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# 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.

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 regulatory guidance and industry standards as listed in Appendix C of the FPL Topical Quality Assurance Report. Corrective Actions, Confirmatory Actions, and Administrative Controls apply to all aging management programs credited for license renewal, and are performed, or in the case of new programs, will be performed, in accordance with the FPL Quality Assurance Program.

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.

For the extended power uprate (EPU), aging management programs and time-limited aging analyses associated with systems, structure and components within the scope of the EPU were evaluated to determine the impact of the EPU on previous results. The evaluation determined that the aging management programs are not impacted by the implementation of the EPU. Time-limited aging analyses were shown to remain valid with the implementation of the EPU for the period of extended operation. C26

## 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 was performed on the cast iron bonnets of the ‘C’ auxiliary feedwater pump lube oil cooler to assess the extent of loss of material due to corrosion. The results of this inspection were evaluated to determine the need for additional inspections/programmatic corrective actions. The ‘A’ auxiliary feedwater pump lube oil cooler was replaced in 1999. The evaluation states that this lube oil cooler is acceptable through the period of extended operation. The ‘B’ auxiliary feedwater pump lube oil cooler is scheduled for replacement prior to the end of the initial operating license for Unit 3. The ‘C’ auxiliary feedwater pump lube oil cooler was replaced in 2012 following the inspection. An inspection of one of the turbine governor controller oil coolers was performed. The bonnets on this oil cooler were determined not to be cast iron and not susceptible to graphitic corrosion. These inspections and evaluations were implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. C26

### 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. These inspections were implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. C26

### 16.1.3 EMERGENCY CONTAINMENT COOLERS INSPECTION

A one-time volumetric examination of a sample of emergency containment coolers (ECC) tubes was performed to determine the extent of loss of material due to erosion in the ECC tubes. This inspection and evaluation was implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. Based on this inspection and evaluation, all six ECCs are expected to maintain a wall thickness above the design code minimum wall thickness of 0.010 inches per ASME Section III, Class C, during the extended life of the coolers. C26

### 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 -- was performed. No material loss greater than the design corrosion allowance was identified. The results of these inspections were evaluated to determine the need for additional inspections/programmatic corrective actions. These inspections were implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. C26

### 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. Baseline examinations in select systems were performed and evaluated to establish that the corrosion mechanism was not active. Based on the results of these inspections, no need for followup examinations or programmatic corrective actions was identified. The program was implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. C26

## 16.1.6 REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

Florida Power & Light Company (FPL) previously developed a Reactor Vessel Internals (RVI) Aging Management Program (AMP) for Turkey Point Units 3 and 4 that was included in its License Renewal Application (LRA) and subsequently approved by the Nuclear Regulatory Commission (NRC) in NUREG 1759. Included in the RVI AMP was a commitment to participate in ongoing, joint industry efforts aimed at further understanding the aging effects of the RVI and to revise the Turkey Point Units 3 and 4 RVI AMP as needed. These joint industry efforts culminated in the issuance of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) topical report MRP-227-A. C28

On December 22, 2011, FPL submitted letter L-2011-531 to notify the NRC of its intent to revise the original RVI AMP to align with MRP-227-A.

By letter dated December 14, 2012 as supplemented by letters dated October 30, 2013, January 29, and December 29, 2014, and July 15, 2015, FPL submitted its RVI commitment implementation report and inspection plan for Turkey Pont Units 3 and 4 to the NRC. FPL's submittal addressed the commitment described in NUREG-1759. FPL's RVI inspection plan credited the implementation of the NRC Staff approved EPRI topical report MRP-227-A. The NRC reviewed the RVI AMP and the Staff's assessment is documented in a report dated December 18, 2015 (ML15336A046) C28

The Staff concluded that the Turkey Point RVI AMP is acceptable because it is consistent with the inspection and evaluation guidelines of MRP-227-A and because FPL addressed the applicant/licensee action items specified in MRP-227-A that are applicable to Turkey Point. The Staff considers the regulatory commitment in Section 3.8.6 of NUREG-1759 fulfilled. The Staff notes that Section 7.0, "Implementation Requirements," of MRP-227-A states that the licensee shall notify the NRC of any deviations from the "Needed" implementation requirements of MRP-227-A.

The Turkey Point Reactor Vessel Internals (RVI) Aging Management Program (AMP), based upon MRP-227-A, manages the effects of aging on the RVI during the extended periods of operation. The RVI AMP is applicable to passive RVI structural components contained within the upper and lower internals assemblies. The RVI AMP specifically excludes welded attachments to the reactor vessel and consumable items such as fuel assemblies and RCCAs.

Specific aging effects/degradation mechanisms managed by the RVI AMP include: 1) cracking due to SCC, IASCC or fatigue; 2) reduction in fracture toughness due to irradiation or thermal embrittlement; 3) loss of material due to wear; 4) dimensional change due to void swelling; 5) loss of mechanical closure integrity (or preload) due to irradiation and thermal enhanced stress relaxation or creep.

Methods employed to manage these aging effects for the above components include periodic visual, volumetric and surface inspections of primary or lead components; similar inspections of expansion components when degradation is detected in primary components; strict control of corrosive chemical species in the RCS as a preventative measure for corrosion related degradation mechanisms; and periodic replacement of components when required. The RVI relies on current ASME Section XI inspection requirements for the management of aging in certain components but in no way replaces or relieves current ASME Section XI inspection requirements, or any other requirements related to inservice inspection. The RVI AMP is a living program that will be revised as necessary in response to ongoing joint industry efforts aimed at further understanding the aging effects of the RVI. C28

### 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. If an acceptable volumetric technique is not available to perform a volumetric inspection for socket welds, a destructive examination may be performed. For each socket weld that is destructively examined credit may be taken as being equivalent to volumetrically examining two socket welds. This one-time inspection will address Class 1 piping less than 4 inches nominal pipe size (less than NPS 4) and greater than or equal to NPS 1. 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. A report describing the details of the inspection plan was submitted to the NRC prior to the implementation of this inspection.

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### 16.1.8 CONTAINMENT CABLE INSPECTION PROGRAM

The Containment Cable Inspection Program manages potential aging of EQ low voltage I&C cables, non-EQ cable, connections, and penetrations. This aging management program consists of periodic visual inspection of accessible EQ low voltage I&C cables, non-EQ cables, connections and penetrations within the scope of license renewal located in the containment structures that may be installed in adverse localized environments. The inspections were implemented before the end of the initial operating license terms for Turkey Point Units 3 and 4.

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### 16.1.9 PRESSURIZER SURGE LINE WELDS - AGING MANAGEMENT PROGRAM

The Pressurizer surge line aging management program incorporates: 1) time limited aging analyses (TLAA) that considers fatigue design, and 2) an aging management inspection program that has been approved by the NRC. Details of each of these aspects of the aging management program for the Pressurizer surge line welds are further discussed.

The fatigue design of the pressurizer surge line is based on: 1) the original transient design limits and 2) reactor water environmental effects using the most recent data from laboratory simulation of the reactor coolant environment. To address the initial 40 year operating period, Idaho National Engineering Laboratories evaluated fatigue sensitive component locations in plants designed by all four U.S. nuclear steam supply system (NSSS) vendors, as reported in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995. This evaluation included calculation of fatigue usage factors for critical fatigue sensitive component locations of early vintage Westinghouse pressurized water reactors (PWR), 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 were then utilized to scale up the Turkey Point plant specific usage factors for the same locations to account for environmental effects.

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Thus, the results are directly relevant to the design code used for Turkey Point Unit 3 and Unit 4, and the transient cycles considered in the evaluation match or bound the Turkey Point design. Other generic industry studies performed by EPRI and NEI were also considered as well as environmental data that have been collected and published subsequent to the generic industry study. Fatigue monitoring by FPL will ensure that these limits are not exceeded. In this manner the original transient design limits for fatigue are confirmed to remain valid for the current 40 year operating period and also for the extended 20 year operating life of Turkey Point Unit 3 and Unit 4.

FPL has previously inspected all surge line welds on both units during the fourth in-service inspection (ISI) interval, prior to entering the extended period of operation. The results of these inspections were utilized to assess fatigue of the surge lines. In addition to these inspections, environmentally assisted fatigue of the surge lines welds is addressed using the following approach:

1. Florida Power & Light elected to manage the effects of environmentally assisted fatigue of the pressurizer surge line welds by an aging management inspection program approved by the NRC.

2. The aging management of the surge line will be accomplished by a combination of flaw tolerance analysis and in-service inspection. The aging effect managed with these inspections is cracking due to environmentally assisted fatigue. The technical justification and inspection frequency are supported by the flaw tolerance analysis based on the methodology noted in ASME Section XI, Nonmandatory Appendix L, "Operating Plant Fatigue Assessment". Based on postulated flaw tolerance analysis, and using the guidelines of ASME Code, Section XI, Appendix L, Table L-3420-1, the periodic inspection schedule is determined to be ten years. C26

3. All pressurizer surge line welds listed in scope of the aging management program will be examined in accordance with ASME Section XI, IWB for Class 1 welds. Inservice examinations for the surge line welds will be both surface and volumetric examinations. In each 10-year ISI interval during the period of extended operation, all surge line welds will be inspected in accordance with the Turkey Point ISI Program under Augmented and other programs.

## 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. This program was enhanced to require VT-1 examinations of the core support lugs during the period of extended operation. C26

#### 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. The program also includes inspection of Containment reinforced concrete above groundwater for evidence of concrete degradation.

### 16.2.2 DELETED

### 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."

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### 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. Section 9.6.1 provides a detailed discussion of the Fire Protection Program. C28

The scope of the Fire Protection Program was enhanced to include inspection of additional components prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. Additionally, Turkey Point has either eliminated or replaced wet pipe sprinkler heads following the guidance of NFPA 25.

### 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 was 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. C26

### 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 was 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. C26

### 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 were implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. C26

### 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 primary water stress corrosion cracking (PWSCC). The program included a one-time volumetric examination of selected Unit 4 reactor vessel head penetrations to detect crack initiation. The Unit 3 and Unit 4 reactor vessel heads were replaced due to the susceptibility of the Alloy 600 penetrations to PWSCC. The replacement reactor heads utilize Alloy 690 materials on all head penetrations due to the Alloy 690 resistance to PWSCC. 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. Turkey Point will continue to participate in industry programs to ensure that PWSCC of the reactor vessel head penetrations is managed for the period of extended life.

### 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 was 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. C26

#### 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.

The Systems and Structures Monitoring Program (SSMP) has been enhanced via issuance of procedures 0-ADM-561, "Systems and Structures Monitoring Program", and 0-ADM-564, "Systems/Programs Monitoring". These procedures address inspection requirements to manage certain aging effects in accordance with 10 CFR 54, modifying the scope of specific inspections, and improving documentation requirements.

### 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 was performed prior to the end of the initial operating license term for Turkey Point Unit 3 in October 2004.

This aging management program was converted to a periodic inspection program, which provides for periodic eddy current inspection to ensure that thimble tube thinning will be adequately managed for the period of extended operation. C30

### 16.2.17 METAMIC® INSERT SURVEILLANCE PROGRAM

PTN is the first application of the use of Metamic® inserts in spent fuel pool racks. Consequently, the surveillance program is rigorous with respect to monitoring the physical properties of these inserts while in use as a neutron absorber. The surveillance program will perform periodic examinations to assess the physical properties of selected inserts, see below. Periodically, two (2) Metamic® coupons installed in the spent fuel pit racks will be removed and tested to determine their neutron attenuation capabilities.

Certain acceptance criteria have been established for these inspections, see below. If the inserts inspected fail to satisfy the acceptance criteria, the inspections will be expanded on the "Lead" unit. Once the condition is confirmed on the "Lead" unit it would be assumed to be present on the "non-Lead" unit also. Confirmation inspections will be performed and corrective actions taken as required under the FPL corrective action program.

The frequency of the Metamic® surveillance program ensures timely detection of potential problems which might impact the intended function of the inserts and is consistent with regulatory requirements.

<u>Metamic® Insert Surveillance Testing Requirements and Associated Acceptance Criteria</u>

The following are the surveillance testing requirements and associated acceptance criteria for Metamic® inserts. Surveillance testing is only applicable to those inserts and coupons installed in the "lead unit" unless a condition is identified that warrants testing of the inserts on the "non-lead unit."

1. <u>Metamic® Insert Selection Criteria for Surveillance Testing</u>

   Metanmic® inserts inspected as a part of surveillance testing program will be selected by considering the following criteria and generally selecting the most challenging conditions:

   a. Results of site receipt and pre-installation inspections (i.e., inserts shall be selected if they have more pre-existing conditions);

   b. Experience gained during installation (i.e., select inserts that experienced higher insertion or removal forces);

   c. Post-Installation spatial distribution of inserts in Region II racks and within the individual storage rack modules (i.e., selecting inserts surrounded by fuel assemblies and located in areas not adjacent to the pit walls);

d. Spatial variations in cooling water flow within the spent fuel pit, specifically considering the effects of the Spent Fuel Pit Cooling system suction and discharge piping;

e. Storage arrangements and the characteristics of fuel assemblies proximate to each insert, especially heat generation rates;

f. Noteworthy or unique aspects of Turkey Point fuel pit-related operating experience during the in-service interval, such as atypical water chemistry or impact by a foreign object;

g. Relevant operation experience from other plants.

2. <u>Surveillance Testing Requirements</u>

a. Camera aided visual inspection will be performed on five (5) inserts at 4, 8, 12, 20 and 30 years to observe for visual indications such as through-wall corrosion/damage, bubbling, blistering, corrosion pitting, cracking or flaking. In addition, the following examinations will be performed depending on the type of insert design, either formed or welded: For the Formed Design - inspect insert base material, edges and material near both the interior and exterior bend radii for cracking, deformation, corrosion pitting and other surface based defects also inspect any Metamic® to Metamic® or Metamic® to Aluminum welds for indications of cracking, deformation, corrosion pitting and other surface based defects; For the Welded Design – inspect insert base material, edges, areas near welds (heat affected zone) on external surfaces (those areas that can be observed without disassembly or destructive examination) for cracking or voids, excessive discoloration, corrosion pitting and other surface based flaws. For both types of design, mechanical hardware attachments shall be examined for indications of anomalies. Anomalies observed during inspections will be recorded using photographs, video tape or some similar media.

b. Dimensional measurements of length, width, and thickness will be performed on two (2) inserts at 4, 12, 20 and 30 years.

c. Weight measurements will be performed on two (2) inserts at 4, 12, 20 and 30 years.

d. Neutron attenuation testing will be performed on two (2) coupons at 4, 12, 20 and 30 years.

3. <u>Surveillance Testing Acceptance Criteria</u>

a. Camera Aided Visual Examinations:

1) Evidence of visual indications.

b. Dimensional. Measurements (compared to the Metamic® Inserts initial dimensions)
   1) Length: any change of ± 1.0 inch
   2) Width: any change off 0.5 inch
   3) Thickness: any increase in thickness > 0.010 inch or a decrease in thickness > 0.004 inch.
c. Weight Measurements
   1) Any change of ± 10%
d. Neutron Attenuation Testing
   1) A significant decrease in areal density. Significant' is defined as any unexpected decrease in areal density from the as-fabricated condition outside the statistical inaccuracies of the testing methodology.

4. <u>Surveillance Test Results Corrective Actions, Documentation and Reporting Requirements</u>

Should any inserts exceed the acceptance criteria, the discrepancy will be documented in the FPL Corrective Action Program. The corrective actions will include doubling the sample size to a total of 10 inserts in order to determine the extent of condition. If more than one additional insert is found to exceed the acceptance criteria, then an inspection of all in-service Metamic® inserts at that unit will be performed. Any inserts not meeting the acceptance criteria will be removed from service.

FPL will document the results of each in-service surveillance campaign performed on Metamic® inserts. Results of any diagnostic measurement campaigns performed to assess anomalous conditions found after installation will also be documented.

A summary level report of the results of each surveillance campaign will be provided to the NRC within six months after completing the campaign.

## 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 (RTPTS) whenever a significant change occurs in projected values of RTPTS, or upon request for a change in the expiration date for the operation of the facility.

The calculated RTPTS 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).

The analysis associated with pressurized thermal shock was evaluated to consider EPU conditions. The results of this evaluation concluded the weld and base metal values remain below NRC screening criteria values using the EPU fluence projections through 48 EFPYs. As a result, the analysis associated with pressurized thermal shock remains valid with the implementation of the EPU through the period of extended operation. C26

### 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).

The analysis associated with upper shelf energy was evaluated to consider EPU conditions. the results of this evaluation showed that two circumferential welds were below the NRC screening criteria and required an equivalent margin analysis to demonstrate the requirements of 10 CFR 50, Appendix G, for low upper shelf energy are met. The equivalent margin analysis concluded that the requirements of 10CFR 50, Appendix G, for low upper shelf are satisfied at 48 EFPYs of plant operation with EPU. As a result, the analysis associated with upper shelf energy remains valid with the implementation of the EPU through the period of extended operation. C26

### 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 intermediate shell to lower shell circumferential girth weld. As discussed in the NRC Safety Evaluation for Technical Specification Amendments 208/202 Turkey Point 32 EFPY Pressure-Temperature Curves, future submittals will ensure the consideration of the chemistry factor ratio adjustment in accordance with Regulatory Guide 1.99, Revision 2, Position 2.1, and that the reactor pressure vessel circumferential weld (heat number 72442) is tracked and considered.

A license amendment to incorporate the pressure-temperature limit curves projected to 48 effective full power years was submitted to the NRC in letter L-2010-113, License Amendment Request for Extended Power Uprate (LAR 205), for review and approval prior to exceeding the licensed operating period for these curves. Approval of the revised pressure-temperature curves was obtained under the NRC Safety Evaluation for the EPU License Amendments 249/245, issued June 15, 2012. C26 C26

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, 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 reactor vessel internals and core support were designed prior to the creation of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NG, using internal Westinghouse design criteria that effectively evolved to become the original NG criteria. 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. C26 C26

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. For EPU conditions, the reactor internals baffle-former bolts have reduced design cycles in Table 4.1-8. 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. C26

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.

The analysis associated with the structural integrity of the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps and pressurizers surge lines were evaluated considering EPU conditions. Except for the baffle-former bolts, the maximum number of design cycles remains unchanged by the implementation of the EPU and are not projected to be exceeded through the period of extended operation. Fatigue usage factors remain acceptable with EPU conditions. As a result, these fatigue analyses remain valid with the implementation of the EPU through the period of extended operation. C26

### 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).

The Unit 3 and Unit 4 RVCHs were replaced. During the manufacturing process of the new forging and the cladding process of the new heads, precautions were taken to preclude the causes of underclad cracking. These precautions preclude the formation of segregated areas on the surface to be clad, the presence of stresses in the underclad heat affected zone (HAZ), and the presence of coarse grain areas in the cladding HAZ.

The analysis associated with reactor vessel underclad cracking was reviewed to consider EPU conditions. The original evaluation performed a fracture mechanics evaluation based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service life. The representative set of design transients remains unchanged with the implementation of the EPU. As a result, the original analysis associated with reactor vessel underclad cracking remains valid with the implementation of the EPU through the period of extended operation. C26

### 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).

The analysis associated with the reactor coolant pump flywheel was evaluated to consider EPU conditions. The results of this evaluation demonstrated that the original conclusions reached for license renewal remain valid with the implementation of the EPU. As a result, the analysis associated with the reactor coolant pump flywheel is valid with the implementation of the EPU through the period of extended operation. C26

### 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).

The original evaluation for license renewal determined that ANSI B31.1 piping would not exceed the number of assumed thermal cycles for 60 years of plant operation. The maximum number of thermal cycles remains unchanged by the implementation of the EPU. Operation of the piping at EPU conditions will not result in exceeding the number of assumed thermal cycles. As a result, the original evaluation remains valid with the implementation of the EPU through the period of extended operation. C26

### 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.

The evaluations used actual projected cycle counts for the reactor pressure vessel outlet nozzles, the reactor pressure vessel shell at core support pads, and the pressurizer spray nozzles. To demonstrate acceptable Cumulative Usage Factors (CUF) at these locations for a 60 yr time period, FPL applied more refined fatigue evaluations using the ANSYS computer program. For the reactor pressure vessel outlet nozzles and the reactor pressure vessel shell at core support pads, the CUF for Environmentally Assisted Fatigue were demonstrated to be less than 1.0. For the pressurizer spray nozzles the CUF for Environmentally Assisted Fatigue were demonstrated to be less than 1.0 based on the application of a multiplication factor of four in accordance with the NRC Safety Evaluation Report, NUREG 1759, Section 4.3.2. C26

For the pressurizer surge line, FPL inspected all surge line welds on both units during the fourth inservice inspection interval, and prior to entering the extended period of operation. The results of these inspections were utilized to assess fatigue of the surge lines. In addition to these inspections, environmentally assisted fatigue of the surge lines is addressed by an NRC approved inspection program. The aging management of the surge line will be accomplished by a combination of flaw tolerance analysis and in-service inspection. The surge line welds that are in scope will be inspected in accordance with the requirements of ASME Section XI, Subsection IWB, using the Turkey Point In-service Inspection Program. The technical justification and inspection frequency are supported by the flaw tolerance analysis based on the methodology noted in ASME Section XI, Nonmandatory Appendix L, "Operating Plant Fatigue Assessment".

### 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 was conducted under the Thimble Tube Inspection Program, described in Subsection 16.2.16. Based on the results of this inspection, this program was converted to a periodic inspection program. 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). C30

### 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 initial operating period of 40 years. In order to ensure emergency containment cooler tube reliability, a one-time inspection for minimum tube wall thickness was conducted on a sample of cooler tubes prior to the end of the initial operating period to further assess the actual tube wall thinning. The inspection was conducted in accordance with the Emergency Containment Coolers Inspection, described in Subsection 16.1.3.

The Emergency Containment Coolers Inspection ensures 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).

For the EPU, the Reactor Coolant primary loop piping Leak-Before-Break analysis was evaluated considering EPU conditions. The evaluation demonstrated that the Leak-Before-Break conclusion provided in the existing Leak-Before-Break analysis developed for Turkey Point Units 3 and 4 are unchanged for the EPU conditions. Therefore, the Reactor Coolant primary loop piping Leak-Before-Break analysis remains valid with the implementation of the EPU for the period of extended operation C26

### 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).