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

May 31, 2015

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE PLANT, UNIT NO. 2 – FOURTH 10-YEAR INTERVAL RELIEF REQUEST NO. 4, REVISION 1 (TAC NO. MF4473)

Dear Mr. Nazar:

By letter dated June 30, 2014, as revised by letter dated November 6, 2014, Florida Power and Light Company (the licensee) submitted Relief Request (RR) No. 4, Revision 1, 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). The licensee requested relief from the ASME requirements 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 as provided in ASME Code Section XI, Articles IWA-5000, IWB-5000, and IWC-5000.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee proposed 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. By *Federal Register* Notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014, the paragraph headings in 10 CFR 50.55a were revised. Accordingly, RRs that had been previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1). RRs that had been previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

As set forth above, the NRC staff determined that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the reactor pressure vessel bottom head and the inaccessible portions of Class 1 and 2 pipes under consideration. Complying with the specified requirement 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 of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of RR No. 4, Revision 1 at St. Lucie for the fourth 10-year inservice inspection interval, which started on August 8, 2013, and will end on August 7, 2023.

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.

If you have any questions, please contact the Project Manager, Farideh Saba, at 301-415-1447 or Farideh.Saba@nrc.gov.

Sincerely,

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 4, REVISION 1 - 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 June 30, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14203A005), as revised by letter dated November 6, 2014 (ADAMS Accession No. ML14325A691), 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). The licensee requested relief from the ASME requirements 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 as provided in ASME Code Section XI, Articles IWA-5000, IWB-500, and IWC-5000.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee proposed 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. By *Federal Register* Notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014, the paragraph headings in 10 CFR 50.55a were revised. Accordingly, relief requests (RRs) that had been previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1). RRs that had been previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

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 requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Relief Request

Applicable Code Edition and Addenda

The code of record for the fourth 10-year inservice inspection (ISI) interval is the 2007 Edition with 2008 Addenda to the ASME Code.

Duration of Relief Request

The licensee submitted RR No. 4, Revision 1, for the fourth 10-year ISI interval, which started on August 8, 2013, and will end on August 7, 2023.

Applicable Code Requirements (as stated)

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

Paragraph IWA-5241(b) states, "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, "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 such leakage may be channeled, shall be required."

Paragraph IWB-5221(a) states, "The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power."

Paragraph IWC-5221 states the following:

> The system leakage test shall be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or 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).

3.1.1 Reactor Pressure Vessel Bottom Head

ASME Code Components Affected and Requirements

The affected component is the ASME Code Class 1 Reactor Vessel-Pressure Retaining Bottom Head, classified as ASME Code Class 1, and is identified as Item Number B15.10 in Examination Category B-P of ASME Code Section XI, Table IWB-2500-1.

ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, "All Pressure Retaining Components," requires that the system leakage test be conducted in accordance with IWB-5220 and the associated VT-2 visual examination in accordance with IWA-5240 prior to plant startup, following a reactor refueling outage and every following refueling outage. IWA-5241(a) requires that the VT-2 examination be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage. If the external surface is inaccessible for direct VT-2 visual examination, IWA-5241(b) allows examination of only the surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage. Furthermore, IWB-5221(a) requires that the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power.

ASME Code Requirements for Requested Relief

The licensee requested relief from performing the System Leakage Test (Visual VT-2) for each inspection period per ASME Section XI, Table IWC-2500-1, Examination Category B-P, Item B15.10.

The licensee requests approval for an alternative to perform the examination of the reactor vessel head area at conditions that differ from those required by Section XI, paragraphs IWA-5241 and IWB-5221.

Proposed Alternative and Reason for Relief

Pursuant to 10 CFR 50.55a(z)(2), the licensee proposed an alternative to the requirements set forth in IWB-5221(a) for the reactor vessel bottom head area. Due to the inaccessibility at 100 percent rated reactor power, in its letter dated November 6, 2014, the licensee requested approval to open the inaccessible areas and perform a VT-2 examination of the reactor vessel bottom head each refueling outage following plant depressurization and cooldown.

The licensee stated that the objective of the required visual examination at normal operating conditions is to detect evidence of leakage and thereby verify the integrity of the reactor coolant system pressure boundary. FPL stated that the same 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 ASME Code, Section XI with no relevant indications identified. There is no expectation of leakage due to the solid configuration of the bottom of the reactor pressure vessel (RPV). In addition, the reactor cavity is monitored for leakage continuously during operation, and inventory balance is performed daily throughout the operating cycle. Therefore, FPL concluded that the proposed alternative provides reasonable assurance of system integrity and an acceptable level of quality and safety comparable to an examination performed at normal operating conditions.

The licensee further stated that FPL will continue to perform required system pressure tests as required by IWB-5000 each refueling outage and all accessible components will be examined in accordance with the IWA-5241 requirements.

Basis for Hardship

The licensee explained in its submittals that St. Lucie has three pathways that lead to the reactor vessel bottom head area. Two of the pathways are blocked by the reactor cavity relief dampers that are usually closed and not intended for human passage. The third pathway is through the reactor cavity sump that is limited to a crawl space due to a cooling duct that runs through it. The heightened heat stress of the environment at normal operating conditions, along with a nearly impossible exit pathway, would create an excessively hazardous environment for the personnel conducting the required examinations. During a refueling outage, following depressurization and cooldown, the licensee will be able to access and examine the reactor vessel bottom head area by opening the dampers; this is not possible during normal operation.

The licensee added that the purpose of performing the examinations during normal operation is to find any evidence of leakage that would compromise the integrity of the reactor coolant system. Evidence of leakage or corrosion that occurred during the previous fuel cycle should be able to be seen following cooldown and depressurization. Therefore, conducting the examinations at normal operating conditions would result in hardship and unnecessary difficulty without a compensating increase in the level of quality and safety.

3.1.2. Class 1 and 2 Reactor Support Piping in Covered Trenches

Components Affected

The components affected are ASME Code Class 1 and 2 piping. In accordance with IWB-2500 (Table IWB-2500-1), the Class 1 piping is categorized as Examination Category B-P, Items B15.10 and B15.20. In accordance with IWC-2500 (Table IWC-2500-1), the Class 2 piping is classified as Examination Category C-H, Item Number C7.10.

In the table in Section 1 of Attachment to RR No. 4, Revision 1, the licensee identified the Class 1 reactor support piping as the safety injection headers, the charging, and the letdown. The licensee identified the Class 2 reactor support piping as the shutdown cooling suction and the hot leg injection. In its November 6, 2014, letter, the licensee stated that the material of construction for these pipes is stainless steel.

ASME Code Requirement

ASME Code, Section XI, IWB-2500, Table IWB-2500-1, Examination Category B-P, requires the system leakage test be conducted according to IWB-5220 and the associated VT-2 visual examination according to IWA-5240 prior to plant startup following each refueling outage. In accordance with IWB-5221(a), the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power. In accordance with IWB-5222(a), the pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity. In accordance with IWB-5222(b), the Class 1 pressure retaining boundary, which is not pressurized when the system valves are in the position required for normal reactor startup, shall be pressurized and examined at or near the end of the inspection interval. This boundary may be tested in its entirety, or in portions, and testing may be performed during the testing of the boundary of IWB-5222(a).

ASME Code, Section XI, IWC-2500, Table IWC-2500-1, Examination Category C-H, requires the system leakage testing according to IWC-5220 and the associated VT-2 visual examinations according to IWA-5240 during each inspection period. In accordance with IWC-5221, the system leakage test shall be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or 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). In accordance with IWC-5222(a), the pressure retaining boundary includes only those portions of the system required to operate or support the safety function up to, and including, the first normally closed valve (including a safety or relief valve) or valve capable of automatic closure when the safety function is required.

Proposed Alternative and Reason for Relief

The licensee proposed an alternative to IWB-5221(a) for the inaccessible portion of Class 1 piping and an alternative to IWC-5221 for the inaccessible portion of Class 2 piping located in covered trenches in the reactor bottom area. These portions of the Class 1 and 2 piping pass through trenches that are covered and secured during normal operation. Therefore, they are not accessible for direct VT-2 visual examinations that accompany the system leakage test as required by the ASME Code. The licensee proposed to perform the system leakage test and the VT-2 visual examinations of these inaccessible portions of the Class 1 and 2 pipes following plant cooldown and depressurization during each refueling outage. The licensee stated that evidence of leakage can be identified by the VT-2 visual examinations performed in accordance with IWA-5241 following plant cooldown and depressurization for refueling.

In addition, the licensee stated that the reactor cavity is continuously monitored for leakage during operation, and inventory balance is performed daily throughout operating cycle; therefore, should any leakage occur in these portions of the pipes, the leakage will be detected.

Basis for Hardship

The licensee stated that portions of Class 1 and 2 pipes located in the reactor bottom area pass through trenches that are covered and secured during normal operation. These trenches shall be covered and secured prior to entering Mode 4 following a shutdown to ensure the containment sump recirculation flow paths are maintained. This is outlined in the St. Lucie response to NRC Bulletin 2003-01 (FPL Letter L-2003-201). The trench covers prohibit direct examination of horizontal insulation joints and low points as required by IWA-5241. Furthermore, the nearly impossible exit pathway and extreme high heat in the reactor bottom area during normal operation create safety hazards for personnel conducting the ASME Code VT-2 visual examinations that accompany the leakage test.

In its letter dated November 6, 2014, the licensee stated that there is no direct access to the bottom area of the reactor to perform the ASME Code examination at the required pressure and temperature. During refueling outage following plant cooldown and depressurization, the licensee can obtain access to the reactor bottom area by propping open the damper. Access in this manner is not possible while the plant is in normal operation. Due to gaps and hand-holes in the trench covers and use of grating in some locations, the licensee can observe the surrounding areas for evidence of leakage during refueling outages following plant cooldown and depressurization. Areas to which leakage may be channeled are also open in many locations throughout the containment for observation during the proposed system leakage test. This visual examination is in compliance with the requirements of IWA-5241(h).

3.2 NRC Staff Evaluations

The NRC staff has evaluated RR No. 4, Revision 1, pursuant to 10 CFR 50.55a(z)(2). The NRC staff focused on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

3.2.1 Reactor Pressure Vessel Bottom Head

The fourth 10-year ISI interval RR No. 4, Revision 1, for St. Lucie pertains to VT-2 visual examinations of the bottom of the reactor vessel at different plant conditions than required by the ASME Code. The ASME Code requires that the system leakage tests be conducted at a pressure no less than what corresponds to 100 percent rated reactor power. The NRC staff finds that conducting the examinations under these conditions would be excessively hazardous, due to the extreme heat stress environment from the high pressure, paired with a nearly impossible exit pathway due to blast dampers and a cooling duct. Therefore, imposing the examination requirements would result in hardship or unusual difficulty on the licensee.

As an alternative to the requirements, the licensee proposed to perform the required system pressure tests as established by IWB-5000 each refueling outage, and the licensee will examine all accessible components in accordance with IWA-5241. For the portions of components that were deemed inaccessible due to the configuration of the containment building, the licensee initially proposed to open the inaccessible areas and perform VT-2 examinations of the reactor vessel bottom and other associated piping once each period during a refueling outage. The ASME Code, however, requires that the Reactor Vessel – Pressure Retaining Boundary Bottom

Head Area be examined during each refueling outage. Therefore, the staff requested that the licensee explain its reasoning for extending its examination schedule. The licensee revised its RR and submitted an updated version in its November 6, 2014, letter, where the licensee agreed to perform the examinations during each refueling outage.

The licensee also proposed to perform the examinations following plant cooldown and depressurization, which will create an environment without the hazardous conditions associated with normal operating pressure and temperature. The licensee stated that the examinations will include checking insulation surfaces and joints for signs of leakage or residue, and any evidence of leakage will be evaluated in accordance with IWA-5250, which may include additional inspections and insulation removal as deemed necessary. Any evidence of leakage and boric acid corrosion that occurred during the previous fuel cycle can be detected by visual examination of this area at the end of the cycle during the outage. The NRC staff finds that conducting VT-2 visual examinations during each refueling outage would provide reasonable assurance that leaks throughout the RPV that occurred during the previous cycle would be detected. Furthermore, the boric acid crystals and corrosion resulting from leakage should also be detected by the proposed VT-2 visual examination during each refueling cycle.

Lastly, operational experience shows that RPV leakage is most likely to occur in the closure (top) head area originating from penetrations with dissimilar metal welds. In order to ensure that this leakage is found in a timely manner, 10 CFR 50.55a(g)(6)(ii)(D) requires that all licensees of pressurized-water reactors (PWRs) augment their ISIs with ASME Code Case N-729-1, which requires extensive examination of the top head, nozzles, and partial-penetration welds. Operational experience in PWRs shows little or no leakage outside of the closure head region, but would most likely occur in penetrations and dissimilar metal welds if it were to occur. In PWRs designed by Combustion Engineering, Inc., with the exception of the closure head, penetrations or dissimilar metal welds are generally associated with the flange leak detection lines in which there is no operational experience with leakage. Therefore, other than in the closure head, which is subject to other augmented inspections to detect leakage and/or cracking, operational experience demonstrates that leakage throughout the RPV is unlikely. Consequently, conducting the VT-2 visual examinations after cooldown and depressurization would not substantially decrease the level of quality and safety. After reviewing the licensee's proposed alternative, the staff finds that the alternative examination is sufficient to detect evidence of RPV leakage in the vessel bottom head area.

3.2.2 <u>Class 1 and 2 Reactor Support Piping in Covered Trenches</u>

The NRC staff found that requiring the licensee to comply with IWB-5221(a) 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. The basis for the hardship is as follows: The portions of Class 1 and 2 pipes under consideration are located in the reactor vessel bottom head area in trenches that are required to be covered and secured during normal operation to ensure containment sump recirculation flow paths are maintained. Extreme heat and the nearly impossible pathway to the reactor vessel bottom area during normal operation create hazardous conditions for personnel that perform the VT-2 visual examinations of these portions of piping. In addition, the requirement that these portions of piping shall be secured and covered in trenches during normal operation makes them inaccessible for the VT-2 examinations according to IWA-5241. Therefore, the NRC staff determined that personnel safety hazards caused from extreme heat and lack of access pathway and access difficulties due to the trench covers requirement constitute a hardship.

In evaluating the licensee's proposed alternative, the NRC staff assessed whether it appeared that the licensee adequately performed the testing of inaccessible portions of Class 1 and 2 pipes in covered trenches for leakage. The NRC staff found that the licensee will open the inaccessible areas following plant cooldown and depressurization for refueling and perform the VT-2 visual examinations of these portions of pipes in covered trenches in accordance with IWA-5241 during every refueling outage. Therefore, the NRC staff determined that the licensee's proposed system leakage test conducted following plant cooldown and depressurization for refueling in conjunction with the IWA-5240 required VT-2 visual examinations is adequate, because any evidence of leakage or boric acid residue will be discernable during examinations.

In addition to the analysis described above, the NRC staff evaluated the safety significance of the alternative system leakage test performed following plant cooldown and depressurization for refueling. The NRC staff notes the Class 1 and 2 pipes under consideration are made of stainless steel materials. Potential degradation mechanism of these pipes can include fatigue and stress corrosion cracking. However, fatigue cracks are known to have relatively slow growth, and field experience has shown that stress corrosion cracking under the conditions associated with the piping under consideration is not expected. It is expected that any significant degradation of the Class 1 and 2 pipes under consideration would be detected by the proposed system leakage test and the VT-2 visual examinations performed following plant cooldown and depressurization for refueling.

The NRC staff notes that if in an unlikely event these inaccessible portions of the Class 1 and 2 pipes developed a through-wall flaw and a leak, the St. Lucie existing reactor coolant leakage detection systems will be able to identify the leakage during normal operation, and the licensee will take appropriate corrective actions in accordance with the plant technical specifications. Therefore, the NRC staff determines that based on the alternative system leakage testing and the required VT-2 visual examinations performed following plant cooldown and depressurization during each refueling outage, it is reasonable to conclude that if significant service-induced degradation occurs, evidence of that degradation will be detected either by the proposed examinations 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 is adequate to provide a reasonable assurance of structural integrity and leak tightness of the inaccessible portions of Class 1 and 2 pipes. Complying with the requirements specified in IWB-5221(a) and IWC-5221 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 RPV bottom head and the inaccessible portions of Class 1 and 2 pipes under consideration, and complying with the specified requirement 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 of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of RR No. 4, Revision 1, at St. Lucie for the fourth 10-year ISI interval, which will end on August 7, 2023.

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: Austin Young
Ali Rezai

Date: May 31, 2015

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.

If you have any questions, please contact the Project Manager, Farideh Saba, at 301-415-1447 or Farideh.Saba@nrc.gov.

Sincerely,

/RA/

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosure:
Safety Evaluation

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