the charging pump supply header that was previously heat traced.

Cracking due to fatigue is an aging effect requiring management for stainless steel in isolated cases, such as the emergency diesel generator expansion joints.

Cracking due to embrittlement is an aging effect requiring management for rubber in indoor – not air conditioned environments.

7.2.4 INDUSTRY EXPERIENCE WITH AN INDOOR – NOT AIR CONDITIONED ENVIRONMENT

To validate the aging effects requiring management for components exposed to an indoor – not air conditioned external operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

7.3 INDOOR – AIR CONDITIONED

7.3.1 ATTRIBUTES OF AN INDOOR – AIR CONDITIONED ENVIRONMENT

The indoor – air conditioned environment consists of atmospheric air with a temperature/humidity range dependent on the building/room and no exposure to weather. Typical temperature is 70ºF and humidity is 60%-80%.

Note that a component is considered susceptible to a wetted environment when it is submerged, has the potential to pool water, or is subject to external condensation.

7.3.2 MATERIALS USED IN AN INDOOR – AIR CONDITIONED ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to an indoor – air-conditioned environment contain the following materials:

aluminum	carbon steel - galvanized	copper
carbon steel	coated canvas	stainless steel

7.3.3 AGING EFFECTS IN AN INDOOR – AIR-CONDITIONED ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with materials in a wetted indoor-air conditioned environment.

7.3.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for carbon steel and copper when wetted in indoor – air conditioned environments.

Loss of material due to pitting corrosion is an aging effect requiring management for carbon steel, copper, and stainless steel when wetted in indoor – air conditioned environments.

Loss of material due to galvanic corrosion is an aging effect requiring management for aluminum, carbon steel, and copper when wetted in indoor – air conditioned environments.

Loss of material due to microbiologically influenced corrosion is an aging effect requiring management for carbon steel when wetted in indoor – air conditioned environments.

7.3.3.2 CRACKING

Cracking due to embrittlement is an aging effect requiring management for coated canvas and rubber in indoor – air conditioned environments.

7.3.4 INDUSTRY EXPERIENCE WITH AN INDOOR – AIR CONDITIONED ENVIRONMENT

To validate the aging effects requiring management for components exposed to an indoor – air conditioned external operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

7.4 CONTAINMENT AIR

7.4.1 ATTRIBUTES OF A CONTAINMENT AIR ENVIRONMENT

The containment air consists of atmospheric air, a temperature of 120ºF maximum, humidity 5%-95%, radiation total integrated dose rate of 1 rad/hr (excluding equipment located inside the reactor cavity), and no exposure to weather.

Safety-related equipment in the Containment has been analyzed to 122ºF continuous and 125ºF for two weeks/year.

Note that a component is considered susceptible to a wetted environment when it is submerged, has the potential to pool water, or is subject to external condensation.

7.4.2 MATERIALS USED IN A CONTAINMENT AIR ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to a containment air environment contain the following materials:

admiralty brass	bronze	coated canvas
aluminum	carbon steel	neoprene
brass	carbon steel - galvanized	stainless steel

7.4.3 AGING EFFECTS IN A CONTAINMENT AIR ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with materials in a containment air environment.

7.4.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for carbon steel in containment environments and admiralty brass when wetted in containment environments.

Loss of material due to pitting corrosion is an aging effect requiring management for carbon steel in containment environments and admiralty brass and stainless steel when wetted in containment environments.

Loss of material due to crevice corrosion is an aging effect requiring management for carbon steel when wetted in containment environments.

Loss of material due to microbiologically influenced corrosion is an aging effect requiring management for carbon steel when wetted in containment environments.

7.4.3.2 CRACKING

Cracking due to embrittlement is an aging effect requiring management for coated canvas and neoprene in containment environments.

7.4.3.3 FOULING

Fouling due to particulates is an aging effect requiring management for aluminum heat exchanger fins in containment environments.

7.4.4 INDUSTRY EXPERIENCE WITH A CONTAINMENT AIR ENVIRONMENT

To validate the aging effects requiring management for components exposed to a containment air external operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

7.5 BORATED WATER LEAKS

7.5.1 ATTRIBUTES OF A BORATED WATER LEAKS ENVIRONMENT

The concentrations of boric acid in the Reactor Coolant System and other borated water systems are lower than the concentration necessary to cause corrosion. However, borated water that leaks out of systems loses substantial volume due to evaporation, resulting in highly concentrated boric acid solutions or deposits of boric acid crystals. These concentrated solutions may be very corrosive for carbon steel.

7.5.2 MATERIALS USED IN A BORATED WATER LEAKS ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to a borated water leaks environment contain the following materials:

carbon steel	cast iron
carbon steel - galvanized	low alloy steel

7.5.3 AGING EFFECTS IN A BORATED WATER LEAKS ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with materials exposed to a borated water environment.

7.5.3.1 LOSS OF MATERIAL

Loss of material due to aggressive chemical attack is an aging effect requiring management for carbon steel, low alloy steel, cast iron, and galvanized carbon steel susceptible to potential borated water leaks.

7.5.3.2 LOSS OF MECHANICAL CLOSURE INTEGRITY

Loss of mechanical closure integrity due to aggressive chemical attack is an aging effect requiring management for mechanical closure carbon and low alloy steel bolting susceptible to potential borated water leaks. Components located in proximity to borated water systems are considered susceptible.

7.5.4 INDUSTRY EXPERIENCE WITH A BORATED WATER LEAKS ENVIRONMENT

To validate the aging effects requiring management for components exposed to a borated water leaks external operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

7.6 BURIED

7.6.1 ATTRIBUTES OF A BURIED ENVIRONMENT

The buried environment applies to components buried in the soil and exposed to soil and groundwater. The soil and groundwater are untreated and could be corrosive materials. The factors affecting corrosiveness are moisture, pH, permeability of soil for water and air, oxygen, salts, stray currents, and biological organisms. Buried components are assumed susceptible to corrosion.

7.6.2 MATERIALS USED IN A BURIED ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to a buried environment contain the following materials:

carbon steel

gray cast iron

cast iron

7.6.3 AGING EFFECTS IN A BURIED ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. Below is a summary of aging effects requiring management associated with materials in a buried environment.

7.6.3.1 LOSS OF MATERIAL

Loss of material due to general corrosion is an aging effect requiring management for carbon steel and cast iron in buried environments.

Loss of material due to pitting corrosion is an aging effect requiring management for carbon steel and cast iron in buried environments.

Loss of material due to galvanic corrosion is an aging effect requiring management for carbon steel and cast iron in buried environments.

Loss of material due to crevice corrosion is an aging effect requiring management for carbon steel and cast iron in buried environments.

Loss of material due to selective leaching is an aging effect requiring management for gray cast iron in buried environments.

7.6.4 INDUSTRY EXPERIENCE WITH A BURIED ENVIRONMENT

To validate the aging effects requiring management for components exposed to a buried external operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

7.7 EMBEDDED/ENCASED

7.7.1 ATTRIBUTES OF AN EMBEDDED/ENCASED ENVIRONMENT

An embedded/encased environment involves steel piping embedded or encased in concrete.

7.7.2 MATERIALS USED IN AN EMBEDDED/ENCASED ENVIRONMENT

The non-Class 1 components within the scope of license renewal exposed to an embedded/encased environment contain the following material:

steel

7.7.3 AGING EFFECTS IN AN EMBEDDED/ENCASED ENVIRONMENT

Section 5.0 provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation. No aging effects requiring management for non-Class 1 components in an embedded/encased environment have been identified.

7.7.4 INDUSTRY EXPERIENCE WITH AN EMBEDDED/ENCASED ENVIRONMENT

To validate the aging effects requiring management for components exposed to an embedded/encased external operating environment, industry experience was reviewed. The survey of industry experience included a review of NRC generic communications and NUREG documents. No aging effects were identified in these documents beyond those discussed in this section.

8.0 REFERENCES

C-1 C. I. Grimes (NRC) letter to D. J. Walters (NEI), “License Renewal Issue No. 98-0013, Degradation Induced Human Activities," June 5, 1998.

C-2 NEI 95-10, Revision 1, “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” Nuclear Energy Institute, January 2000.

C-3 C. I. Grimes (NRC) letter to D. J. Walters (NEI), “License Renewal Issue No. 98-12, “Consumables," March 10, 2000.

APPENDIX D

TECHNICAL SPECIFICATION CHANGES

The Code of Federal Regulations, Title 10 at 54.22, requires applicants to include any Technical Specification changes, or additions, necessary to manage the effects of aging during the period of extended operation as part of the renewal application. Based on a review of the information provided in the Turkey Point License Renewal Application and Technical Specifications, no license amendment requests for Turkey Point Units 3 and 4 Technical Specifications are being submitted with this Application.