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

June 27, 2024

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

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 – RELIEF REQUEST - PSL2-I5-RR-01: PROPOSED ALTERNATIVE TO AMSE CODE XI CODE EXAMINATION REQUIREMENTS - SYSTEM LEAKAGE TEST OF REACTOR PRESSURE VESSEL BOTTOM HEAD AND CLASS 1 AND 2 PIPING IN COVERED TRENCHES (EPID L-2023-LLR-0038)

Dear Bob Coffey:

By letter dated July 26, 2023, Florida Power and Light Company (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission for relief from certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements for the St. Lucie Plant, Unit No. 2 facility. The licensee requested relief from the requirements of the ASME Code, Section XI, Articles IWA-5000, IWB-5000, and IWC-5000 for performing visual examinations of the reactor vessel and associated Class 1 and 2 piping in covered trenches rendered inaccessible in conjunction with the pressure testing of Class 1 and 2 components.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2) the licensee submitted Relief Request PSL2-I5-RR-01, Revision 0 proposing an alternative system leakage test on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Sincerely,

DAVID
WRONA

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WRONA
Date: 2024.06.27
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David 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
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST PSL2-I5-RR-01, REVISION 0

SYSTEM LEAKAGE TEST OF REACTOR PRESSURE VESSEL BOTTOM HEAD AND

CLASS 1 AND 2 PIPING IN COVERED TRENCHES

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated July 26, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23207A157), Florida Power and Light Company (FPL, the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for relief from certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements at the St. Lucie Plant, Unit No. 2 (St. Lucie, Unit 2). The licensee requested relief from the requirements of the ASME Code, Section XI, Articles IWA-5000, IWB-5000, and IWC-5000 for performing visual examinations of the reactor vessel and associated Class 1 and 2 piping in covered trenches rendered inaccessible in conjunction with the pressure testing of Class 1 and 2 components.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2) the licensee submitted Relief Request PSL2-I5-RR-01, Revision 0 proposing an alternative system leakage test on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), the ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” to the extent practical, within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The licensee must demonstrate (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the NRC to authorize, the alternative proposed by the licensee.

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Relief Request

3.1.1 ASME Code Components Affected

Examination Category	Item No.	Component Description
B-P	B15.10 B15.20	Reactor Vessel – Pressure Retaining Boundary Bottom Head Area. Material: Carbon Steel
		Piping - Pressure Retaining Boundary (Covered portions only) Material: Stainless Steel SI [safety injection] Headers: Lines 12-SI-148, 149, 150, 151 Charging Line: 2-CH-147 Letdown Line: 2-RC-142
C-H	C7.10	Piping - Pressure Retaining Components (Covered portions only) Material: Stainless Steel SDC [shutdown cooling] Suction Lines:10-SI-362, 363 Hot Leg Injection Lines: 3-SI-179, 181

3.1.2 Applicable Code Edition and Addenda

The code of record for the fifth 10-year inservice inspection (ISI) interval is the ASME Code, Section XI, 2019 Edition.

3.1.3 Applicable Code Requirements

The requirements for performing visual examinations in conjunction with the pressure testing of Class 1 and 2 components are provided in the ASME Code, Section XI, IWA-5000, IWB-5000, and IWC-5000.

Paragraph IWA-5241(b) states, in part, that: "For components whose external surfaces are inaccessible for direct VT-2 visual examination, only the examination of the surrounding area (Including floor areas or equipment surfaces located underneath the components) for evidence of leakage shall be required."

Paragraph IWA-5241(h) states, in part, that: "When examining insulated components, the examination of surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage, or other areas to which leakage may be channeled, shall be required."

Paragraph IWB-5221(b) states, in part, that: "For all other systems, the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power."

Paragraph IWC-5221(a) states, in part, that: "For Class 2 [Table IWC-2500-1 (C-H)] components operated continuously or routinely during normal plant operation, cold shutdown, or refueling operations, the system leakage test shall be conducted at the system pressure obtained while the system, or portion of the system, is performing its safety function. If portions of a system are associated with more than one safety function, the visual examination need only be performed during a test conducted at the higher of the operating pressures for the respective system safety function."

Paragraph IWC-5221(b) states, in part, that: "For Class 2 [Table IWC-2500-1 (C-H)] components in standby systems (or portions of standby systems) that are not operated routinely except for testing, the leakage test shall be conducted at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy technical specification surveillance requirements). If portions of a system are associated with more than one safety function, the visual examination need only be performed during the test conducted at the higher of the test pressures for the respective system safety function."

3.1.4 Reason for the Request

The licensee stated that the design of St. Lucie, Unit 2 does not provide access for a direct visual examination of the reactor vessel bottom area during the ASME Code, Section XI system leakage test and associated VT-2 visual examination. The licensee explained that three possible pathways could lead to the reactor vessel bottom area. Two are in the electrical tunnel at the bottom of the containment "keyway" and are blocked by the reactor cavity relief dampers (blast dampers). These dampers consist of horizontal louvers approximately 11-inch wide and normally remain in the closed position. They are not intended for human passage. The third pathway is through the reactor cavity sump, a small tunnel from the cavity to the weir pit. A cooling duct runs through this tunnel limiting the height to a crawl space of approximately 1-foot high and 6 to 8 feet long. The licensee stated that ambient conditions during VT-2 examinations at normal operating conditions create an extreme heat stress environment and combined with a nearly impossible exit pathway, make examination of this area an excessively hazardous work situation. The licensee has considered the reactor bottom area to be inaccessible for examination at normal operating conditions.

The licensee stated that some segments of Class 1 and Class 2 reactor support piping pass through trenches that are covered and secured during normal operation. These trenches are required to be covered and secured prior to entering Mode 4 following a shutdown to ensure containment sump recirculation flow paths are maintained. The licensee further stated that the trench covers prohibit direct examination of horizontal insulation joints and low points as required by the ASME Code, Section XI, IWA-5241. According to the licensee, due to gaps and handholes in the trench covers and the use of grating in some locations, surrounding areas can be observed for evidence of leakage. The licensee explained that areas to which leakage may be channeled are also open in many locations throughout the containment for observation during the system leakage test at St Lucie, Unit 2.

3.1.5 Proposed Alternative

The licensee requested to visually examine the reactor vessel bottom head area and piping in covered trenches at different plant conditions than those required by the ASME Code, Section XI. The licensee stated that it will continue to perform the required system pressure tests as prescribed by the ASME Code, Section XI, IWB-5000 each refueling outage and IWC-5000 each period and will examine all accessible components in accordance with IWA-5241.

The licensee stated that for those portions of components rendered inaccessible by containment building configuration, it will open the inaccessible areas each refueling outage and perform a VT-2 examination of the reactor vessel bottom and other associated piping following plant cooldown and depressurization. This inspection will check insulation surfaces and joints for signs of leakage or residue. The licensee further stated that it will evaluate any evidence of leakage in accordance with IWA-5250, which may include additional inspections and insulation removal as deemed necessary.

3.1.6 Basis for Use

The licensee stated that the evidence of leakage can be identified by visual examination following cooldown for refueling. The St. Lucie reactors have no bottom head penetrations and have been volumetrically examined in accordance with the rules of the ASME Code, Section XI with no relevant indications identified. The licensee further stated that it does not expect leakage due to the configuration of the bottom of the reactor pressure vessel. According to the licensee, the reactor cavity is monitored for leakage continuously during operation, and inventory balance is performed daily throughout the operating cycle.

3.1.7 Duration of Relief Request

The licensee proposed the alternative for the fifth 10-year ISI interval of the ISI Program for St. Lucie, Unit 2 that commences on August 8, 2023, and is scheduled to end on August 7, 2033.

3.2 NRC Staff Evaluations

The NRC staff evaluated whether (1) compliance with the requirements of the ASME Code, Section XI, IWA-5000, IWB-5000, and IWC-5000 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and (2) the proposed alternative would provide reasonable assurance of structural integrity of the affected components.

3.2.1 Reactor Pressure Vessel Bottom Head Area

The ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, Item Nos. B15.10 and B15.20 requires that a system leakage test and VT-2 visual examination be performed per IWB-5220 during every refueling outage. The reactor vessel bottom head area is part of Item Nos. B15.10 and B15.20. The ASME Code, Section XI, IWB-5221(b) requires that the system leakage test and associated VT-2 examination be conducted at a pressure not less than 100 percent rated reactor power. The licensee proposed to perform the VT-2 visual examination of the reactor vessel bottom head area at different plant conditions than the 100 percent rated reactor power because of the hazardous conditions for the plant examiner. The licensee stated that the plant design does not provide a direct visual examination of the reactor vessel bottom head area. The three possible pathways to visually examine the reactor vessel bottom head area are inaccessible. The NRC staff noted that conducting the VT-2 visual examination of the reactor vessel bottom head area under the 100 percent rated reactor power would be excessively hazardous due to the extreme high temperature environment. In addition, the pathways are inaccessible which are blocked by blast dampers and cooling conduits. Therefore, the NRC staff determined that imposing the VT-2 examination of the reactor vessel bottom head area at the 100 percent rated power condition would result in hardship or unusual difficulty for the licensee.

As an alternative to the ASME Code, Section XI requirements, the licensee proposed to perform a VT-2 examination of the reactor vessel bottom area following plant cooldown and depressurization. The licensee stated that it will open the inaccessible areas to perform the visual examination. The licensee stated that it will perform the system pressure test as established by the ASME Code, Section XI, IWB-5000 each refueling outage, and it will examine all accessible components in accordance with IWA-5241. The licensee further stated that it will evaluate any evidence of leakage in accordance with IWA-5250, which may include additional inspections and insulation removal as deemed necessary. The NRC staff noted that the licensee will check insulation surfaces and joints for signs of leakage or residue.

The NRC staff noted that operational experience in pressurized-water reactors (PWRs) shows that reactor vessel leakage is most likely to occur in the reactor vessel closure head area originating from the control rod drive mechanism nozzle penetrations, control rod drive housing, or the valves located above the control rod drive mechanism equipment. However, regulations in 10 CFR 50.55a(g)(6)(ii)(D) require that all licensees of PWRs inspect their reactor vessel closure head in accordance with ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1." Other than in the reactor vessel closure head, operational experience demonstrates that leakage from other areas of the reactor vessel shell is unlikely. The NRC staff noted that the reactor vessel bottom head at St. Lucie, Unit 2 has no penetrations which eliminates a source of potential leakage.

In addition, the NRC staff expects that if the reactor vessel does develop leakage, the existing reactor coolant leakage detection systems will be able to detect the leakage during normal operation and the control room operators will be notified. As a result, the licensee will take appropriate corrective actions in accordance with the plant technical specifications.

The NRC staff finds that conducting the VT-2 visual examination at the cooldown temperature and depressurization during the system leakage test does not significantly reduce the effectiveness of the visual examination as compared to the visual examination conducted at the 100 percent power given the hazardous conditions for the plant examiner. The NRC staff finds that the proposed alternative is sufficient to detect evidence of reactor vessel leakage in the vessel bottom head area.

3.2.2 Class 1 and 2 Reactor Support Piping in Covered Trenches

For the affected Class 1 piping, the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, Item Nos. B15.10 and B15.20 require that a system leakage test and VT-2 visual examination be performed per IWB-5220 during every refueling outage.

For the affected Class 2 piping, the ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H, Item No. C7.10 requires that a system leakage test and VT-2 visual examination be performed per IWC-5220 during every inspection period. Examination Category C-H requires that the VT-2 examination be conducted per IWA-5240.

The portions of affected Class 1 and 2 piping are in trenches that are required to be covered and secured during normal operation to ensure containment sump recirculation flow paths are maintained. As such, the affected piping is inaccessible for the VT-2 examination in accordance with IWA-5241.

The NRC staff determined that requiring the licensee to comply with IWB-5221 for the system leakage test of inaccessible portions of Class 1 piping and IWC-5221 for the system leakage test of inaccessible portions of Class 2 piping would result in hardship.

Regarding the inspection of the affected piping in the covered trenches, the NRC staff determined that the licensee would open the inaccessible areas following plant cooldown and depressurization for refueling and perform the VT-2 visual examinations of the affected pipes in covered trenches. Therefore, the NRC staff determined that the licensee's proposed system leakage test and associated VT-2 examination conducted following plant cooldown and depressurization is adequate, because any evidence of leakage or boric acid residue will be discernable during the VT-2 examination.

The NRC staff noted that the affected Class 1 and 2 pipes are made of stainless steel. Potential degradation mechanisms of these pipes may include fatigue and stress corrosion cracking. The NRC staff noted that low cycle fatigue cracks are known to have relatively slow growth, and field experience has shown that stress corrosion cracking under the conditions associated with the affected piping is not expected. The NRC staff recognized that if a leak does occur in the affected pipes, the proposed VT-2 visual examination and the system leakage test would be able to detect such a leak. In addition, the NRC staff expects that if the affected pipes developed leakage, the existing reactor coolant leakage detection systems will be able to detect the leakage during normal operation and the control room operators will be notified as discussed above. Therefore, the NRC staff determines that, based on the alternative system leakage testing and the VT-2 visual examinations performed following plant cooldown and depressurization, it is reasonable to conclude that pipe leakage, if it occurs, will be detected either by the proposed system leakage test or the reactor coolant system leakage detection systems.

Therefore, the NRC staff finds that the proposed system leakage test performed, following plant cooldown and depressurization during each refueling outage for the affected Class 1 piping and during each inspection period for the affected Class 2 piping, is adequate to provide reasonable assurance of structural integrity and leak tightness of the inaccessible portions of the affected Class 1 and 2 piping. The NRC also finds that performing a system leakage test in compliance with the requirements of the ASME Code, Section XI, IWA-5000, IWB-5000, and IWC-5000 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the reactor vessel bottom head area and the inaccessible portions of the affected Class 1 and 2 pipes. The NRC staff further determines that complying with the specified ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the

NRC staff concludes that the licensee has adequately addressed all regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of Relief Request PSL2-I5-RR-01, Revision 0 at St. Lucie, Unit 2 for the fifth 10-year ISI interval, which will end on August 7, 2033.

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

Principal Contributors: Eric Palmer, NRR
John Tsao, NRR
Mat Burton, NRR

Date June 27, 2024

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 – RELIEF REQUEST - PSL2-I5-RR-01: PROPOSED ALTERNATIVE TO AMSE CODE XI CODE EXAMINATION REQUIREMENTS - SYSTEM LEAKAGE TEST OF REACTOR PRESSURE VESSEL BOTTOM HEAD AND CLASS 1 AND 2 PIPING IN COVERED TRENCHES (EPID L-2023-LLR-0038) DATED JUNE 27, 2024

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