UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 30, 2021

Mr. Robert Coffey
Executive Vice President Nuclear
and Chief Nuclear Officer
Florida Power & Light Company
Mail Stop: NT3/JW
15430 Endeavor Drive
Jupiter, FL 33478

SUBJECT: ST. LUCIE PLANT, UNIT 2 – AUTHORIZATION OF RR#15 REGARDING EXTENSION OF ASME REQUIREMENTS RELATED TO REACTOR PRESSURE VESSEL WELD EXAMINATIONS FROM 10 TO 20 YEARS (EPID L-2020-LLR-0283)

Dear Mr. Coffey:

By letter dated October 30, 2020, Florida Power & Light Company (FPL, the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1, for Category B-A and B-D examinations for the St. Lucie Plant, Unit No. 2 (St. Lucie, Unit 2) reactor pressure vessel welds and nozzle welds.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative for the remainder of the Fourth 10-year inservice inspection (ISI) Interval and Fifth 10-year ISI interval at St. Lucie, Unit 2, for Category B-A and B-D examinations, on the basis that the alternative provides an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of Relief Request RR#15 at St. Lucie, Unit 2 for the remainder of the Fourth 10-Year ISI interval, which began on August 8, 2013, and is scheduled to end on August 7, 2023, and to the Fifth 10-Year ISI interval, which is scheduled to start on August 8, 2023, and end on August 7, 2033.

All other ASME BPV Code, Section XI, requirements for which an alternative was not specifically requested and authorized remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

All other ASME OM Code requirements for which an alternative was not specifically requested and authorized remain applicable.

If you have any questions, please contact Michael Mahoney at 301-415-3867 or via email at Michael.Mahoney@nrc.gov.

Sincerely,

David J. Wrona

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Date: 2021.09.30 09:33:52 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosure:
Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST RR#15 REGARDING EXTENSION OF ASME REQUIREMENTS RELATED

TO REACTOR PRESSURE VESSEL WELDS FROM 10 TO 20 YEARS

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated October 30, 2020 (Reference 1) Florida Power & Light Company (FPL, the licensee), requested relief from the requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1, for Category B-A and B-D examinations for the St. Lucie Plant, Unit No. 2 (St. Lucie 2) reactor pressure vessel welds and nozzle welds.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative for the remainder of the Fourth 10-year inservice inspection (ISI) Interval and the Fifth 10-year ISI interval at St. Lucie, Unit 2, for Category B-A and B-D examinations, on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Regulatory Requirements

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), which states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI.

Paragraph 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of 10 CFR 50.55a(b) through (h) may be used, when authorized by the Director, Office of Nuclear Reactor Regulation, if (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

10 CFR 50.61a, “Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

The licensee has submitted this request on the basis that compliance with the specified ASME Code requirements would provide an acceptable level of quality and safety.

Guidance

U.S. NRC Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” May 1988 (ADAMS Accession No. ML003740284).

U.S. NRC Regulatory Guide 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” January 2018 (ADAMS Accession No. ML17317A256).

U.S. NRC, NUREG-1874, “Recommended Limits for Pressurized Thermal Shock (PTS),” March 2010 (ADAMS Accession No. ML15222A848).

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected by the Proposed Alternative

The affected component is the St. Lucie Unit 2 reactor vessel (RV), specifically, the following ASME BPV Code, Section XI examination categories and item numbers covering examinations of the RV. These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV Code, Section XI.

Examination Category	Item Number	Description
B-A	B1.10	Shell Welds
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.20	Head Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inside Radius Section

Note: Category B-A welds are defined as “Pressure Retaining Welds in Reactor Vessel.” Category B-D welds are defined as “Full Penetration Welded Nozzles in Vessels.”

3.2 Applicable Code Edition, Addenda, and Requirements

The Fourth 10-Year ISI Interval for St. Lucie Unit 2 began on August 8, 2013 and is scheduled to end on August 7, 2023. The Fifth 10-Year ISI Interval for St. Lucie Unit 2 is scheduled to start on August 8, 2023, and end on August 7, 2033. The Code of Record for the fourth inspection interval is ASME Code, Section XI, 2007 Edition through 2008 Addenda.

The IWB-2411, Inspection Program, requires volumetric examination of essentially 100% of RV pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. The Fourth 10-year ISI interval for St. Lucie Unit 2 is scheduled to end on August 7, 2023. The applicable ASME Code for the Fifth 10-year ISI interval will be selected in accordance with the requirements of 10 CFR 50.55a.

3.3 Reason for Request

The licensee proposed an alternative from the requirement of the IWB-2411 Inspection Program, that volumetric examination of essentially 100% of RV pressure-retaining Examination Category B-A and B-D welds be performed once each 10-year interval. The licensee stated that extension of the interval between examinations of Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs.

3.4 Licensee's Proposed Alternative

The licensee proposed to not perform the ASME Code required volumetric examination of the St. Lucie 2 RPV full penetration pressure-retaining Examination Category B-A and B-D welds for the Fourth 10-Year ISI interval. Instead, the licensee will perform these volumetric examinations in the Fifth 10-Year ISI interval. The licensee stated that the proposed inspection date is a slight deviation from the latest revised implementation plan, since the implementation plan reflects the next inspection being performed in 2030 for St. Lucie, Unit 2. The licensee further stated that this proposed inspection schedule is considered to have a minor impact on the future inspection plan and distribution of inspections over time.

3.5 Licensee's Bases for Use

In accordance with 10 CFR 50.55a(z)(1), the licensee proposed an alternate inspection interval on the basis that the current interval can be revised with negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174. The licensee stated that the methodology used to conduct this analysis is based on that defined in the study WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval." The licensee stated they compared the results of the calculations for St. Lucie, Unit 2 to those obtained from the Combustion Engineering (CE) pilot plant evaluated in WCAP-16168-NP-A, Revision 3. The licensee stated that the parameters for St. Lucie, Unit 2 are bounded by the results of the CE pilot plant evaluation.

3.6 Duration of Proposed Alternative

The licensee requests NRC approval and authorization to defer performance of the required volumetric inspections from the Fourth 10-Year ISI interval to the Fifth 10-Year ISI interval for St. Lucie, Unit 2.

3.7 NRC Staff Evaluation

3.7.1 Licensee's Methodology and Flaw Evaluation Assessment Basis

The licensee's alternate ISI requirement in RR#15 is based on a risk-informed RPV fracture analysis that was performed in accordance with the NRC staff-approved, risk-informed flaw

analysis methods in WCAP-16168-NP-A, Revision 3 (Reference 3; henceforth WCAP-A for the objectives of this safety evaluation (SE)). The methodology in WCAP-A was developed by the Pressurized Water Reactor Owners Group (PWROG) to satisfy the 95th percentile total through-wall cracking frequency ($TWCF_{95\text{-}TOTAL}$) criteria for pressurized water reactors (PWRs) established in NRC NUREG-1874 (Reference 4) and the delta large early release frequency (ΔLERF) criteria that are specified in Regulatory Guide 1.174, Revision 3. The analysis in RR#15 includes the risk-informed flaw evaluation tables and component-specific and total through-wall cracking frequency results (i.e., $TWCF_{95\text{-}XX}$ and $TWCF_{95\text{-}TOTAL}$ parameter results) that were called for in the WCAP-A methodology.

Based on these criteria, the NRC may grant authorization of the risk-informed ISI request using the ΔLERF criteria specified in Regulatory Guide 1.174, Revision 3 if the licensee can demonstrate that: (1) the $TWCF_{95\text{-}TOTAL}$ value for the St. Lucie, Unit 2 RPV is less than a maximum, upper-bound $TWCF_{95\text{-}TOTAL}$ of 3.16×10^{-7} through-wall cracking events per reactor year (as specified in WCAP-A for Combustion Engineering (CE)-designed PWRs), and (2) the proposed ISI interval extension will not result in a ΔLERF in excess of 1.0×10^{-7} events per reactor year.

3.7.2 Analysis Parameters Used for These Types of Risk-Informed Alternatives

The NRC staff defines the risk-informed parameters and the type of data requested for these types of relief request submittals, along with the minimum PWR pilot-plant acceptance criteria defined for those parameters in Section 3.4 of the NRC staff's July 26, 2011 SE (Reference 5) for WCAP-A. For comparisons to St. Lucie, Unit 2, the acceptance criteria for the parameters of reference in WCAP-A are based on those that were established for the evaluation of the Palisades nuclear power plant design in WCAP-A. The NRC staff's evaluations of licensee's risk-informed parameter results are given in the SE subsections that follow.

3.7.3 Identification of Limiting Design Basis Transients

In the NRC staff's SE of July 26, 2011, for WCAP-A, the NRC staff requested confirmation that the dominant transients for PWR pressurized thermal shock (PTS) in NRC PTS Risk Study (i.e., NUREG-1874) are applicable as the dominant design basis transients for the licensee's proposed risk-based alternative. The NRC staff also requested information relative to the design of the RPV cladding layer.

Regarding the PTS transients, the licensee identified (in Table 1 of Attachment 1 in RR#15) that the transients are defined in NRC letter report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants" (Reference 6; henceforth referred to as the "NRC PTS Generalization Study") and that those transients serve as the limiting design basis transients for the RPV welds that were assessed in RR#15. The NRC staff verified that, for CE-type PWRs like St. Lucie, Unit 2, the PWROG's methodology in WCAP-A uses the PTS transients that were defined in NUREG-1874 and clarified in the PTS Generalization Study as the limiting PTS transients for the PWROG's risk-informed flaw analysis that was included in WCAP-A. Therefore, the NRC staff finds the licensee's transient basis to be acceptable based on the information in NUREG-1874 and the PTS Generalization Study, and the NRC staff's conclusions in the July 26, 2011, SE for WCAP-A that establish that the PTS transient characteristics for a given nuclear steam supply shutoff system (NSSS) design of U.S. PWR light water reactor facilities are generically applicable for all PWRs designed by the same reactor NSSS vendor (i.e., CE for the unit designs at St. Lucie, Unit 2 and at the Palisades nuclear power plant).

Regarding the cladding layers, the licensee reported that the cladding for the RPV at St. Lucie, Unit 2 was deposited using a "single layer," with the welding of the cladding being controlled using the methods defined in Regulatory Guide 1.43 (Reference 7). The NRC staff noted that the design of the RPV cladding layer at St. Lucie, Unit 2 is consistent with the design of the cladding layer in the Palisades RPV. Thus, for the proposed alternative, the NRC staff concludes that the licensee did not need to evaluate the impacts that multiple pass layers would have on the design of the RPV cladding at St. Lucie, Unit 2 because: (1) the cladding layer at St. Lucie, Unit 2 was deposited as a single layer austenitic stainless layer, and (2) the design of the cladding layer at St. Lucie, Unit 2 is consistent with and bounded by the NRC staff's assessment of the Palisades cladding layer in the NRC staff's July 26, 2011, for WCAP-A.

3.7.4 Frequency and Severity of Design Basis Transients

In the NRC staff's SE of July 26, 2011, the NRC staff established its position that licensees submitting risk-informed ISI extension alternatives for their RPVs should report whether the frequency of the limiting design basis transients during past plant operations are less than the frequency of the design basis transients identified in WCAP-A report as being significant contributors to fatigue. In Table 1 of Attachment 1 in the RR#15 submittal, the licensee stated that the limiting design transients for fatigue flaw growth of detected flaws are the reactor coolant system (RCS) heatup and cooldown transients for St. Lucie, Unit 2. The licensee also stated that the frequency of allowed heatups and cooldowns per reactor year at St. Lucie, Unit 2 is bounded by the 13 cycles/reactor year value that was set as an upper bound limit for CE-designed PWRs in WCAP-A.

The NRC staff confirmed that, in the NRC staff-approved WCAP-A methodology, the PWROG established 13 cycles/reactor year as the maximum bounding frequency of reactor heatup and cooldowns that could occur (per reactor year) for CE-designed PWRs. The NRC staff noted that, consistent with RPV weld and base metal assessments for Palisades in WCAP-A, the licensee's limiting design transients for fatigue flaw growth are the RCS heatup and cooldown transients for St. Lucie, Unit 2.

The NRC staff reviewed Chapter 3 of the St. Lucie, Unit 2 updated final safety analysis report (UFSAR) (Reference 16), to verify the validity of the licensee's heatup and cooldown frequency basis in RR#15. The NRC staff noted that UFSAR Table 3.9-2 establishes an upper bound limit of 500 cycles of RCS heatups and cooldowns over the design life of the plant. Thus, for a cumulative 60-year licensing term, this correlates to 8.33 RCS heatup and cooldown cycles per reactor year for St. Lucie, Unit 2, and demonstrates licensee's yearly frequency value is bounded by the maximum frequency of 13 cycles/reactor year that is established for RCS heatups and cooldowns of CE-designed PWRs in WCAP-A. Thus, the NRC staff finds that the licensee's fatigue flaw growth basis is acceptable because the licensee's fatigue flaw growth basis is sufficiently bounded by that analyzed and established for CE-design PWR units in the WCAP-A report.

3.7.5 Scope and Schedule for Inspecting the RPV Welds During the 20-Year ISI Interval

In the NRC staff's SE of July 26, 2011, the NRC staff stated that licensees submitting risk-informed ISI extensions for their RPVs should identify ISI schedule for RPV weld examinations that will be performed during the proposed 20-Year ISI Interval. The SE also established the NRC staff's position that the dates for the weld inspections must be within one refueling cycle of

the revised dates identified for inspection in the implementation plan in PWROG Letter No. OG-10-238 (Reference 8).

In Section 5 and Table 2 of Attachment 1 in the RR#15 submittal, the licensee states that the proposed alternative would defer the resulting ISI inspections of the applicable RPV weld components from year 2023 in the Fourth 10-Year ISI interval to years 2030 or 2032 in the Fifth 10-year ISI interval for the reactor unit. The licensee stated that the resulting impact to the implementation plan in OG-10-238 "would increase the number of inspections in 2032 (from three to four) and decrease the number of inspections in 2030 (from five to four)." The licensee also stated that, "based on Figure 3 and Figure 4 of OG-10-238, this proposed inspection schedule is considered to have a minor impact on the future inspection plan and the distribution of inspections over time."

The NRC staff verified that the licensee had included the inspection schedule for the RPV welds in RR#15 and that the proposed RPV weld inspections in 2030 and 2032 are within one cycle of the time for performing the ISI inspections in OG-10-238. However, the NRC staff determined that it would need further clarifications from the licensee relative to the alternate basis for performing the RPV weld examinations and the ISI examinations of the RPV nozzle inside radius section locations during the Fifth 10-Year ISI interval for the unit. Specifically, the NRC staff noted that the criteria in ASME Section XI Paragraph IWB-2411 and Table IWB-2411-1 require the licensee to perform an inspection of 100% of the population of accessible components covered by the applicable ASME Section XI inspection items once every 10-Year ISI interval. The NRC staff also noted that, based on the licensee's reference of WCAP-18275-NP, Revision 0 (as docketed as an attachment in Reference 9) in RR#15, there are many more RPV weld locations that would need to be inspected during the Fifth 10-Year ISI interval (i.e., in Years 2030 or 2032) than just a total of eight specific RPV pressure retaining weld locations being scheduled for inspection.

The NRC staff issued request for additional information (RAI) number (#)1 (Reference 10) to request clarification on whether the "number of inspections" terminology reference was being made in reference to a specific set of eight RPV weld or nozzle insider radius section locations or to the set of RPV components (100% population) for each RPV component type that is required to be inspected in accordance with the specific ASME Section XI Inspection Items referenced in RR#15. In RAI #2 (Reference 10), NRC staff also asked the licensee to identify the percent population of components that will be inspected during the 20-Year ISI interval proposed in RR#15 for each of the ASME Section XI, Table IWB-2500-1, Examination Category B-A and B-D inspection items referenced in RR#15.

The licensee responded to the RAIs in a letter dated April 1, 2021 (Reference 11). In its response to RAI #1, the licensee clarified that the term "number of inspections" in Section 5 of RR#15 represents the number of expected subsequent RV ISI examinations of the U.S. PWR fleet during a specific year based on the implementation plan in OG-10-238. In its response to RAI #2, the licensee clarified that, consistent with the requirements in ASME Section XI, 2007 Edition through 2008 Addenda, Table IWB-2500-1, Examination Categories B-A, "Pressure Retaining Welds in Reactor Vessel" and B-D, "Full Penetration Welded Nozzles in Vessels," FPL will examine 100% of the required RPV welds applicable to St. Lucie, Unit 2. Based on the licensee's responses to RAI #1 and RAI #2, the NRC staff finds the scope of the RPV weld inspections that will be performed in either 2030 or 2032 to be acceptable because, consistent with the referenced ASME Section XI examination categories and inspection items, the licensee will be inspecting 100% of the accessible RPV shell welds, head welds, and nozzles welds that

are within the scope of the referenced ASME inspection items during the alternate 20-Year ISI interval proposed for the component-specific locations.

Under this alternative ISI interval basis, the licensee will be required under 10 CFR 50.55a(z)(1) and the alternate relief request basis to perform and complete the required ISI inspections of the specified RPV welds and nozzle inner radius section locations by August 7, 2033 (which corresponds to the end of the Fifth 10-Year ISI interval for St. Lucie, Unit 2).

3.7.6 Relevant Operating Experience - Summary of ISI Inspection Results

In the NRC staff's SE of July 26, 2011, the NRC staff established its position that licensees submitting risk-informed ISI extensions for their RPVs should report the results of its prior ISI inspections of the applicable RPV weld locations.

The NRC staff noted that, in Table 2 of RR#15, the licensee identifies that it performed past volumetric ISI examinations of the RPV pressure retaining welds in years 1989, 2000, and 2012. FPL reported that there was a total of 60 RPV subsurface indications that were reported as being detected during Year 2012 inspections and that the flaws were located in either the upper-to-intermediate shell girth weld or in the longitudinal (or axial) seam welds located in the RPV upper, intermediate, or lower shell courses. FPL also stated that the 60 flaw indications were found to be acceptable without need for repair per the acceptance criterion in ASME Section XI, Table IWB-3510-1. Of these 60 indications, FPL reported that 5 of the indications were located within the inner 1/10th or inner 1 inch of the RPV wall thickness and required additional evaluation in accordance with the flaw evaluation requirements in the alternate PTS rule (i.e., 10 CFR 50.61a, Reference 12).

In RAI #3 (Reference 10), the NRC staff requested additional information on the five risk-assessed indications to confirm that any potential flaw growth occurring in the flaws associated with indications is bounded by the fatigue flaw growth assumptions and values used in the WCAP-A methodology.

In its response to RAI #3, the licensee clarified that the 5 flaws in Table 2 of RR#15 are a subset of the total 60 flaw indications that were identified as being acceptable in accordance with the flaw evaluation criteria in ASME Section XI, Table IWB-3510-1. The licensee also clarified that 2012 was the first year in which the past ISI inspections of the RPV shell, head and nozzle welds had resulted in recordable flaw indications in the referenced RPV circumferential or axial weld components. The licensee stated that, since 2012 is the first year in which the licensee had reported relevant flaw indications in the weld components, there is no site-specific flaw growth data for the flaws that were detected in the welds. Based on the licensee's response and explanations, the NRC staff finds that the crack growth rate (CGR) applied and used in the licensee's risk-informed TWCF calculations to be acceptable because the licensee is applying the upper-bound CGR cited in the WCAP-A report as the basis for the component-specific $TWCF_{95\text{-}XX}$ values cited in RR#15.

The NRC staff confirmed that FPL evaluated the five flaws in accordance with the flaw evaluation and flaw distribution criteria in WCAP-A, which are based on the alternate PTS rule. The NRC staff also verified that the flaw distribution assessments in RR#15 demonstrated that: (1) number of flaws in the specified flaw distribution ranges were bounded by the maximum number of flaws allowed for those flaw ranges in WCAP-A, and (2) no assessed flaw was projected to exceed the maximum allowable flaw size (i.e., 0.475 inch for assessed weld flaws and 0.375 inch for assessed plate flaws) specified and approved in WCAP-A. Based on this

review, the NRC staff finds that the licensee has appropriately addressed the ISI history in RR#15 because FPL has demonstrated that any flaws detected in the RPV welds will meet the flaw distribution criteria and allowable flaw sizes for detected flaws in WCAP-A, including conservative estimates to account for potential fatigue flaw growth of the flaws through the end of the Fifth 10-Year ISI interval for the unit.

3.7.7 Need for a Site-Specific Flaw Evaluation of RPV Forgings

The NRC staff verified that the need for performing a site-specific flaw assessment of any RPV nozzle forging is not necessary and not applicable to the risk-informed flaw evaluation in RR#15. Specifically, the NRC staff confirmed that, in UFSAR Section 5.2.3.3.2.1.1, the licensee indicates that it used the guidance in Regulatory Guide 1.43 to control the heat inputs for deposited RPV welds (including nozzle forging-to-vessel welds) during fabrication of the RPV and to provide assurance that underclad cracking would not occur in the RPV.

The NRC staff also verified that the RPV does not include any RPV nozzles that are projected to have neutron fluence exposures in excess of 1×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the period of extended operation such that otherwise, if the nozzle fluences were projected to be in excess of the 1×10^{17} n/cm^2, the nozzles would need to be within the scope of the licensee's 10 CFR 50.61 (Reference 13) PTS assessment for St. Lucie, Unit 2. Therefore, based on this review, the NRC staff finds that licensee did not need to include a site-specific or component-specific through-wall cracking frequency ($TWCF_{MAX-FO}$) assessment of the RPV nozzle-to-forging welds in Table 3 of RR#15 because: (1) the licensee's welding practices were done is a manner that would preclude underclad cracking from occurring in the cladding adjacent to the nozzle forgings, (2) the licensee's projected 60-year fluences for the RPV nozzles provide confirmation that FPL did not need to include any RT_{PTS} calculations for the RPV nozzle forgings within the scope of licensee's 60-year PTS assessment for St. Lucie, Unit 2, and (3) the bases summarized in arguments (1) and (2) of this sentence support the NRC staff's finding that the licensee did not need to perform RT_{MAX-FO} and $TWCF_{95-FO}$ calculations for the RPV nozzle forgings as part of the RR#15 risked-informed TWCF calculations.

3.7.8 Submittal of Information Required by Section (e) in 10 CFR 50.61a

The NRC staff verified that the licensee does not need to include the information required by Section (e) of 10 CFR 50.61a (Reference 12) in RR#15 because the information would only need to be included in RR#15 if: (1) the licensee has received Commission approval of a subsequent license renewal application (SLRA) for the unit in accordance with the Commission's requirements in 10 CFR 54.29, (2) the Office Director of the Commission's Office Nuclear Reactor Regulation has issued a subsequent renewed operating license for the unit, and (3) the licensee's alternative in RR#15 was requesting NRC staff's authorization for ISI extension into either the Seventh or Eighth 10-Year ISI interval for the unit (i.e., into either of the 10-Year intervals that are applicable to a subsequent period of extended operation for the unit). Since the scope of RR#15 only requests NRC staff authorization to extend the applicable RPV inspections to the end of the Fifth 10-Year ISI interval, the NRC staff concludes that the licensee does not need to include the information required by 10 CFR 50.61a(e).

3.7.9 Through-wall Cracking Frequency Assessment, Including Calculation of ΔT_{30}, RT_{MAX-X}, $TWCF_{95-XX}$ and $TWC_{F95-Total}$ Values

In the NRC staff's SE of July 26, 2011, the NRC staff established its position that the maximum adjusted reference temperatures and 30 ft-lb shifts in adjusted referenced temperature values

(i.e., RT_{MAX-X} and ΔT_{30} values, as defined in 10 CFR 50.61a) may be calculated using the methods documented in the latest version of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," or in an alternate NRC-approved methodology using these types of parameters. The NRC staff also stated the submittals should include the material property and fluence information related to these parameters that were cited in the PWROG's response to RAI #3(a), as given and referenced in Appendix N of WCAP-A.

The NRC staff noted that the licensee included the material property and neutron fluence data, ΔT_{30}, values, and RT_{MAX-XX} values for the RPV base metal and weld components at 55 effective full power years (55 EFPY) in Table 3 of RR#15. The licensee stated that the data and RT-related values are based on information in the WCAP-18275-NP, Revision 0 report (Reference 13), which was included as an enclosure in the licensee's license amendment request of February 18, 2020 (Reference 9) and approved in License Amendment No. 206 (Reference 15).

The NRC staff noted that the licensee did not include any weight-% manganese (Mn) or weight-% phosphorous (P) for the RPV base metal and welds components that were assessed in RR#15. However, the NRC staff found the omission of the Mn and P alloying data to be acceptable based on NRC staff's confirmation that: (1) the licensee still relies on the methods of 10 CFR 50.61 for the licensee's 60-year PTS assessment in the current licensing basis (i.e., for 55 EFPY), and (2) the methods in 10 CFR 50.61 do not rely in Mn and P alloying percentages as inputs for the RT_{PTS} values that are required to be calculated in accordance with 10 CFR 50.61 rules, or as a general basis for calculating the RT_{MAX-XX} values that needed to be included and were included as inputs for the licensee's RR#15 analysis.[1]

The NRC staff noted that FPL reported the following Limiting $TWCF_{95-XX}$ values (XX = AW for axial welds, PL for plate materials, CW for circumferential welds) in the Table 3 of RR#15 to demonstrate that the $TWCF_{95-TOTAL}$ value for the RPV is less than the limiting $TWCF_{95-TOTAL}$ frequency of 3.16×10^{-7} through-wall cracking events per reactor year approved in WCAP-A:

- For the limiting RPV axial weld, a $TWCF_{95-AW}$ value of 1.09×10^{-12} through-wall cracking events per reactor year
- For the limiting RPV plate, a $TWCF_{95-PL}$ value of 2.74×10^{-10} through-wall cracking events per reactor year
- For the limiting RPV circumferential weld, a $TWCF_{95-cW}$ value of 7.54×10^{-17} through-wall cracking events per reactor year

Based on these $TWCF_{95-XX}$ values, the licensee calculated and reported a $TWCF_{95-TOTAL}$ value of 6.75×10^{-10} through-wall cracking events per reactor year for the RPV through 55 EPFY (which corresponds to the end of a cumulative 60-year licensing term for the unit).

The NRC staff performed independent $TWCF_{95-AW}$, $TWCF_{95-CW}$, and $TWCF_{95-PL}$ calculations for the limiting RPV axial weld, circumferential weld, and plate components and an independent calculation of $TWCF_{95-TOTAL}$ value for the St. Lucie, Unit 2 RPV through 55 EPFY. The NRC staff verified that the licensee did not need to perform or include a $TWCF_{95-FO}$ calculation for the RPV nozzle forgings in RR#15 based on the licensee's demonstration and NRC staff's confirmation that the inside surface neutron fluence exposures for the RPV nozzle forgings are all projected to be less than 1×10^{17} n/cm^2 (E > 1.0 MeV) at 55 EFPY. The NRC staff verified that FPL's

[1] The only differences being that, if the methods in 10 CFR 50.61 are used and applied as a basis for calculating RT_{MAX-XX} values: (1) the calculations of RT_{MAX-XX} would not need to include a Margin Term assessment in the RT calculations, and (2) the values of RT_{MAX-XX} are reported in ºR instead of ºF.

$TWCF_{95\text{-}AW}$, $TWCF_{95\text{-}AW}$, and $TWCF_{95\text{-}PL}$ values for the limiting RPV axial weld, circumferential weld, and plate components were at least as conservative as those calculated by the NRC staff for the same limiting RPV components. The NRC staff also verified that FPL's limiting $TWCF_{95\text{-}TOTAL\ was}$ for the RPV that was at least as conservative as that calculated by the NRC staff for the RPV and that the $TWCF_{95\text{-}TOTAL}$ for the RPV is well within by the maximum (upper-bound) $TWCF_{95\text{-}TOTAL}$ value of $3.16x10^{-7}$ through-wall cracking events per reactor year that is established for CE-designed PWRs in WCAP-A.

The NRC staff also noted that the methodology in WCAP-A conservatively sets the $TWCF_{95\text{-}TOTAL}$ equal to the ΔLERF value for the RPV that may result from initiation of the postulated, limiting PTS event at the plant. Thus, based on the NRC staff's independent calculations and verifications, the NRC staff verified that the licensee sets the ΔLERF value to a value of $6.75x10^{-10}$ early release events per reactor year, and that this value meets the upper limit of $1x10^{-7}$ early release events per reactor year that is established for ΔLERF values in RG 1.174, Revision 3.

3.7.10 NRC staff Evaluation Summary

Based on the above assessment, the NRC staff finds that:

(1) the licensee's $TWCF_{95\text{-}XX}$ values for the limiting RPV axial weld, circumferential weld, and plate components were calculated in accordance with the approved methodology in WCAP-A and are acceptable for implementation;

(2) the licensee's $TWCF_{95\text{-}TOTAL}$ value of $6.75x10^{-10}$ events per reactor year for the St. Lucie, Unit 2 RPV was calculated in accordance with the methodology in WCAP-A and meets the acceptance criterion of $3.16x10^{-7}$ events per reactor year set for $TWCF_{95\text{-}TOTAL}$ values of CE-designed PWRs in WCAP-A;

(3) the licensee's ΔLERF value of $6.75x10^{-10}$ events per reactor year meets and is bounded by the ΔLERF value of $1x10^{-7}$ per reactor year specified in RG 1.174, Revision 3; and

(4) the magnitude of the licensee's ΔLERF value may be used as the technical basis for authorizing the licensee's proposed 20-year ISI interval alternative using the risk-informed regulatory position established in RG 1.174, Revision 3.

Based on these criteria, the NRC staff finds the licensee has demonstrated that its proposed risk-informed alternative provides an acceptable level of quality and safety in lieu of complying with the ASME Section XI requirements and inspection items specified and referenced in RR#15.

4.0 CONCLUSION

The NRC staff has determined that the proposed alternative in the licensee's request referenced above would provide an acceptable level of quality and safety.

The NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

The NRC staff authorizes the use of proposed alternative RR#15 at St. Lucie, Unit 2 for the remainder of the Fourth 10-Year ISI interval and for the duration of the Fifth 10-Year ISI interval of the unit.

All other ASME BPV Code, Section XI, requirements for which an alternative was not specifically requested and authorized remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

All other ASME OM Code requirements for which an alternative was not specifically requested and authorized remain applicable.

5.0 REFERENCES

1. FPL Letter No. L-2020-160, "Extension of Unit 2 RPV Welds from 10 to 20 Years," St. Lucie Unit 2, Docket No. 50-389, St. Lucie Operating License No. NPF-16, October 30, 2020 (ADAMS Accession No. ML20304A148).

2. U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (ADAMS Accession No. ML17317A256).

3. Westinghouse Electric Company Class 3 Non-Proprietary Report No. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," October 2011 (ADAMS Accession No. ML11306A084).

4. USNRC Report No. NUREG-1874, "Recommended Limits for Pressurized Thermal Shock (PTS)," March 2010 (ADAMS Accession No. ML15222A848).

5. USNRC Letter and Safety Evaluation from Mr. Robert A. Nelson to Mr. W. Anthony Nowinowski, Program Manager - PWR Owners Group, "Revised Final Safety Evaluation by the Office of Nuclear Reactor Regulation Regarding Pressurized Water Reactor Owners Group Topical Report WCAP-16168-NP-A, Revision 2, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval,'" July 26, 2011 (ADAMS Accession Nos. ML111600295 and ML111600303).

6. USNRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004 (ADAMS Accession No. ML042880482).

7. USNRC Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," May 1973 (ADAMS Accession No. ML003740095).

8. PWR Owners Group Letter No. OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval,'" July 12, 2010 (ADAMS Accession No. ML11153A033).

9. FPL Letter No. L-2020-020, "License Amendment Request for RCS Pressure/Temperature Limits and LTOP Applicable for 55 Effective Full Power Years," St. Lucie. Unit 2,

Docket No. 50-389, Operating License No. NPF- 16, February 18, 2020 (ADAMS Accession No. ML20049A388).

10. NRR Email Capture, Relief Request RR#15 - Extension of St. Lucie Unit 2 RPV Welds from 10 to 20 Years - Request for Additional Information (L-2020-LLR-0157), March 4, 2021 (ADAMS Accession No. ML21063A319).

11. FPL Letter No. L-2021-065, "Response to Request for Additional Information, Relief Request Number RR#15, Extension of St. Lucie Unit 2 RPV Welds from 10 to 20 Years," April 1, 2021 (ADAMS Accession No. ML21091A060).

12. Section 50.61a, Title 10, *Code of Federal Regulations* (10 CFR 50.61a) "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

13. Section 50.61, Title 10, *Code of Federal Regulations* (10 CFR 50.61a) "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

14. USNRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988 (ADAMS Accession No. ML003740284).

15. USNRC License Amendment 206 and Safety Evaluation for St. Lucie Plant, Unit 2 issued to Mr. Don Moul, Executive Vice President and Chief Nuclear Officer, Florida Power and Light Company, "Issuance of Amendment No. 206 to Replace the Current Time-Limited Reactor Coolant System Pressure/Temperature Limit Curves and LTOP Setpoints with Curves and Setpoints That Will Remain Effective for 55 Effective Full Power Years (EPID L-2020-LLA-0029," February 26, 2021 (ADAMS Accession No. ML21022A219).

16. St. Lucie, Unit 2, Amendment 26 to Updated Final Safety Analysis Report and License Renewal 10 CFR 54.37(b) Report (ADAMS Package Accession No. ML20268A114).

Principal Contributor: J. Medoff, NRR

Date: September 30, 2021

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 – AUTHORIZATION OF RR#15 REGARDING EXTENSION OF ASME REQUIREMENTS RELATED TO REACTOR PRESSURE VESSEL WELD EXAMINATIONS FROM 10 TO 20 YEARS (EPID L-2020-LLR-0283) DATED SEPTEMBER 30, 2021

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