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

February 26, 2021

Mr. Don Moul
Executive Vice President, Nuclear Division
and Chief Nuclear Officer
Florida Power & Light Company
Mail Stop: EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 - 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)

Dear Mr. Moul:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 206 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2, in response to your application dated February 18, 2020.

The amendment modifies the St. Lucie Plant, Unit No. 2, Technical Specifications to update the reactor coolant system pressure and temperature limits and revise the low-temperature overpressure protection system settings.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Natreon J. Jordan, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 206 to NPF-16
2. Safety Evaluation

cc: Listserv

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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 206
Renewed License No. NPF-16

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company dated February 18, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 206, Renewed Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206, are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 35 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine S. Shoop

Digitally signed by Undine S. Shoop
Date: 2021.02.26 10:31:39 -05'00'

Undine S. Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: February 26, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 206

ST. LUCIE PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace page 3 of Renewed Facility Operating License No. NPF-16 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove	Insert
Index XXI	Index XXI
Index XXII	Index XXII
3/4 4-31a	3/4 4-31a
3/4 4 -31b	3/4 4-31b
3/4 4-37a	3/4 4-37a

neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.

D. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission's regulations: 10 CFR Part 20, Section 30.34 of 10 FR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 3020 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206, are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

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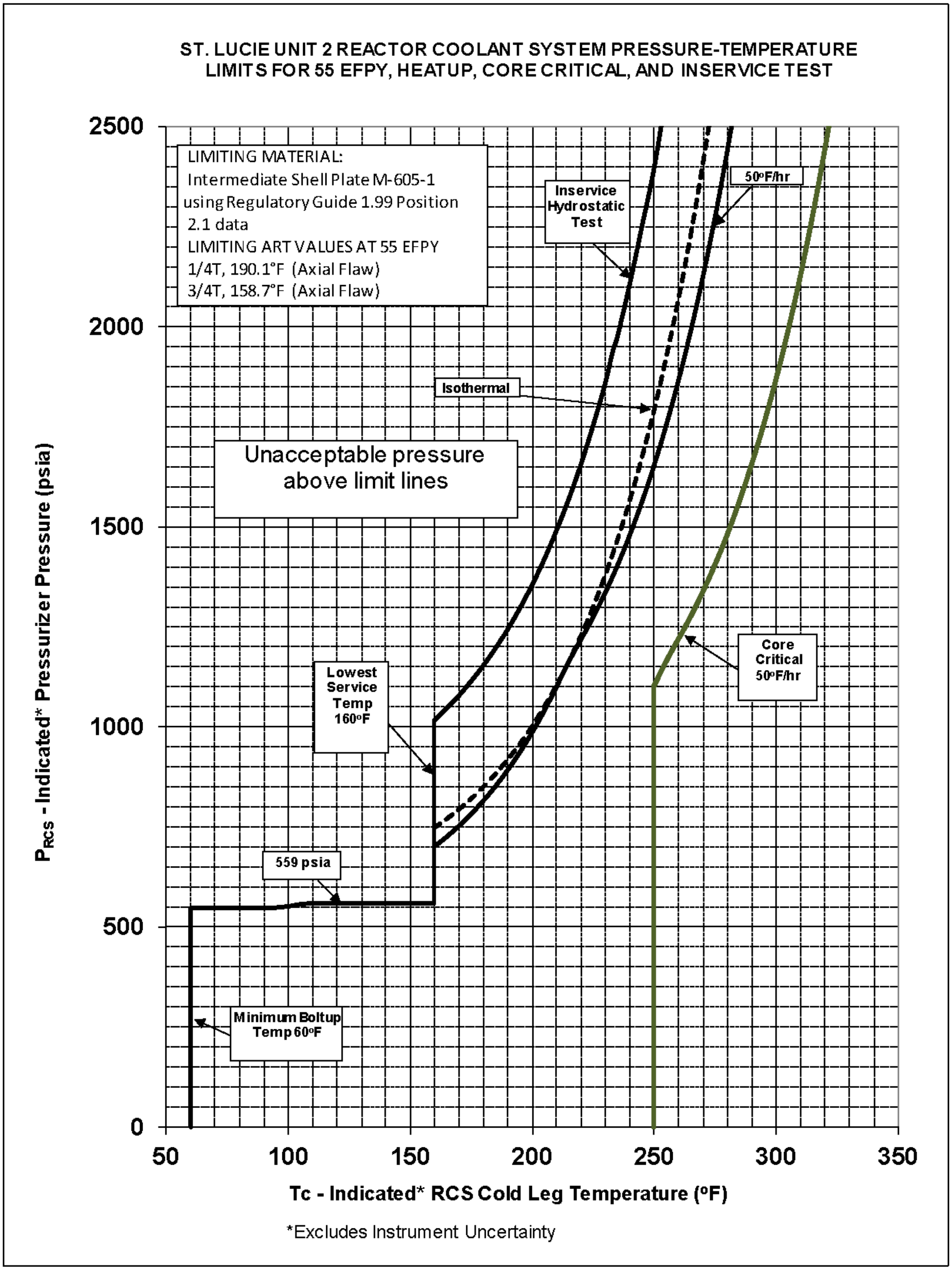
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, HEATUP, CORE CRITICAL, AND INSERVICE TEST
LIMITING MATERIAL:
Intermediate Shell Plate M-605-1
using Regulatory Guide 1.99 Position 2.1 data
LIMITING ART VALUES AT 55 EFPY
1/4T, 190.1°F (Axial Flaw)
3/4T, 158.7°F (Axial Flaw)
Inservice Hydrostatic Test
50°F/hr
Isothermal
Unacceptable pressure above limit lines
Core Critical 50°F/hr
Lowest Service Temp 160°F
559 psia
Minimum Boltup Temp 60°F
2500
2000
1500
1000
500
0
50
100
150
200
250
300
350
P_{RCS} - Indicated* Pressurizer Pressure (psia)
Tc - Indicated* RCS Cold Leg Temperature (°F)
*Excludes Instrument Uncertainty

FIGURE 3.4-3

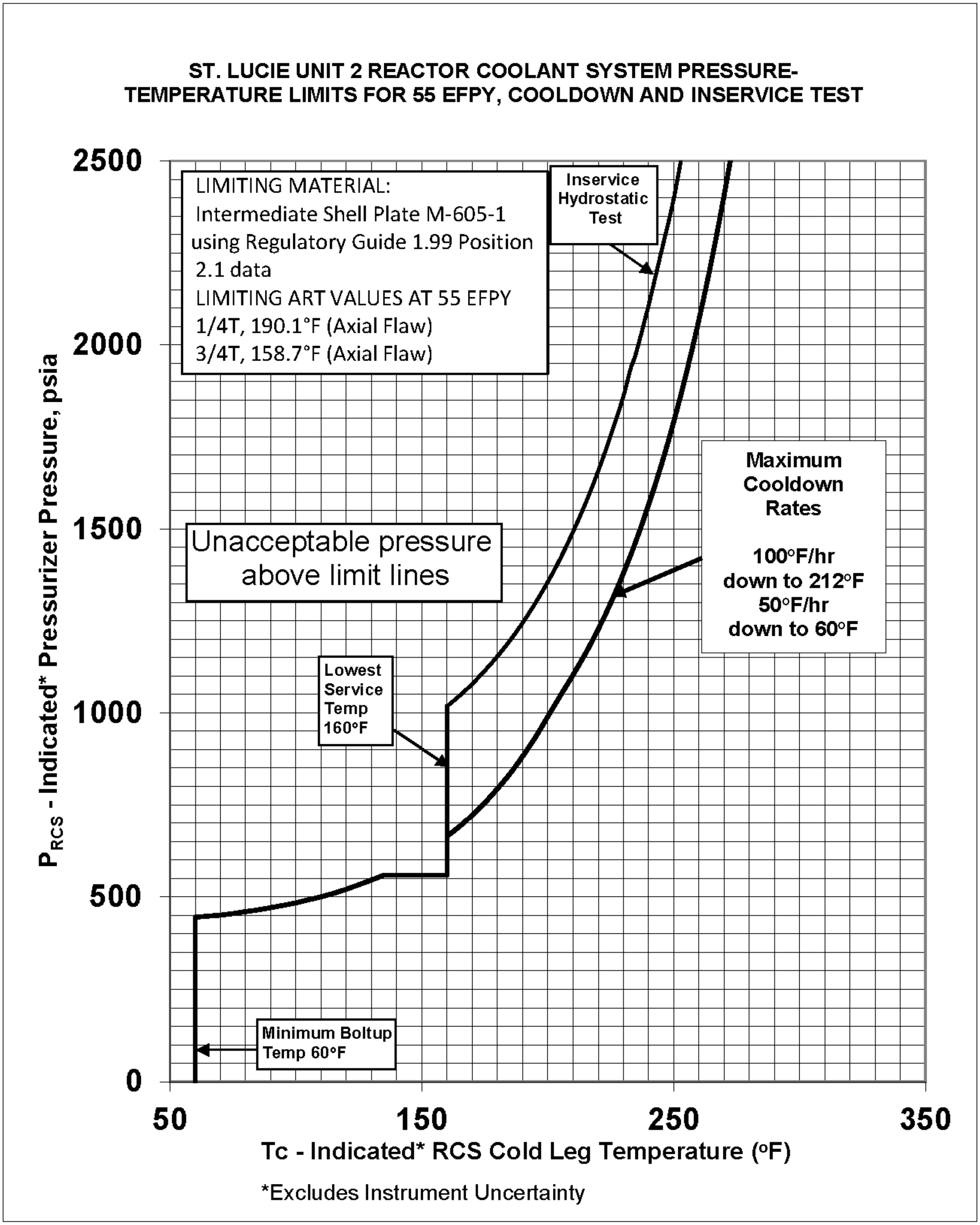
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-
TEMPERATURE LIMITS FOR 55 EFPY, COOLDOWN AND INSERVICE TEST
LIMITING MATERIAL:
Intermediate Shell Plate M-605-1
using Regulatory Guide 1.99 Position
2.1 data
LIMITING ART VALUES AT 55 EFPY
1/4T, 190.1°F (Axial Flaw)
3/4T, 158.7°F (Axial Flaw)
Inservice
Hydrostatic
Test
Maximum
Cooldown
Rates
100°F/hr
down to 212°F
50°F/hr
down to 60°F
Unacceptable pressure
above limit lines
Lowest
Service
Temp
160°F
Minimum Boltup
Temp 60°F
2500
2000
1500
1000
500
0
50
150
250
350
P_RCS - Indicated* Pressurizer Pressure, psia
Tc - Indicated* RCS Cold Leg Temperature (°F)
*Excludes Instrument Uncertainty

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, °F During Heatup	During Cooldown
≤ 55	≤ 252	≤ 240

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	Cold Leg Temperature, °F During Heatup	During Cooldown
≤ 55	60	149

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206

TO RENEWED OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By application dated February 18, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20049A388), Florida Power & Light Company (FPL, the licensee) submitted a license amendment request (LAR) to revise Technical Specifications (TS) 3/4.4.9, “Reactor Coolant System [RCS] Pressure/Temperature [P/T] Limits” for the St. Lucie Nuclear Plant, Unit 2 facility (St. Lucie, Unit 2). Specifically, the licensee proposed to replace the reactor coolant system (RCS) Pressure/Temperature (P/T) limit curves and low-temperature overpressure protection (LTOP) setpoints with curves and setpoints that will remain effective for 55 effective full power years (EFPY).

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization.” to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The NRC staff evaluates the acceptability of a facility’s proposed P/T limits based on the following NRC regulations and guidance:

The regulation in 10 CFR 50.36, “Technical Specifications,” requires that TSs include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

The regulation in 10 CFR 50.36(c)(2), “Limiting conditions for operation,” states that “[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.”

10 CFR 50.60, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” imposes fracture toughness and material embrittlement surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50.

Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50, requires, in part, that facility P/T limits for the reactor pressure vessel (RPV) be at least as conservative as those obtained by following the methods of analysis and the margins of safety in Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” to Section XI of the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50, establishes requirements for a facility’s surveillance program for monitoring RPV embrittlement due to neutron irradiation.

General Design Criterion (GDC) 14, “Reactor coolant pressure boundary,” and GDC 31, “Fracture prevention of reactor coolant pressure boundary,” of Appendix A to 10 CFR Part 50, in part, establish minimum requirements for the principal design criteria with respect to the reactor coolant pressure boundary.

GDC 15, “Reactor coolant system design,” requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

NUREG-0800, "Standard Review Plan for the “Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 5.3.2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock” (ADAMS Accession No. ML070380185) describes acceptance criteria for determining the P/T limits for ferritic materials in the beltline of the RPV based on Appendix G to Section XI, of the ASME Code methodology. Section 5.2.2, “Overpressure Protection” of the SRP, dated March 2007 (ADAMS Accession No. ML070540076), discusses the application of safety and relief valves and the reactor protection system ensures overpressure protection for the reactor coolant pressure boundary (RCPB) during operation at power.

Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains guidance for RPV embrittlement integrity evaluations.

Regulatory Guide 1.190, Revision 0 “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence with respect to, in part, GDCs 14 and 31 in Appendix A to 10 CFR Part 50.

Regulatory Issue Summary (RIS) 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” October 14, 2014 (ADAMS Accession No. ML14149A165) clarifies that P/T limits for ferritic RPV components, such as RPV inlet and outlet nozzles, could be more limiting because higher stress levels from structural discontinuities could result in a lower allowable pressure.
RIS 2014-11 also clarifies that the RPV beltline definition in Appendix G to 10 CFR Part 50 is applicable to all RPV ferritic materials with projected neutron fluence values greater than

1 x 10^{17} neutrons per square centimeter (n/cm^{2}) (E > 1 MeV), and that this fluence threshold remains applicable for the design life as well as throughout the licensed operating period of the reactor.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

The P/T curves currently in the St. Lucie, Unit 2 TS, also called the heatup and cooldown curves, were originally submitted for applicability to 55 EFPY in 2008 but were subsequently reduced to 47 EFPY as a result of the Extended Power Uprate (EPU). The period of applicability for the current P/T curves was further reduced to 31.98 EFPY as a result of the incorporation of the most recent St. Lucie, Unit 2 surveillance capsule test results and are administratively controlled, in the St. Lucie Corrective Action Program, to 31.98 EFPY. Surveillance capsule testing is part of the St. Lucie Unit 2 RPV material surveillance program required by Appendix H to 10 CFR Part 50 to monitor RPV embrittlement due to neutron irradiation. As such, the TS requires revision prior to reaching the 31.98 EFPY, which is expected to occur on April 5, 2021.

This LAR will replace the current time-limited St. Lucie, Unit 2 TS P/T limit curves with new P/T limit curves applicable to 55 EFPY. The LTOP requirements, which are based on the P/T limits, will also be applicable to 55 EFPY. As part of this LAR, the licensee also provided a discussion of the LTOP analysis to support the proposed P/T limits.

3.2 Staff's Evaluation

The NRC staff safety evaluation (SE) focuses on the information presented in the LAR. Staff reviewed information related to RPV embrittlement and development of the 55 EPFY P/T limits. Specifically, these are Sections 3 through 7 of WCAP-18275-NP, "St. Lucie, Unit 2 Heatup and Cooldown Limit Curves for Normal Operation through End of License Extension," Revision 0. This is found in Attachment 2 of the LAR, and also shown in Attachments 4 and 5 of the LAR as replacement P/T limits that are to be incorporated into TS 3/4.4.9 for St. Lucie, Unit 2. Since changes to P/T limits could affect the LTOP system settings, this SE also covers the neutron fluence projections in WCAP-18275-NP and the LTOP evaluation found in Attachment 3 of the LAR.

3.2.1 Evaluation of P/T Limits

The licensee developed P/T limits for the St. Lucie, Unit 2 applicable to 55 EFPY and proposed to replace the P/T limits currently in TS 3/4.4.9 Figures 3.4-2 and 3.4-3 with these 55 EFPY P/T limits, as shown in Attachments 4 and 5 to the LAR application. Details of the development of the 55 EFPY P/T limits are shown in Sections 3 through 7 of WCAP-18275-NP. The licensee stated that the proposed 55 EFPY P/T limits are based on the K_{IC} (plane strain fracture toughness) methodology and are consistent with Appendix G to Section XI of the ASME Code and the NRC-approved methodology in Topical Report WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves," Revision 4 (ADAMS Accession No. ML050120209). The licensee also recalculated LTOP system parameters and discussed these calculations in Appendix D of WCAP-18275-NP and Attachment 3 to the LAR application.

The NRC staff reviewed background information regarding previous submittals for the St. Lucie, Unit 2 RCS P/T limits described in Section 1 of Attachment 1 to the LAR application. The staff noted that the purpose of the current LAR application is to revise the P/T limits and LTOP applicable to 55 EFPY in St. Lucie, Unit 2 TS 3/4.4.9 prior to reaching 31.98 EFPY, as summarized in Section 3.1 of this SE. The staff evaluated the licensee's adjusted reference temperature (ART) projections to 55 EFPY and the development of the P/T limits based on those projections using the guidance in Section 5.3.2 of the SRP. The staff also evaluated the licensee's consideration of LTOP as it relates to P/T limits. These evaluations are discussed below.

The NRC staff noted that, in Section 2 of WCAP-18275-NP, the fluence projections were computed based on the current rated power of 3020 Megawatt thermal (MWt). The staff approved an EPU to 3020 MWt for St. Lucie, Unit 2 in 2012 (ADAMS Accession No. ML12268A167). Thus, the staff determined that the RPV embrittlement projections to 55 EPFY in WCAP-18275-NP included the effects of the 2012 EPU to 3020 MWt.

The NRC staff noted that the licensee included all RPV beltline materials, including extended beltline materials, as shown in Table 3-1 of WCAP-18275-NP, from information found in: the St. Lucie, Unit 2 surveillance capsule report for Capsule 97°, WCAP-17939-NP, Revision 0 (ADAMS Accession Nos. ML15154B077, ML15154B079, and ML15154B080); the previous 55 EFPY P/T limits LAR (ADAMS Accession No. ML080290135); certified material test reports for St. Lucie, Unit 2; and documented engineering analysis for Combustion Engineering (CE)-fabricated RPVs. The staff finds these sources to be reliable sources of RPV material data for St. Lucie, Unit 2 because the sources include at least one of the following: a document that is part of the implementation of a requirement (e.g., surveillance capsule report), a current licensing basis document, a material-specific certified report, or an analysis of record.

The staff verified that the licensee incorporated surveillance capsule data from Capsules 83°, 263°, and 97°, which are the capsules withdrawn and tested to date per the surveillance capsule schedule in Table 5.3-9 of the St. Lucie, Unit 2 Updated Final Safety Analysis Report (UFSAR). The staff noted that the licensee used Position 2.1 of RG 1.99, Revision 2, and verified the calculation of the chemistry factors (CF) in Table 4-1 of WCAP-18275-NP. As stated in Section 2.0 of this SE, RG 1.99, Revision 2 contains guidance for evaluating RPV embrittlement integrity. The credibility of surveillance data should be demonstrated before using Position 2.1 of RG 1.99, Revision 2. In Appendix B of WCAP-18275-NP, the licensee presented the credibility evaluation of the available surveillance data which is based on the five credibility criteria in RG 1.99, Revision 2. The licensee determined that the surveillance data is credible. The NRC staff reviewed the licensee's surveillance data credibility evaluation and finds it acceptable because the surveillance data meets the five criteria in RG 1.99, Revision 2.

The staff noted that the capsule fluence values in Table 4-1 are slightly less than the fluence values reported in the St. Lucie, Unit 2 Surveillance Capsule 97° report. However, the capsule fluence values in Table 4-1 ultimately resulted in slightly larger CF values (per the summing of squares procedure for fitting capsule data in RG 1.99, Revision 2), which led to slightly larger ΔRT_{NDT} values. Accordingly, the NRC staff finds the CF values in Table 4-1 of WCAP-18275-NP to be acceptable because larger ΔRT_{NDT} values lead to more conservative P/T limit curves. The staff also noted that no CF adjustment to the capsule ΔRT_{NDT} values in Table 4-1 is needed since the material with the limiting ART values in Tables 6-2 and 6-3 of WCAP-18275-NP is the same (i.e., the material has the same copper and nickel values) as the surveillance plate material, as noted in Table 3-1 of WCAP-18275-NP. In addition, no temperature adjustment to the capsule ΔRT_{NDT} values in Table 4-1 is necessary because the

capsules were irradiated in the St. Lucie, Unit 2 reactor. For RPV beltline materials that had no surveillance data available, the licensee used Position 1.1 of RG 1.99, Revision 2 to calculate the CF value. The NRC staff verified several of the CF values that were calculated by the licensee by using Position 1.1 of RG 1.99, Revision 2 (shown in Table 4-2 of WCAP-18275-NP) and finds them to be acceptable.

The staff verified the 55 EFPY ART calculations in Tables 6-2 and 6-3 of WCAP-18275-NP and confirmed that the limiting RPV beltline material is the Intermediate Shell Plate M-605-1 (Heat Number A-8490-2), with ART values of 190.1°F and 158.7°F at the quarter thickness (1/4T) and three-quarter thickness (3/4T), respectively. The staff noted that the licensee conservatively added 10°F to the CF value for this limiting material in calculating the ART values. Thus, the staff finds the limiting 55 EFPY ART values of 190.1°F and 158.7°F to be acceptable because the licensee calculated the effect of embrittlement on the ART values using the appropriate considerations and guidance, as discussed above.

The NRC staff evaluated the licensee's proposed 55 EFPY RCS P/T limits for heatup, core critical, and inservice test shown in replacement Figure 3.4-2 in Attachment 4 of the LAR application and the proposed 55 EFPY RCS P/T limits for cooldown and inservice test in replacement Figure 3.4-3 in the same attachment. Using the limiting ART values, the staff independently calculated P/T limits based on the related equation in Appendix G to Section XI of the ASME Code. However, this was done with the thermal stress intensity factor computed from a one-dimensional thermal stress analysis across a vessel wall. The staff compared its calculated values with those in the 55 EFPY P/T limits data in Tables 7-1 through 7-3 of WCAP-18275-NP and determined that the licensee's proposed 55 EFPY P/T limits are consistent with the methodologies in Appendix G to Section XI of the ASME Code and WCAP-14040-A, Revision 4. The staff verified that the cooldown P/T limits in Figure 3.4-3 are those corresponding to the cooldown rates shown in the figure: a rate of 100°F per hour down to a temperature of 212°F and a rate of 50°F per hour down to a temperature of 60°F. In addition, based on the flange and balance of reactor coolant system RT_{NDT} values in Table 3-2 of WCAP-18275-NP, the staff verified that flange requirements of 10 CFR Part 50, Appendix G, Paragraph IV.A.2 are bounded by the lowest service temperature (LST) requirements of NB-3211 and NB-2332 of Section III of the ASME Code.

The staff noted that the y-axis of the proposed 55 EFPY RCS P/T limits in Figures 3.4-2 and 3.4-3 in Attachment 4 of the LAR application is the indicated pressure at the pressurizer. Since there is a pressure head difference between the pressure at the pressurizer and the pressure at the RPV beltline, the staff verified that the licensee applied the actual pressure correction factor (APCF) of 77.1 psi to the proposed 55 EFPY RCS P/T limits to account for the static and dynamic head difference between the pressurizer instrument location and RPV beltline. The licensee applied the APCF by subtracting 77.1 psi from the calculated allowable pressure values in Tables 7-1 through 7-3 of WCAP-18275-NP.

The licensee stated that the proposed 55 EFPY RCS P/T limits shown in replacement Figures 3.4-2 and 3.4-3 in Attachment 4 of the LAR application do not include instrument uncertainty. In Section 4.0 of Attachment 1 of the LAR application, the licensee stated that LTOP system enable temperatures and pressure setpoints included instrument uncertainties in the transient analysis performed to determine those setpoints. The NRC staff finds this acceptable since it means that instrument uncertainties would be accounted for when the LTOP system is enabled (i.e., when RCS temperatures are less than or equal to the LTOP enable temperatures). At RCS temperatures greater than the LTOP enable temperatures, the licensee stated that RCS operation below and to the right of the proposed 55 EFPY P/T limits are

administratively controlled with margins greater than the instrument uncertainty. Accordingly, the staff finds the licensee's treatment of instrument uncertainty acceptable. Based on the evaluation above, the staff finds the licensee's proposed 55 EFPY RCS P/T limits for heatup, cooldown, core critical, and inservice test of St. Lucie, Unit 2 acceptable.

The NRC staff noted that the maximum fluence at 55 EFPY of the St. Lucie, Unit 2 RPV nozzles (hot leg nozzle [i.e., RPV outlet nozzle]) is 0.738 x 10^{17} n/cm^2, which is limiting per Table 6-4 of WCAP-18275-NP. The nozzles are not beltline material but are clarified by the definition of the RPV beltline region in RIS 2014-11, as being those RPV ferritic materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2. Since the maximum fluence at 55 EPFY of the St. Lucie, Unit 2 RPV nozzles is 0.738 x 10^{17} n/cm^2, the staff determined that the nozzles are not beltline materials per the clarification of the definition of beltline materials in RIS 2014-11, and therefore the effect of embrittlement on the P/T limits to 55 EFPY need not be determined for the St. Lucie, Unit 2 nozzles. As such, the 55 EFPY P/T limits of the St. Lucie, Unit 2 RPV beltline shell base metal and weld materials will remain the most limiting. Thus, the staff finds that the licensee adequately addressed other ferritic RPV components described in RIS 2014-11 that could potentially be more limiting.

In Appendix C of WCAP-18275-NP, the licensee addressed ferritic RCS pressure boundary components that are not part of the St. Lucie, Unit 2 RPV. These components include the replacement closure head, the pressurizer, and replacement steam generators. The staff determined that the licensee has adequately addressed the consideration of P/T limits of the replacement closure head by including the flange requirements in the proposed 55 EFPY P/T limits, as discussed above. The staff determined that the pressurizer and replacement steam generators are not expected to receive neutron fluence levels such that they need to be considered for P/T limits evaluation. The staff, therefore, finds the licensee's consideration of ferritic RCS pressure boundary components that are not part of the St. Lucie, Unit 2 RPV acceptable.

<u>P/T Limits Evaluation Conclusion</u>

Based on the above evaluation, the staff finds that the proposed 55 EFPY RCS P/T limits for St. Lucie, Unit 2 satisfy the requirements of Appendices G and H to 10 CFR Part 50 because the licensee has adequately incorporated the effects of RPV beltline embrittlement up to 55 EFPY on the P/T limits consistent with Appendix G to 10 CFR Part 50 and used RPV material monitoring data consistent with Appendix H to 10 CFR Part 50. The staff finds that the proposed 55 EFPY RCS P/T limits for St. Lucie, Unit 2 satisfy the requirements of Appendix G to Section XI of the ASME Code because the P/T limits were determined based on methodologies consistent with Appendix G to Section XI. As such, the staff finds that the proposed 55 EFPY RCS P/T limits for St. Lucie, Unit 2 satisfy the acceptance criteria of 10 CFR 50.60 for fracture prevention of the RPV during normal operation and those aspects of GDCs 14 and 31 associated with fracture prevention of the RCS pressure boundary. Hence, the staff determined that incorporating the proposed 55 EFPY RCS P/T limits into the St. Lucie, Unit 2 TS 3/4.4.9 is acceptable.

3.2.2 LTOP Evaluation

As specified above in Section 3.2 of the SE, the LAR provides a discussion of the LTOP analysis used to support the proposed P/T limits. The NRC staff reviewed the LTOP analysis to determine whether the analysis is acceptable for supporting the P/T limits. The results of the staff evaluation are discussed below.

Section 5.2.2, "Overpressure Protection" of the Standard Review Plan (SRP) specifies that the LTOP system be designed in accordance with the guidance of Branch Technical Position (BTP) 5-2, which specifies that the LTOP system be capable of relieving pressure during all anticipated overpressure events at a rate sufficient to satisfy the TS limits while operating at low temperatures.

LTOP Analysis

Section 5.2.6 of the UFSAR for St. Lucie, Unit 2, discusses the LTOP system and indicated that the overpressure protection of the RCS during low-temperature conditions is provided by two power-operated relief valves (PORVs) connected to the pressurizer steam space, and by the shutdown cooling system (SDCS) relief valves when the SDCS is in operation. The protection provided by the PORVs and SDCS relief valves precludes any over pressurizing transient from exceeding the technical specification P/T operating limits. The LTOP provided by the relief valves is required during heatup and cooldown and during extended periods of cold shutdowns. The most limiting transients during reactor heatup and cooldown operation are an inadvertent safety injection actuation (mass input), and a reactor coolant pump start when a positive steam generator to reactor vessel ΔT exists (energy input).

On page 5 of the LAR, the licensee stated that as part of the LTOP evaluation methodology, the PORV pressure setpoint and LTOP enable temperatures both include instrument uncertainties in the transient evaluation. These allowable pressures are compared to peak pressurizer pressures that occur during the limiting design basis LTOP over-pressurization events. At temperatures greater than the LTOP enable temperatures, operation below and to the right of the P/T curves is administratively controlled with margins greater than the instrument uncertainty. As discussed in Section 3.2.1 of this SE, the P/T limit curve generation methodology is consistent with the NRC-approved methodology, as documented in WCAP-14040-A, Revision 4. In Attachment 3 of the LAR, the licensee discussed its LTOP analysis.

The LTOP evaluation reduces the reactor vessel beltline allowable pressures to that representative of pressure limits in the pressurizer (where plant instrumentation measures RCS pressure). The two design-basis LTOP over-pressurization events for St. Lucie are the mass addition event and the energy addition (RCP start) event. The limiting or controlling peak transient pressures are 546.5 psia during Cold Shutdown (Mode 5, $T_{RCS} \leq 200°F$) and 677 psia during Hot Shutdown (Mode 4, $T_{RCS} > 200°F$). The limiting transient analysis pressure results are based on the PORV setpoint of 490 psia provided in St. Lucie, Unit 2 TS 3.4.9.3.

LCO 3.4.9.3 identifies the enable temperature ranges for the LTOP system, as shown in TS Table 3.4-3, and the minimum temperature for PORV use for LTOP, as shown in TS Table 3.4-4. The enable temperatures in the proposed Table 3.4-3 were increased for the 55 EFPY period to 252°F during heatup, and to 240°F during cooldown. This was to represent more restrictive P/T limit curves, which includes a margin of 14°F to

account for instrument uncertainty. The minimum cold leg temperatures for PORV use for LTOP in the proposed Table 3.4-4 were revised down to the minimum bolt up temperature of 60°F during heatup and increased to 149°F during cooldown with the inclusion of 14°F margin for instrument uncertainty.

As shown in the proposed TS Figure 3.4-2, during heatup, the peak transient pressure of 546.5 psia between 60°F and 200°F does not exceed the P/T limits within that temperature range. In addition, the peak transient pressure of 677 psia between the LTOP enable temperatures of 200°F and 252°F does not exceed the P/T limits within that temperature range. The staff, therefore, finds this acceptable.

The proposed TS Figure 3.4-3 shows that during cooldown, the peak transient pressure of 546.5 psia between 149°F and 200°F does not exceed the allowable cooldown limits within that temperature range. In addition, the peak transient pressure of 677 psia between the LTOP enable temperature range of 200°F and 240°F does not exceed the P/T limits within that temperature range. Therefore, the staff finds this acceptable.

<u>LTOP Conclusion</u>

The NRC staff reviewed the LAR and related documentation, including the St. Lucie, Unit 2 UFSAR and the TS. The staff concludes that the proposed amendment to replace the current time-limited RCS P/T limit curves with the new curves effective for 55 EFPY and the protection provided by the relief valves provide reasonable assurance to preclude any over pressurizing transient from exceeding the TS P/T operating limits at low temperatures for the following reasons:

1. The evaluation methodology was NRC-approved and was based on a conservative valve setpoint such that the PORV pressure setpoint and LTOP enable temperatures both included appropriate instrument uncertainties in the transient evaluation.

2. The LTOP system is capable of relieving pressure during anticipated overpressure events at a rate sufficient to satisfy the TS limits while operating at low temperatures. Furthermore, the limiting LTOP transient event pressures are bounded by the proposed new TS allowable P/T limit curves depicted in Figures 3.4-2 and 3.4-3 of St. Lucie, Unit 2 TS for heatup and cooldown, respectively.

Based on the discussion above and in Section 3.1 of this safety evaluation, the NRC staff determined the following:

(1) the LTOP analysis adequately supports the limiting pressure for the heatup and cooldown curve at 55 EFPY;

(2) the limiting pressure for the heatup and cooldown curve at 55 EFPY in combination with the TS 3.4.9.3 requirement of the PORV setpoint of 490 psia would reasonably assure that the LTOP analysis remained valid in meeting GDC 15. This ensures that the requirements of the reactor coolant pressure boundary design conditions are not exceeded and are consistent with BTP 5-2 (as it relates to the guidance of over-pressurization protection of pressurized water reactors while operating at low temperatures); and

(3) the limiting pressure for the heatup and cooldown curve at 55 EFPY and relief valve setpoints in TS LCO 3.4.9.3 would meet 10 CFR 50.36(c)(2) because the licensee is required to comply with TS Tables 3.4-3 and 3.4-4 in order to operate within a low temperature RCS overpressure protection operating range in order to assure safe operation of a nuclear reactor. Therefore, the staff determined that the limiting pressure for the heatup and cooldown curve, the relief valve setpoints in TS LCO 3.4.9.3, and the low temperature overpressure protection operating range in TS Tables 3.4-3 and 3.4-4 are acceptable for plant operation to 55 EFPY for St. Lucie, Unit 2.

Based on the above, the NRC staff concludes that the proposed LTOP evaluation for St. Lucie, Unit 2 is acceptable.

3.2.3 RPV Neutron Fluence Evaluation

The NRC staff evaluated the acceptability of the licensee's reactor pressure vessel (RPV) fluence evaluation used as input to determine the revised P/T limits. The licensee provided, in Attachment 2 to the LAR, WCAP-18275-NP, Revision 0, "St. Lucie, Unit 2 Heatup and Cooldown Limit Curves for Normal Operation through End of License Extension," in which Chapter 2 describes the RPV neutron fluence evaluation. The evaluation was performed using methods described in WCAP-14040-A, Revision 4. Fluence projections were provided by the licensee for the vessel shells, girth welds, longitudinal welds, and for the nozzle forging attachment welds and nozzle forging postulated ¼ T flaws.

The methods described in WCAP-14040-A, Revision 4, have been generically reviewed and approved by the NRC staff, and have been found to be consistent with RG 1.190. The methods are also qualified for use with the CE RPV geometry, as noted, for example, in Section 4.2.1 of NUREG-1961, "Safety Evaluation Report Related to the License Renewal of Palo Verde Nuclear Generating Station, Units 1, 2, and 3" (ADAMS Accession No. ML11095A011). Since the methods are generically approved for use and are also qualified for use with the reactor vessel geometry, the NRC staff determined that the fluence methods are acceptable for use for this application at St. Lucie, Unit 2.

The fluence methods described in WCAP-14040-A are adequate for estimating fluence at reactor pressure vessel elevations that are in close proximity to active fuel in the core. However, Girth Weld 201-141 and Nozzle Attachment Welds 103-121 and 105-121 are located significantly below (201-141) and above (103-121 and 15-121) the core (i.e., they are considered extended beltline welds). The applicability of the methods described in WCAP-14040-A is not generically established for such weld locations, and the use of such methods requires further evaluation when incorporating those fluence estimates into the P/T limits curve development. The licensee screened the fluence estimates against two thresholds to determine whether further analysis for these welds was necessary.

According to RIS 2014-11, any reactor vessel materials that have an exposure higher than 1 x 10^{17} n/cm^2 require consideration in the development of P/T limit curves. As referenced by the licensee, according to Pressurized Water Reactor Owners Group (PWROG), PWROG 15109-NP-A, Revision 0 (ADAMS Accession No. ML20024E573), the fluence for components that are located at significant distances from the core could have fluences as high as 4.28 x 10^{17} n/cm^2 without contributing a significant impact on the P/T limit curves. Because the fluence estimates for these welds at 55 EFPY did not exceed 8 x 10^{16} n/cm^2, the licensee concluded that these welds do not require consideration in the development of the P/T limit curves. As the

threshold in RIS 2014-11 is lower than that described in PWROG-15109-NP-A, the NRC staff based its conclusions upon the threshold in RIS 2014-11, rather than the threshold identified in PWROG-15109-NP-A. The NRC staff observed that the licensee's peak estimated fluence for the extended beltline welds was below both the RIS 2014-11 and PWROG-15109-NP-A threshold levels. Since the fluence was less than the threshold contained in RIS 2014-11, the NRC staff determined that the licensee's treatment of these welds (i.e., not considering them in the development of the P/T limit curves) was acceptable.

RPV Fluence Conclusion

Based on the considerations discussed above, the NRC staff determined that the fluence analysis described in Chapter 2 of WCAP-18275-NP is acceptable. The values were determined using NRC-approved methods consistent with RG 1.190 guidance, and as such, satisfy the criteria of GDCs 14 and 31. In the extended beltline, the estimated exposure is adequately low, and reasonable amounts of margin exist to the 1×10^{17} n/cm^2 threshold such that these welds do not require treatment as RPV beltline materials at 55 EFPY exposure.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official on January 19, 2021, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area, as defined in 10 CFR Part 20, "Standards for Protection Against Radiation." The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding, which was published in the *Federal Register* on July 28, 2020 (85 FR 45448), that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: February 26, 2021

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 - 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) DATED FEBRUARY 26, 2021

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