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CHAPTER 18

AGING MANAGEMENT PROGRAMS AND TIME-LIMITED AGING ANALYSES ACTIVITIES

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CHAPTER 18

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18.0 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 and completed implementation.

FPL has established and implemented a Quality Assurance Program discussed in Section 17.2 to provide assurance that the design, procurement, modification and operation of nuclear power plants conform to applicable regulatory requirements. For all aging management programs credited for license renewal, the program attributes of Corrective Actions, Confirmation, and Administrative Controls are performed or, in the case of new programs will be performed, in accordance with the FPL Quality Assurance Program, and will apply to all components and structural components within the scope of the programs, including non safety-related components and structural components.

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 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 St. Lucie Unit 1.

18.1 NEW PROGRAMS

18.1.1 CONDENSATE STORAGE TANK CROSS-CONNECT BURIED PIPING INSPECTION

A one-time visual inspection was attempted with the intent to determine the extent of the loss of material due to pitting and microbiologically influenced corrosion on the external surfaces of the buried piping that connects the St. Lucie Unit 1 and Unit 2 condensate storage tanks prior to the end of the initial operating license term for St. Lucie. The subject pipe for the Condensate Storage Tank Cross-Connect Buried Piping Inspection was found encased in a concrete duct and was inaccessible for inspection. Corrosion/degradation of embedded metals is not an applicable aging effect. Therefore, no further inspections are required during the extended period of operation. The Condensate Storage Tank Cross-Connect Buried Piping Inspection (Unit 1 only) is a One Time Inspection Program only.

18.1.2 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 surfaces of susceptible piping and components. The program involved selected, one-time inspections on the surfaces of piping and components with the greatest susceptibility to galvanic corrosion. Baseline examinations (visual inspection or volumetric examinations) in select systems were performed and evaluated to establish if the corrosion mechanism is active. Evaluation of the inspection results considered the minimum required wall thickness for the component consistent with the applicable design codes. Based on the results of these inspections, follow-up examinations or programmatic corrective actions are not required. The program was implemented prior to the end of the initial operating license term for St. Lucie Unit 1.

18.1.3 PIPE WALL THINNING INSPECTION PROGRAM

The Pipe Wall Thinning Inspection Program manages the aging effect of localized loss of material due to erosion of the internal surfaces of stainless steel Auxiliary Feedwater System piping downstream of the recirculation orifices. Examinations will be performed using volumetric techniques such as ultrasonic testing or radiography. This program has been implemented prior to the end of the initial operating license term for St. Lucie Unit 1. Pipe Wall Thinning Inspection Program is a continuing program.

18.1.4 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The Reactor Vessel Internals (RVI) Inspection Program manages the aging effects on the RVI during the period of extended operation. The RVI consists of three major structural assemblies, plus three other sets of major components. The three major assemblies include: 1) upper internals assembly; 2) core support barrel assembly, and 3) lower internals assembly. In addition, the three other sets of major components are the control element assembly (CEA) shroud assemblies, core shroud assembly, and in-core instrumentation support system. The RVI Inspection Program is applicable to passive RVI structural components and specifically excludes welded attachments to the reactor vessel and consumable items such as fuel assemblies, control element assemblies (CEAs) and in core instrumentation (ICI).

Aging effects and the causative degradation mechanisms addressed by the RVI Inspection Program include: 1) cracking due to stress corrosion cracking (SCC), irradiation assisted stress corrosion cracking (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.

The RVI Inspection Program is based upon the guidance provided in the latest NRC accepted revision of EPRI MRP-227, “EPRI Materials Reliability Program, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines.” The RVI Inspection Program 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 RV Internals. EC294622

FPL has satisfied the following commitments concerning the RVI Inspection Program: 1) Submit an integrated report for St. Lucie Units 1 and 2 to the NRC prior to the end of the initial operating license term for St. Lucie Unit 1 that summarizes its understanding of the aging effects applicable to the reactor vessel internals and contains a description of the St. Lucie inspection plan, including methods for detection and sizing of cracks and acceptance criteria; 2) Adopt the latest NRC accepted version of MPR-227 in place of the previously approved RVI Inspection Program that was included in the St. Lucie Units 1 and 2 License Renewal Applications; and 3) Perform a one time inspection of the reactor vessel internals, which was completed March, 2018. EC294622

18.1.5 SMALL BORE CLASS 1 PIPING INSPECTION

This program was a one-time inspection of a sample of ASME Code Class 1 piping less than NPS 4 and greater than or equal to NPS 1. Inspection of a sample of small bore Class 1 piping was performed to determine if cracking is an aging effect requiring management during the period of extended operation.

The inspection was performed prior to the end of the initial operating license term for St. Lucie Unit 1. No service induced defects were found during the inspections. Therefore, additional inspections or programmatic corrective actions are not required. EC290827

A report describing the details of the inspection plan was submitted to the NRC prior to the implementation of this inspection. The inspection plan addressed how the risk informed methodology was applied.

This program was consistent with the ten attributes of the aging management program (AMP) XI.M35, "One Time Inspection of ASME Class 1 Small Bore-Piping" specified in NUREG-1801, GALL Report Revision 2, with the exception that a destructive examination was performed on a butt weld instead of volumetric examination, when an opportunity presented itself. EC290827

The GALL recognizes that because more information can be obtained from a destructive examination than from nondestructive examination, the applicant may take credit for each partial penetration weld destructively examined equivalent to having volumetrically examined two welds. If an acceptable volumetric technique is not available to perform a volumetric inspection of partial penetration welds, a destructive examination will be performed. St. Lucie took credit for destructively examining partial penetration (socket) welds.

For the same reason that destructive examination yields more information, destructive examination of full penetration welds was used in lieu of non-destructive volumetric examination on one butt weld. EC290827

18.1.6 THERMAL AGING EMBRITTLEMENT OF CASS PROGRAM

The St. Lucie Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program Includes a determination of the susceptibility of Class 1 CASS piping components to thermal aging embrittlement and provides the requirements for subsequent aging management of those components that have been identified as being potentially susceptible. This aging management program (AMP) performed a determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite, and identifies "potentially susceptible" components. In accordance with the Generic Aging Lessons Learned (GALL) Revision 2, Aging Management Program (AMP) XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," for "potentially susceptible" components, aging management is accomplished through either ;

(a) qualified visual inspections, such as enhanced visual examination (EVT-1);

(b) a qualified ultrasonic testing (UT) methodology; or

(c) a component-specific flaw tolerance evaluation in accordance with ASME Code, Section XI, 2004 edition.

Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components that are not susceptible to thermal aging embrittlement.

A Component-specific flaw tolerance evaluation in accordance with ASME Code, Section XI, 2004 edition (i.e., Option (c)) was performed which demonstrates the final projected flaw sizes are within the maximum tolerable flaw.

This program is consistent with the ten attributes of aging management program X1.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," specified in NUREG-1801, GALL Report, Revision 2.

18.1.7 CONTAINMENT CABLE INSPECTION PROGRAM

The Containment Cable Inspection Program manages the potential aging of non-EQ cables and connections. This program includes non-EQ cables and connections associated with sensitive low-level signal circuits. The only non-EQ cables and connections associated with sensitive low-level signal circuits within the scope of license renewal for St. Lucie are those associated with the neutron detectors. This aging management program consists of periodic visual inspection of accessible non-EQ cables and connections within the scope of license renewal located in the containment that may be installed in adverse localized environments, and review of calibration tests results for indication of age related degradation of cables associated with the neutron detectors. The inspections were implemented prior to the end of the initial operating license term for St. Lucie Unit 1.

NRC RIS 2003-09 issued the NRC's final resolution to Generic Safety Issue (GSI)-168. Although the RIS indicated that no new actions were mandated, the NRC indicated that there were some concerns over low voltage I&C EQ cables. To address these concerns, FPL decided to add low voltage I&C EQ cables to the Containment Cable Inspection Program.

18.1.8 METAMIC™ INSERT SURVEILLANCE PROGRAM (References 1 and 2)

The purpose of the Metamic™ insert surveillance program is to ensure the Metamic™ panels continue to meet the licensing bases requirements. This will be done by validating that physical and chemical properties of Metamic™ behave in a similar manner in-situ as portrayed in the pre-installation qualification data. The surveillance program will monitor how Metamic™ absorber material properties behave over time as a result of the radiation, chemical, and thermal environment found in the spent fuel pool.

There are three (3) essential elements in the Metamic™ surveillance program:
1. Visual inspection of the Metamic™ inserts
2. Physical measurement of Metamic™ coupons
3. Neutron attenuation testing of Metamic™ coupons

1. Visual Inspection

Visual inspections of Metamic™ inserts are performed to assess the physical condition of the Metamic™ material. Key goals of these inspections are to identify any surface-based abnormalities such as through-wall corrosion/damage, bubbling, blistering, corrosion pitting, cracking or flaking.

Visual inspection of five (5) Metamic™ inserts will be performed at 4, 8, 12, 20, and 30 years following initial installation of Metamic™. Visual inspection of the Metamic™ inserts is based on the previous testing performed by EPRI and Holtec International. The results of these tests indicated a potential for local corrosion pitting on the surface of the Metamic™.

This was later traced to surface contaminants left on the Metamic™ during either extrusion or rolling. While the EPRI testing pointed to the need to thoroughly clean and/or anodize the Metamic™ surfaces, only glass bead cleaning was used, as anodizing was extremely difficult due to the length of each chevron. Therefore, visual inspection is implemented to check for signs of corrosion pitting.

Should insert anomalies be noted on the visual inspections, then an additional set of five inserts will be inspected. Issues identified during the visual inspections will be included in the corrective action program for investigation and resolution.

2. Physical Measurements

Physical measurements of two (2) Metamic™ coupons are used to confirm the stability of the dimensions of Metamic™ material as noted in the testing. These measurements confirm the absence of swelling and shrinkage.

Weight and physical dimensional measurements (length, width, and thickness) of Metamic™ coupons will be performed at 4, 12, 20, and 30 years following initial installation of Metamic™.

18.1.8 METAMIC^{TM} INSERT SURVEILLANCE PROGRAM (continued)

Should physical measurement inspections of the coupons result in a failure to meet acceptance criteria for thickness, then an additional two coupons will be inspected. Issues identified during physical measurements inspection will be included in the corrective action program for investigation and resolution.

3. Neutron Attenuation Testing

   Neutron attenuation testing is required to provide a periodic validation of certain assumptions embedded in the fuel pool rack’s criticality analysis, and to also confirm that the neutron absorption capability of Metamic™ would remain unchanged throughout its service lifetime.

   While this has been demonstrated through the earlier Metamic™ qualification program, 10 smaller coupons will be added to the pool, from which 2 coupons would be periodically retrieved for each of the 4 scheduled neutron attenuation tests with 2 spares remaining. The tested coupons will not be returned to the spent fuel pool.

   Neutron attenuation testing of coupons will be performed at 4, 12, 20, and 30 years following initial installation of Metamic™. The Metamic neutron absorber inserts have a ^{10}B areal density greater than or equal to 0.015 grams $^{10}B/cm^2$. EC 299 478

   Should a coupon fail to meet acceptance criteria during neutron attenuation testing, then an additional two coupons will be tested. Issues identified during neutron attenuation testing will be included in the corrective action program for investigation and resolution.

Anomalies and Corrective Actions

Each of the elements of the surveillance program described above will have pre-established acceptance criteria, which will be used to reject any Metamic™ inserts that do not meet the minimum functionality requirements of the material or the assumptions of the SFP rack criticality analysis.

Should any of the data obtained during surveillances fail to meet the established acceptance criteria, the subject insert(s) shall be removed from service and replacements(s) installed, as necessary, to comply with the criticality analysis requirements. Any anomalies and discrepancies identified during the testing will be entered into the corrective action program for proper evaluation and corrective actions.

Some surface scratches and local marks in the Metamic™ inserts are expected and are not necessarily indicative of an out-of-specification condition.

Reference(s)

1) Letter L-2011-253, Additional information related to Metamic™ inserts (ADAMS Accession No. ML11364A044)
2) Extended Power Uprate License Amendment Request and included in the Nuclear Regulatory Commission Safety Evaluation Report(s) date July 9, 2012 for St. Lucie Plant Unit 1 (ML12181A019) and September 24, 2012 for St. Lucie Unit 2 (ML12235A463).

18.2 EXISTING PROGRAMS

18.2.1 ALLOY 600 INSPECTION PROGRAM

This program manages the aging effect of cracking due to primary water stress corrosion for susceptible Alloy 600 components within the Reactor Coolant System (RCS) pressure boundary.

The St Lucie Plant Alloy 600 inspection program is based on the requirements of 10 CFR 50.55a(g)(6)(ii)(E) "Reactor coolant pressure boundary visual inspections" (i.e., ASME Code Case N-722)* and (F) "Examation requirements for class 1 piping and nozzle dissimilar-metal butt welds" (i.e., ASME Code Case N-770)*. EC291839

Visual examination of external surfaces of susceptible locations during outages, which is included as part of the Boric Acid Wastage Surveillance Program, is also utilized to manage cracking.

Alloy 600 components which have been replaced with Alloy 690 are removed from the program.

The reactor vessel head was replaced and no longer contains susceptible Alloy 600 components. The reactor vessel head inspections are completed in accordance with 10 CFR 50.55a(g)(6)(ii)(D) "Augmented ISI requirements -Reactor Vessel Head Inspections" (i.e., ASME Code Case N–729)* which is separate from the Alloy 600 Program that is implemented per 10 CFR 50.55a(g)(6)(ii) (E). EC291839

NUREG-1779, "Safety Evaluation Related to the License Renewal of St. Lucie Nuclear Plant, Units 1 and 2", Appendix D Table 1 Commitment 9 reads: "Perform Inspections and examinations of the reactor vessel, incorporate NRC requirements, FPL responses to NRC IE Bulletins, and industry recommendations including the ERPI Materials Reliability Project." This commitment was associated with the Alloy 600 inspection program described in this UFSAR Section and has been completed. EC291839

10 CFR 50.55a(g)(6)(ii)(D) states that Licensees of existing operating reactors as of September 10, 2008, must implement their augmented inservice inspection program by December 31, 2008. Once a licensee implements this requirement, the First Revised NRC Order EA–03–009 no longer applies to that licensee and shall be deemed to be withdrawn. As the First Revised NRC Order EA–03–009 no longer applies, the previous commitments associated with the Order are replaced by the updated Alloy 600 program and the reactor vessel inspections completed under the ISI program in accordance with 10 CFR 50.55a(g)(6)(ii)(D).

*The latest code case endorsed by 10CFR50.55a. EC291839

18.2.2 ASME SECTION XI INSERVICE INSPECTION PROGRAMS

18.2.2.1 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, loss of preload, reduction in fracture toughness, and loss of mechanical closure integrity. This program is consistent with the ten attributes of aging management program X1.M1, "ASME Section XI Inservice Inspections, Subsections IWB, IWC and IWD" specified in NUREG-1801, GALL Report Revision 2.

The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program was enhanced to require VT-1 inspections of the core stabilizing lugs and core support lugs and to require evaluation of surge line flaws (if identified) with regard to environmentally assisted fatigue (Reference Section 18.3.2.3). This action was implemented prior to the end of the initial operating license term for St. Lucie Unit 1.

18.2.2.2 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 to the Class MC steel Containment. The program manages the aging effects of loss of material and loss of seal. This program is consistent with the ten attributes of aging management programs X1.S1, "ASME Section XI, Subsection IWE" and X1.S4, "10 CFR Part 50 Appendix J" specified in NUREG-1801, GALL Report Revision 2 with the exception that XI.S8 "Protective Coatings Monitoring and Maintenance Program" is not implemented. The program provides for inspection and examination of Containment surfaces, pressure-retaining welds, seals, gaskets and moisture barriers, pressure-retaining bolting, and pressure-retaining components in accordance with the requirements of ASME Section XI, Subsection IWE. EC290874

18.2.2.3 <u>ASME Section XI, Subsection IWF Inservice Inspection Program</u>

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. This program is consistent with the ten attributes of aging management program X1.S3, “ASME Section XI, Subsection IWF” specified in NUREG-1801, GALL Report Revision 2. 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.

18.2.3 DELETED

18.2.4 BORIC ACID WASTAGE SURVEILLANCE PROGRAM

The Boric Acid Wastage Surveillance Program manages the aging effects of loss of material and loss of mechanical closure integrity due to aggressive chemical attack resulting from borated water leaks. The program addresses the RCS, other systems which contain borated water and structures and components containing, or exposed to, borated water. The program includes portions of the waste management system which is included in the scope of License Renewal This program is consistent with the ten attributes of aging management programs X1.M10, “Boric Acid Corrosion” specified in NUREG-1801, GALL Report Revision 2. This 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 the structural integrity of components, supports, or structures in proximity to borated water systems. This program includes commitments in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

18.2.5 CHEMISTRY CONTROL PROGRAM

The Chemistry Control Program manages the aging effects of loss of material, cracking, and fouling for primary and secondary systems, closed cooling water, and fuel oil 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 effects. The program includes sampling activities and analysis. The program provides assurance that elevated levels of contaminants and oxygen do not exist in the systems, structures, and components covered by the program, and thus prevents and minimizes the occurrences of aging effects.

The Chemistry Control Program – Water Chemistry Control Subprogram is consistent with ten attributes of aging management program XI.M2, “Water Chemistry” specified in NUREG-1801, GALL Report Revision 2, except that the GALL program credits inspection of select components to verify the effectiveness of the chemistry control program and to ensure that significant degradation is not occurring and the component intended function will be maintained during the period of extended operation. No special one-time inspections are required to be performed at St. Lucie. Further, the program will continue to upgrade to later revisions of the EPRI Pressurized Water Reactor Primary and Secondary Water Chemistry Guidelines as compared to the revisions referenced in GALL Revision 2.

18.2.5 CHEMISTRY CONTROL PROGRAM (continued)

The Chemistry Control Program – Closed-Cycle Cooling Water System Chemistry Subprogram is consistent with the ten attributes of aging management program XI.M21, "Closed-Cycle Cooling Water System" specified in NUREG-1801, GALL Report Revision 2, except that the program does not address surveillance testing and inspection. Further, the program will continue to upgrade to later revisions of the EPRI Closed Cooling Water Chemistry Guideline as compared to the revision referenced in GALL Revision 2.

The Intake Cooling Water Inspection Program Section 18.2.10 implements the applicable surveillance testing and inspection aspects of NUREG-1801, GALL program.

The Chemistry Control Program - Fuel Oil Chemistry Subprogram is plant-specific.

18.2.6 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification Program is an aging management program. For those passive components where analysis cannot justify a qualified life in excess of the period of operation allowed by the renewed facility operating license, then the component parts will be replaced, refurbished, or requalified prior to exceeding their qualified lives in accordance with the Environmental Qualification Program. This program is consistent with the ten attributes of aging management programs X.E1, Environmental Qualification (EQ) of Electrical Components specified in NUREG-1801, GALL Report Revision 2.

Where analysis demonstrates the equipment is qualified for the period of operation allowed by the renewed facility operating license the analysis is considered a Time Limited Aging Analysis (TLAA) as described in Subsection 18.3.3.

18.2.7 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program is considered a confirmatory program to ensure that fatigue time-limited aging analysis assumptions remain valid for the period of extended operation; it is not credited as an aging management program.

The Fatigue Monitoring Program is designed to track design cycles to ensure that RCS components remain within their design fatigue limits. Design cycle limits for St. Lucie Unit 1 are provided in Sections 3.9.2.2, 5.2.1.2, and 5.5.1.1. The specific fatigue analyses validated by the Fatigue Monitoring Program are associated with the reactor vessel, reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, and RCS Class 1 piping. Administrative procedures provide the methodology for logging design cycles. These procedures were enhanced to provide guidance in the event design cycle limits are approached. This enhancement was completed prior to the end of the initial operating license term for St. Lucie Unit 1. Environmentally Assisted Fatigue of the pressurizer surge line is discussed in Section 18.3.2.3.

18.2.8 FIRE PROTECTION PROGRAM

The Fire Protection Program manages the aging effect of loss of material for the components of the Fire Protection System. Additionally, this program manages the aging effect of loss of material for structural components associated with fire protection. Fire Protection Design Basis Document, DBD-FP-1 contains a detailed discussion of the Fire Protection Program. Additionally, St. Lucie Unit 1 will perform testing of wet pipe sprinkler heads following the guidance of NFPA 25 (1998) commencing in the year 2026 or replace sprinkler heads which have been in service for 50 years.

18.2.9 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 in high energy carbon steel piping associated with the Main Steam, Reactor Coolant (steam generators), Main Feedwater and Blowdown Systems, and is based on industry guidelines and experience. This program is consistent with the ten attributes of aging management programs X1.M17, “Flow-Accelerated Corrosion” specified in NUREG-1801, GALL Report Revision 2. The program includes analysis and baseline inspections; determination, evaluation, and corrective actions for affected components; and follow-up inspections.

The Flow Accelerated Corrosion Program was enhanced to address internal and external loss of material of drain lines and selected steam trap lines due to flow accelerated corrosion and external general corrosion. The enhancement was completed prior to the end of the initial operating license term for St. Lucie Unit 1.

18.2.10 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, and particulate and biological fouling for Intake Cooling Water (ICW) System components and the ICW side of the Component Cooling Water heat exchangers. The program includes inspections, performance testing, evaluations, and corrective actions that are performed as a result of FPL commitments in response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

18.2.11 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

The Periodic Surveillance and Preventive Maintenance Program manages the aging effects of loss of material, cracking, loss of seal, and fouling (mechanical components only) for various plant systems, structures, and components. The scope of the program provides for visual examination of selected surfaces of specific systems, structures, and components. Additionally, the program provides for replacement/"refurbishment of selected components on a specified frequency, as appropriate, and periodic sampling and water removal from fuel oil storage tanks. The frequency of inspections varies depending on the specific component, the aging effect being managed, and plant operating experience.

Specific enhancements to the scope of this program were implemented prior to the end of the initial operating license term for St. Lucie Unit 1 by providing for inspection of components such as selected filter housings, radiator fins, flexible hoses, door seals and expansion joints.

18.2.12 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 aspects of the Reactor Vessel Integrity Program prior to the end of the initial operating license term for St. Lucie Unit 1.

18.2.12.1 Reactor Vessel Surveillance Capsule Removal and Evaluation

This subprogram manages the aging effect of reduction in fracture toughness of the reactor vessel materials (beltline plates and 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 an NRC-approved program that meets the requirements of 10 CFR 50, Appendix H. The surveillance capsule withdrawal schedule is specified in Table 5.4-3.

18.2.12.2 Fluence and Uncertainty Calculations

This subprogram provides an accurate prediction of the reactor vessel accumulated fast neutron fluence values at the reactor vessel beltline plates and welds.

18.2.12.3 Monitoring Effective Full Power Years

This subprogram accurately monitors and tabulates the accumulated operating time experienced by the reactor vessel to ensure that the pressure-temperature limits and end-of-life reference temperatures are not exceeded.

18.2.12.4 Pressure-Temperature Limit Curves

This subprogram provides pressure-temperature limit curves for the reactor vessel to establish the RCS operating limits. The pressure-temperature limit curves are included in the Technical Specifications.

18.2.13 STEAM GENERATOR INTEGRITY PROGRAM

The Steam Generator Integrity Program is consistent with the guidelines provided by the Nuclear Energy Institute’s NEI 97-06, "Steam Generator Program Guidelines." The program ensures that 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. This program is consistent with the ten attributes of aging management program XI.M19, “Steam Generator Tube Integrity” specified in NUREG-1801, GALL Report Revision 2 with the exception that the program will continue to upgrade to approved revisions of NEI 97-06 and the EPRI Pressurized Water Reactor Primary and Secondary Water Chemistry Guidelines. EC290874

18.2.14 SYSTEMS AND STRUCTURES MONITORING PROGRAM

The Systems and Structures Monitoring Program manages the aging effects of loss of material, cracking, fouling (for mechanical components only), 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 was enhanced to provide guidance for managing the aging effects of inaccessible concrete, inspection of insulated equipment and piping, and evaluating masonry wall degradation and uniform corrosion. These enhancements were made prior to the end of the initial operating license term for St. Lucie Unit 1.

18.3 TIME-LIMITED AGING ANALYSIS ACTIVITIES

18.3.1 REACTOR VESSEL IRRADIATION EMBRITTLEMENT

The St. Lucie Unit 1 reactor vessel is described in Chapters 4 and 5. Time-limited aging analyses (TLAAs) applicable to the reactor vessel are:

- pressurized thermal shock
- upper-shelf energy
- pressure-temperature limits

The Reactor Vessel Integrity Program, described in Section 18.2.12, 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 limits to ensure continuing vessel integrity through the period of extended operation.

18.3.1.1 <u>Pressurized Thermal Shock</u>

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.

RT_{PTS} values calculated for the St. Lucie Unit 1 reactor vessel beltline materials at fluence values ($E > 1.0$ MeV) projected for 60 years (52 EFPY) were re-evaluated for the Extended Power Uprate (EPU). The highest projected value of RT_{PTS} is 234°F and corresponds to the St. Lucie Unit 1 lower shell axial weld 3-203 A/C.

The calculated RT_{PTS} values that bound the 60-year period of operation for the St. Lucie Unit 1 reactor vessel 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 reactor vessel 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).

18.3.1.2 <u>Upper-Shelf Energy</u>

The requirements on reactor vessel Charpy upper-shelf energy (USE) 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 USE of any reactor vessel material is predicted to drop below 50 ft-lbs, as measured by Charpy V-notch specimen testing.

An evaluation was performed to demonstrate continued acceptable margins of safety against fracture through the end of the period of extended operation. In the assessment of the impact of the EPU on the USE, USE values were projected to EOL for all reactor vessel beltline materials using the EPU fluence projections. All of the beltline materials are projected to have a USE greater than 50 ft-lb through EOL (52 EFPY) in compliance with 10 CFR 50, Appendix G.

The analysis associated with USE 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).

18.3.1.3 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. Therefore, in order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within the limits of Appendix G defined by the reactor vessel fluence.

The heatup and cooldown pressure-temperature limits are presented in the Unit 1 Technical Specifications. The effect of the EPU on the applicability of the current St. Lucie 1 P-T limit curves was evaluated by comparing the adjusted reference temperature (ART) values for the current licensing basis with the ART values after the EPU. New P-T limits were generated for operation to 60 years (based on 54 EFPY fluence projection). In addition, low temperature overpressure protection (LTOP) requirements were updated for EPU to ensure that the pressure-temperature limits are not exceeded for postulated plant transients.

The analyses associated with reactor vessel pressure-temperature limits and LTOP for St. Lucie Unit 1 was available prior to entering the period of extended operation, and submitted to the NRC. The analysis report is in accordance with the requirements of the Reactor Vessel Integrity Program and consistent with 10 CFR 54.21(c)(1)(ii). EC290827

18.3.2 METAL FATIGUE

The thermal and mechanical fatigue analyses of plant mechanical components have been identified as time-limited aging analyses for St. Lucie Unit 1. Specific components have been designed considering design cycle assumptions, as listed in vendor specifications and in Sections 3.9.2 and 5.2.1.2.

18.3.2.1 ASME Boiler and Pressure Vessel Code, Section III, Class 1 Components

The reactor vessel (including control element drive mechanisms), reactor vessel internals, pressurizer, steam generators, and the reactor coolant pumps have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. The reactor coolant piping was originally designed in accordance with ANSI B 31.7, "Nuclear Power Piping." The pressurizer surge line was reanalyzed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, in response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification." These design codes require a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Reactor vessel internals fatigue is addressed in Section 18.3.7.1.

Fatigue usage factors for critical locations in the St. Lucie Unit 1 Nuclear Steam Supply System Class 1 components were determined using design cycles that were specified in the plant design process or as a result of industry fatigue issues (e.g., thermal stratification). 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 the Class 1 components satisfying ASME fatigue usage design requirements.

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.

18.3.2.1 **ASME Boiler and Pressure Vessel Code, Section III, Class 1 Components** (continued)

The actual frequency of occurrence for the fatigue sensitive design cycles was determined and compared to the design cycle set. The severity of the actual plant cycles was also compared to the severity of the design cycles. These comparisons were performed in order to demonstrate that on an event-by-event basis the design cycle profiles envelope actual plant operation. In the case of Loss of Letdown, additional analysis was performed to increase design cycles to bound actual frequency of occurrence. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the Fatigue Monitoring Program. The reviews described above concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation. EC 0296766

The analyses associated with verifying the structural integrity of the Class 1 components have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i) & (ii). EC 0296766

For license renewal, continuation of the 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.

18.3.2.2 **ASME Boiler and Pressure Vessel Code, Section III, Class 2 and 3 and ANSI B31.1 Components**

St. Lucie Unit 1 has a number of piping systems within the scope of license renewal that were originally designed to the requirements of ANSI B31.7, Class 2 and 3, or ANSI B31.1, "Power Piping." Subsequently, piping systems originally designed to the requirements of ANSI B31.7, Class 2 and 3, were reconciled to ASME Section III, Class 2 and 3. Piping systems designed to these requirements include a stress range reduction factor to provide conservatism in the design to account for cyclic conditions due to operations. The stress range reduction factor is 1.0 as long as the location does not exceed 7000 full temperature thermal cycles during its operation. 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.

A review of ASME Section III, Class 2 and 3, and ANSI B31.1 piping within the scope of license renewal was undertaken in order to establish the cyclic operating practices of those systems that operate at elevated temperatures. Based on industry guidance, any piping system with operating temperatures less than 220°F (carbon steel) or 270°F (stainless steel) may be conservatively excluded from further consideration of thermal fatigue.

Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subjected to cyclic operation. Typically these systems are subjected to continuous steady-state operation and operating temperatures vary only during plant heatup and cooldown, during plant transients, or for periodic testing. The review of applicable plant systems determined that, except for the RCS hot-leg sample piping, components will not exceed 7000 equivalent full temperature thermal cycles during the period of extended operation. Therefore, the current piping analyses remain valid for the period of extended operation.

The RCS hot-leg sample lines could exceed the 7000 equivalent full temperature thermal cycles during the period of extended operation based on the current sampling practices. The sample piping and tubing were re-evaluated to consider the projected number of cycles and the analyses were found acceptable for the period of extended operation.

Except for the RCS hot-leg sample lines, the ASME Section III, Class 2 and 3, and ANSI B31.1 piping fatigue analyses within the scope of license renewal remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The RCS hot-leg sample lines' fatigue analyses have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

18.3.2.3 Environmentally Assisted Fatigue

Generic Safety Issue (GSI) 190 was initiated by the NRC staff because of concerns about the potential effects of reactor water environments on RCS component fatigue life during the period of extended operation. The FPL 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 that 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 included in NUREG/CR-6260, especially for the "Older Vintage Combustion Engineering Plant," match St. Lucie relatively closely with respect to design codes used, as well as the analytical approach and techniques used. In addition, the design cycles considered in the evaluation match or bound the St. Lucie Unit 1 design.

Environmental fatigue calculations have been performed for St. Lucie Unit 1 for those component locations included in NUREG/CR-6260 using the appropriate methods contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998, or NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999, as appropriate. Based on these results, all component locations were determined to be acceptable for the period of extended operation, with the exception of the pressurizer surge line (specifically the surge line elbow below the pressurizer).

An aging management program (AMP) was submitted to the NRC for their approval via FPL letter L-2015-272. The NRC approved the EAF analysis and the aging management program for PSL in SER dated October 13, 2016 (Ref: 1).

The Aging Management Program (AMP) is accomplished by a combination of flaw tolerance analysis as per ASME Boiler & Pressure Vessel Code, Section XI, 2001 Edition, with Addenda through 2003, Appendix L and inspection under the PSL In-Service Inspection Program.

The aging effect managed is cracking due to environmentally assisted fatigue (EAF). The initial inspection, technical justification and subsequent inspection frequency are supported by a flaw tolerance analysis based on the methodology noted in ASME Section XI, Non-mandatory Appendix L, “Operating Plant Fatigue Assessment”. Based on postulated flaw tolerance analysis, and per the guidelines of ASME Code, Section XI, Appendix L, Table L-3420-1, the successive inspection interval is determined to be ten years.

During PEO, the pressurizer surge line welds which were analyzed as part of the ASME Appendix L fatigue analysis, will be examined in accordance with ASME Section XI, IWB for Class 1 welds as per the Pressurizer Surge Line Aging Management Program. The ASME Appendix L Environmental Fatigue Analysis screened out welds with overlays (i.e., FSWOL), as such those welds are not included in the scope of inspection for AMP.

Reference(s):

1. NRC SER, St. Lucie Nuclear Plant, Units 1 and 2 - Review Of License Renewal Commitment For Pressurizer Surge Line Welds Inspection Program (ML16235A138)

18.3.3 ENVIRONMENTAL QUALIFICATION

The thermal, radiation, and wear cycle aging analyses of plant electrical and I&C components have been identified as TLAAs for St. Lucie Unit 1. In particular, the environmental qualification evaluations of electrical equipment with a 40-year qualified life or greater have been determined to be TLAAs.

Equipment included in the St. Lucie 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 as it is for equipment currently qualified at St. Lucie 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 Section 18.2.6.

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)(ii).

18.3.4 CONTAINMENT PENETRATION FATIGUE

Containment penetration bellows are specified to withstand a lifetime total of 7000 cycles of expansion and compression due to maximum operating thermal expansion, and 200 cycles of other movements (seismic motion and differential settlement).

The containment penetrations are categorized as follows:

- Type I — Those which must accommodate considerable thermal movements (hot penetrations)
- Type II — Those which are not required to accommodate thermal movements (cold penetrations)
- Type III — Those which must accommodate moderate thermal movements (semi-hot penetrations)
- Type IV — Containment sump recirculation suction lines
- Type V — Fuel transfer tube

The containment penetration bellows fatigue analyses have been evaluated and determined to remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

18.3.5 LEAK-BEFORE-BREAK FOR REACTOR COOLANT SYSTEM PIPING

A Leak-Before-Break (LBB) analysis was performed for Combustion Engineering designed Nuclear Steam Supply Systems (NSSS), which included St. Lucie Unit 1. The LBB analysis was performed to show that any potential leaks that develop in the RCS primary coolant 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 a March 5, 1993, NRC letter to FPL, the NRC approved the St. Lucie LBB analysis. The NRC safety evaluation concluded that since the St. Lucie Units are bounded by the Combustion Engineering Owners Group analyses and the leakage detection systems are capable of detecting the specified leakage rate, the dynamic effects associated with postulated pipe breaks in the primary coolant system piping can be excluded from the licensing and design bases of the St. Lucie Units.

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 RCS loop piping is embrittlement of the duplex ferritic cast austenitic stainless steel (CASS) components. This effect results in a reduction in fracture toughness of the material.

A review concluded that the LBB analysis used conservative material toughness properties relative to correlations developed for fully aged cast stainless steel, which covers the extended period of operation. Therefore, the thermal aging assumptions used for the CASS piping do not satisfy one of the six criteria for a TLAA.

The LBB fatigue crack growth analysis assumes 40-year design cycles. The plant design cycles are consistent with those utilized in the fatigue crack growth analysis and bound the period of extended operation. Fatigue crack growth for the period of extended operation is negligible.

The RCS primary loop piping LBB fatigue crack growth analysis has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

18.3.6 CRANE LOAD CYCLE LIMIT

The following cranes have load cycle assumptions that result in the fatigue analyses being TLAAs:

- Reactor Building Polar Crane
- Intake Structure Bridge Crane
- Reactor Containment Building Auxiliary Telescoping Jib Crane

(Note: Fuel handling equipment does not require a TLAA evaluation because its lifting function is not in the scope of license renewal.)

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).

18.3.7 CORE SUPPORT BARREL

During the 1983 refueling outage, the reactor vessel internals core support barrel (CSB) and thermal shield assembly were observed to be damaged. The thermal shield was permanently removed and the CSB was repaired at the thermal shield support lug locations. Four lugs were separated from the CSB and through-wall cracks were adjacent to some damaged lug areas. Through-wall cracks were arrested with crack arrestor holes, non-through-wall cracks were machined out, and lug tear out areas were machined and patched as necessary. The crack arrestor holes were sealed by inserting expandable plugs.

Analysis of the CSB repair method was performed by the NSSS supplier to demonstrate that the repair patches and expandable plug designs were acceptable for the remaining (40-year) life of the plant consistent with ASME code allowable stresses.

The analyses and follow-up inspection reports for the repaired CSB and the expandable plugs were screened against the six TLAA criteria. It was determined that two specific elements of the repair qualify as TLAAs: 1) fatigue analysis of the CSB middle cylinder; and 2) acceptance criteria for the CSB expandable plugs' preload based on irradiation-induced stress relaxation.

18.3.7.1 Core Support Barrel Fatigue Analysis

As discussed in Section 18.3.2.1, the 40-year design cycles bound the extended period of operation. Therefore the CSB fatigue analysis has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

18.3.7.2 Core Support Barrel Repair Plug Analysis

The repair plugs are of an expandable design that allows the plugs to be preloaded against the CSB. Preload is required to provide proper seating of the plugs and patches and to prevent movement of the plugs due to hydraulic drag loads. The original evaluation of plug design preload verified that the design preload was sufficient to accommodate normal operating hydraulic loads and thermal deflections for the original operating life of the plant.

The original CSB plug preload analysis was revised for increased fluence (60-year period of operation) and irradiation-induced relaxation input. The analysis concluded that all the repair plug flange deflection measurement readings are sufficient to meet the minimum required values and maintain the plugs' preload. The CSB repair plugs will therefore perform their intended function for the period of extended plant operation.

The CSB preload stress relaxation 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).

18.3.8 ALLOY 600 INSTRUMENT NOZZLE REPAIRS

Small diameter Alloy 600 nozzles, such as pressurizer and RCS hot-leg instrumentation nozzles in Combustion Engineering designed PWRs, have developed leaks or partial through-wall cracks as a result of primary water stress corrosion cracking. The residual stresses imposed by the partial penetration "J" welds between the nozzles and the low alloy or carbon steel pressure boundary components are the driving force for crack initiation and propagation.

A repair technique known as the "half nozzle" weld repair has been used to repair selected Alloy 600 instrument nozzles. In the half nozzle technique, the Alloy 600 nozzle is cut outboard of the partial penetration weld and replaced with a short Alloy 690 nozzle section that is welded to the outside surface of the pressure boundary component. This repair leaves a short section of the original nozzle attached to the inside surface with the "J" weld.

A half nozzle repair was implemented on a Unit 1 RCS hot-leg instrumentation nozzle in April 2001. In response to NRC questions regarding this repair, FPL documented that the indications in the "J" weld were bounded by the fracture mechanics analysis provided in Combustion Engineering Owner’s Group (CEOG) Topical Report CE NPSD-1198-P.

CEOG Topical Report CE NPSD-1198-P was submitted to the NRC February 15, 2001 to obtain generic approval of the Alloy 600/690 nozzle repair/replacement programs.
The CEOG report provides a bounding flaw evaluation that covers all small diameter Alloy 600/690 nozzle repairs in accordance with ASME Section XI requirements. Topical Report WCAP-15973 was submitted to revise CEOG Topical Report CE-NSPD-1198-P and was approved by the NRC on January 12, 2005.

As a result of issues identified in this Topical Report, a plant-specific analysis by Westinghouse, Report No. CN-CI-02-69 of the small bore nozzles located in the hot leg piping and the pressurizers was completed using plant-specific data. These nozzle locations, where half nozzle or similar repairs would be utilized, thereby leaving flaws in the original weldment which could potentially grow into adjacent ferritic material, were assessed. Postulated flaws were assessed for the flaw growth and flaw stability as specified in the ASME Code Section XI. The flaw growth analysis included in the report assumes the total number of design cycles, consistent with the St. Lucie Unit 1 UFSAR. This analysis bounds the Class 1 fatigue design requirements of St. Lucie Unit 1. As discussed in UFSAR Section 18.3.2.1, review of actual plant operation concludes that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation.

As part of the issuance of the safety evaluation report (SER) NUREG-1779, the NRC issued the following commitment, which is listed as commitment No. 21 in Appendix D of NUREG-1779.

> “Implement all reasonable alternative inspection / evaluation methods that may be required by the NRC, as appropriate, as conditions for approval of the relief request. Subsequent to the disposition of the relief request and prior to the period of extended operation, the TLAAs for the St. Lucie Units 1 and 2 half-nozzle replacement designs will be dispositioned pursuant to 10 CFR 54.21 (c)(1). These TLAAs shall address: 1) the potential growth of the original flaw due to thermal or mechanical cycling, and 2) the potential wastage of the ferritic material that is adjacent to the half-nozzle configuration and exposed to borated reactor coolant. If acceptability of the St. Lucie Units 1 and 2 half-nozzle designs cannot be demonstrated for the period of extended operation pursuant to 10 CFR 54.21 (c)(1)(i) or 54.21 (c)(1)(ii), then these TLAAs will be dispositioned in accordance with 10 CFR 54.21 (c)(1)(iii) which may include appropriate nozzle replacement to comply with ASME Section III and ASME Section XI replacement criteria”.

By Implementation of the above commitment, NRC Open Item No. 4.6.4-1 noted in Section 1 of the NRC SER – NUREG 1779 is closed.

The TLAA analyses for two items noted in the above commitment have been completed in St. Lucie plant specific calculation CN-CI-02-69, WCAP-15973-P-A, ASME Relief Request no.26 for the 3rd Interval, and in ASME Relief request no. 6 for the 4th interval.

1. St. Lucie plant specific calculation CN-CI-02-69 was submitted by letter L-2002-222.

2. WCAP-15973 was submitted to the NRC by the Owners Group and was approved by the NRC via their SER dated January 12, 2005 and became WCAP-15793-P-A Revision 0 after incorporation of the NRC's SER.

3. Plant specific analysis in support of WCAP-15973-P-A was submitted to the NRC by FPL as part of the ASME Relief Request no. 26 for the Third 10-year ISI Program. It was approved by the NRC in their SER dated November 22, 2005.

4. Plant specific analysis in support of WCAP-15973-P-A was submitted to the NRC by FPL as part of the ASME Relief Request no. 6 for the Fourth 10-year ISI Program. It was approved by the NRC in their SER dated January 31, 2011.

These four documents show that the TLAA analyses are applicable for the period of extended operation. The referenced NRC SERs were issued as an ASME Code Relief for the duration 10 years of the third and fourth 10-year intervals. However, ASME Relief Request no.26 for the 3rd Interval, and in ASME Relief request no. 6 for the 4th interval provide the TLAA analysis in accordance with 10 CFR 54.21(c)(1)(ii).

In conclusion, the flaw growth analyses due to thermal or mechanical cycling and the potential wastage of the ferritic material that is adjacent to the half nozzle configuration exposed to reactor coolant of the Unit 1 RCS hot-leg Alloy 600 instrument nozzle repairs have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

18.3.9 REACTOR COOLANT PUMP FLYWHEEL CRACK GROWTH

EC296122

The reactor coolant pump (RCP) flywheels are discussed in Sections 5.5.5.3 of the St. Lucie Unit 1 UFSAR. During normal operation, the RCP flywheel possesses sufficient kinetic energy to potentially produce high-energy missiles in the unlikely event of failure. Conditions that 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.

The Unit 1 RCP flywheel crack growth calculation was determined to be a TLAA as discussed in Section 4.1.2 of NUREG-1779. In Section 5.5.5.3 of the Unit 1 UFSAR, the RCP flywheel crack growth calculation indicates that the number of starting cycles required to cause a reasonably small crack to grow to critical size is more than 100,000. For the 60-year Period of Extended Operation (PEO), the 100,000 RCP start cycles required to cause a crack to grow to critical size for Unit 1 is far greater than the number of start cycles for this time period.

In conclusion, Reactor Coolant Pump Flywheel Crack Growth analysis has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).