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

September 18, 2015

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

SUBJECT: ST. LUCIE PLANT UNIT NO. 1 – INSERVICE INSPECTION PLAN FOURTH 10-YEAR INTERVAL RELIEF REQUEST NO. 9, REVISION 0 (TAC NO. MF5352)

Dear Mr. Nazar:

By letter dated December 3, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14351A076), Florida Power and Light Company (the licensee) submitted a request for relief to the U.S. Nuclear Regulatory Commission (NRC), proposing an alternative to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI requirements at the St. Lucie Plant Unit No. 1 (SL-1). Relief Request (RR) No. 9 (RR-9) pertains to the system leakage test of the reactor pressure vessel bottom head and the associated Class 1 and 2 piping in the covered trenches at SL-1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 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 (79 FR 65776), dated November 5, 2014, 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) and 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).

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation (SE), that the ASME Code required examinations of the reactor vessel bottom head and associated Class 1 and 2 piping in covered trenches during system leakage testing at normal operating pressure and temperature would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety for SL-1. Therefore, the NRC staff authorizes the use of the proposed alternative in RR-9 at SL-1 from the date of the issuance of the enclosed SE up to the end of the fourth 10-year ISI interval (February 10, 2018).

Pursuant to 10 CFR 50.55a(z), a proposed alternative must be submitted and authorized prior to implementation. The licensee submitted RR-9 on December 3, 2014, which is over 6 years into the fourth 10-year ISI interval, which began on February 11, 2008. The NRC staff concluded that the licensee's request for authorization of the proposed alternative in RR-9 prior to the date it was authorized does not meet the regulatory requirements set forth in 10 CFR 50.55a(z), and the NRC does not have the regulatory authority to authorize the proposed alternative at SL-1 prior to the date of the enclosed safety evaluation (SE) for the fourth 10-year Inservice Inspection interval. Therefore, the proposed alternative in RR-9 is NOT authorized at SL-1 for the fourth 10-year Inservice Inspection interval from February 10, 2008, to the date of the enclosed SE. The matter of non-compliance with the ASME Code requirements in the fourth 10-year interval prior to the date of this SE has been forwarded to NRC Region II for review and action, as appropriate.

The NRC staff also determined that no safety concerns exist regarding the components under consideration during the period between the beginning of the fourth 10-year ISI interval (February 11, 2008) and the date of the enclosed SE, during which this proposed alternative has not been authorized.

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

If you have any questions, please contact Robert L. Gladney at 301-415-1022 or Robert.Gladney@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. 50-335

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 9, REVISON 0, REGARDING

FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated December 3, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14351A076), Florida Power and Light Company (FPL, the licensee), requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Relief Request (RR) No. 9, Revision 0, pertains to the system leakage test of the reactor pressure vessel (RPV) bottom head and the associated Class 1 and 2 piping in the covered trenches at the St. Lucie Plant, Unit 1 (SL-1).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 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.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

By *Federal Register* Notice (79 FR 65776), dated November 5, 2014, 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) and 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).

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.

Enclosure

The licensee must demonstrate that (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.

The code of record for the SL-1 fourth 10-year interval inservice inspection (ISI) program is the 2001 Edition through the 2003 Addenda of Section XI of the ASME Code. The fourth 10-year ISI interval at SL-1 began February 11, 2008, and ends February 10, 2018.

Based on the above, and subject to the following technical evaluation, the U.S. Nuclear Regulatory Commission (NRC) staff finds that regulatory authority exists for the licensee to request, and the NRC to authorize, the alternative requested by the licensee from the date of the issuance of this safety evaluation (SE) up to the end of the fourth 10-year ISI interval (February 10, 2018).

3.0 TECHNICAL EVALUATION

The licensee has proposed an alternative to the ASME Code requirements pursuant to 10 CFR 50.55a(a)(3)(ii) [retitled 10 CFR 50.55a(z)(2)] 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. The information provided by the licensee in support of the alternative to the ASME Code requirements has been evaluated and the bases for disposition are below.

ASME Code Components Affected (as provided):

The components are the ASME Code Class 1 reactor pressure vessel and the associated Class 1 and Class 2 piping in covered trenches rendered inaccessible due to containment building configuration. The licensee provided description of components under consideration in the following table.

ASME Code, Examination Cat	ASME Code Item Nos.	Examination Component Description
B-P	B15.10	Reactor Pressure Vessel (RPV) – Pressure Retaining Boundary Bottom Head Area (carbon steel) Piping – Pressure Retaining Boundary (covered portions only) SI Headers 12"-SI-148 (Stainless Steel), 12"-SI-149 (Stainless Steel), 12"-SI-150 (Stainless Steel), 12"-SI-151 (Stainless Steel) Charging 2-CH-147 (Stainless Steel) Letdown 2-RC-142 (Stainless Steel)
C-H	C7.10	Piping - Pressure Retaining Components (covered portions only) SDC Suction 10"-SI-420 (Stainless Steel) 10"-SI-422 (Stainless Steel)

ASME Code Requirement:

The 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 pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary.

The ASME Code, Section XI, IWC-2500, Table IWC-2500-1, Examination Category C-H, requires 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.

As stated by the licensee:

> The requirements for performing visual examinations in conjunction with the pressure testing of Class 1 and 2 components are provided in ASME Code, Section XI, Articles IWA-5000, IWB-5000, and IWC-5000.
>
> Paragraph IWA-5241(b) states the following:
>
> > 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-5242(c) states the following:
>
> > When examining insulated components, the examination of the 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 following:

> 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).

Licensee's Proposed Alternative (as stated):

The licensee proposed an alternative examination, as stated below:

> Pursuant to 10 CFR50.55a(a)(3)(ii) [retitled 10 CFR 50.55a(z)(2)], FPL requests approval to perform the examination of the reactor vessel bottom head area and piping in covered trenches at different plant conditions than those required by the ASME Code. FPL will continue to perform the required system pressure tests as prescribed by IWB-5000 each refueling outage and IWC-5000 each period, and will examine all accessible components in accordance with IWA-5241.
>
> For those portions of components rendered inaccessible by containment building configuration, as an alternative to the requirements of IWA-5241, IWB-5221(a), and IWC-5221, FPL has been and will continue to 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. Any evidence of leakage will be evaluated in accordance with IWA-5250, which may include additional inspections and insulation removal as deemed necessary.

Licensee's Basis for Use of the Alternative (as stated):

The licensee's basis for proposing the alternative examination is stated below:

> The objective of the required [VT-2] visual examination at normal operating conditions is to detect evidence of leakage and thereby verify the integrity of the reactor coolant system (RCS) pressure boundary. FPL believes the same evidence of leakage can be identified by [VT-2] visual examination following cooldown for refueling.
>
> The St. Lucie reactor has no bottom head penetrations, and the vessel welds have been volumetrically examined in accordance with [Section XI of the ASME

> Code] with no relevant indications identified. There is no expectation of leakage due to the solid configuration of the bottom of the reactor pressure vessel. In addition, the reactor cavity is monitored for leakage continuously during operation, and inventory balance is performed daily throughout the operating cycle.

The licensee also stated the following:

> There is no plant-specific, NextEra fleet, or industry operating experience regarding potential degradation specific to the items included in this relief request. However, isolated occurrences of stress corrosion cracking have occurred in stainless steel materials in the industry. To address the concerns of these isolated cases, the periodic inspections made possible by removal of the access limitations provides assurance that any isolated degradation would be identified at the onset before a safety concern could develop.

The licensee indicated that the primary method for quantifying and characterizing RCS identified and unidentified leakage is through the use of a reactor coolant water inventory balance. The licensee stated, as reflected below, the leakage detection capabilities at the plant to monitor and identify leakage in an unlikely event of a through wall leak in the components under consideration. As part of the discussion, the licensee defined Action levels 1, 2, and 3 on the absolute value of unidentified RCS inventory balance.

> St. Lucie Unit 1 RCS inventory balance procedure ensures that RCS leakage is within Technical Specification 3.4.6.2. The procedure also provides early detection of negative trends based on statistical analysis. The inventory balance leak rate calculation is performed more frequently (at a 24 hour rather than 72 hour interval) than required by Technical Specification 4.4.6.2.c.

The licensee further stated:

> These action levels trigger condition report initiation, various investigations of leakage up to and including containment entry to identify the source of the leakage.
>
> RCS leak detection at St. Lucie Unit 1 is also provided by three separate monitoring systems: 1) reactor cavity (containment) sump inlet flow monitoring system; 2) containment atmosphere radiation gas monitoring system; 3) and containment atmosphere radiation particulate monitoring system. These systems have high level and alert status alarms in the control room. These systems also have Technical Specification required monitoring (TS 4.4.6.2 a. & b.) at least once every 12 hours. The sensitivity of the containment atmosphere radiation monitoring system depends on the amount of radioactivity in the primary coolant system which is dependent on the percentage of failed fuel. Calculation results conclude that the containment atmosphere radiation monitors are capable of detecting a change of 1 gpm in the leak rate within one hour using design basis reactor water activity assuming 0.1% failed fuel.

> The containment sump alarm response is also highly variable based on the location of the leak, how much vapor condenses and where it condenses. All drains entering the sump are routed first to a measurement tank. When the water level corresponding to 1 gpm or more into the tank is reached, a sump level alarm is actuated in the control room. The combination of Technical Specification required inventory balance, reactor cavity sump monitoring, gas and particulate monitoring systems provide diverse measurement means for acceptable monitoring of RCS leakage.

Licensee's Basis for Hardship (as stated):

The licensee stated the following regarding its reason for requesting the proposed alternative.

> St. Lucie Plant does not have access for [the VT-2 visual examination] of the reactor vessel bottom area during the ASME Code, Section XI, system leakage test of the reactor vessel. There are three possible pathways that lead to the 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 to approximately one foot high and six to eight feet long. Ambient conditions during VT-2 visual 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 increase in the level of quality and safety gained by performing a visual inspection at normal operating conditions does not compensate for the safety hazard the inspector would be subjected to.
>
> Some segments of the 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 flowpaths are maintained. This is outlined in the St. Lucie response to NRC Bulletin 2003-01 (FPL Letter L-2003-201) [ML032240419]. The trench covers prohibit direct VT-2 visual examination of horizontal insulation joints and low points as directed by IWA-5242(b). However, 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. Areas to which leakage may be channeled are also open in many locations throughout the containment for observation during the system leakage test. This is in compliance with the requirements of IWA-5242(c).

Duration of Proposed Alternative

The licensee requested authorization to implement the proposed alternative during the fourth 10-year ISI interval at SL-1, which began on February 11, 2008 and ends on February 10, 2018.

NRC Staff Evaluation:

The NRC staff has reviewed the information concerning the ISI program RR-9, Revision 0, for SL-1 pertaining to the system leakage test and the associated VT-2 visual examinations of the bottom of the reactor vessel and the associated Class 1 and 2 piping in covered trenches at different plant conditions than required by the ASME Code. The ASME Code required the system leakage test to be conducted at normal operating pressure and temperature. These conditions create an extreme heat stress environment that, when combined with a nearly impossible exit pathway, make examination of this area excessively hazardous. Thus, imposition of the examination requirements would cause a hardship or unusual difficulty on the licensee.

The licensee proposed, as an alternative, to perform the required VT-2 visual examination for evidence of leakage and boric acid corrosion during each refueling outage, following plant cooldown and depressurization, instead of at normal operating pressure and temperature. Any evidence of leakage and boric acid corrosion that occurred during the previous fuel cycle can be detected by VT-2 visual examination of this area at the end of the cycle during the refueling outage. In addition, the RCS temperatures will be substantially lower under the vessel area during the refueling outage, which removes the hazardous temperature conditions. The NRC staff determined that the VT-2 visual examination for evidence of leakage and boric acid corrosion conducted during each refueling outage would provide reasonable assurance that leaks through the bottom of the vessel and associated Class 1 and 2 piping in covered trenches that occurred during the previous cycle would be detected. Additionally, leakage and boric acid corrosion of the vessel bottom head and piping would result in the formation of boric acid crystals, which can be detected by the proposed VT-2 visual examination during each refueling cycle. This examination is sufficient to inspect the condition of the external surface of the vessel bottom head area and piping in covered trenches.

The NRC staff, therefore, has determined that the ASME Code required examinations of the reactor vessel bottom head and associated Class 1 and 2 piping in covered trenches during system leakage test at normal operating pressure and temperature would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety for SL-1.

The NRC staff evaluated the licensee's requested duration of the proposed alternative against the regulatory requirements in 10 CFR 50.55a(z). The licensee requested authorization to implement the proposed alternative at SL-1 for the fourth 10-year ISI interval, which began February 11, 2008 and ends February 10, 2018. The licensee also stated that the proposed alternative has been previously implemented during the fourth 10-year ISI interval. Pursuant to 10 CFR 50.55a(z), a proposed alternative must be submitted and authorized prior to implementation. Section 50.55a(z) of 10 CFR does not provide a basis for NRC authorization of alternatives to ASME Code requirements for the inspections that were implemented in the fourth

10-year Interval prior to the date of this SE. As noted in the "NRC's Analysis of Public Comments" (ADAMS ML110280240) for the June 21, 2011 revisions to 10 CFR 50.55a "Implementation is considered to occur at the time the licensee needs to rely on the alternative to satisfy ASME code requirements." Therefore, the NRC staff concluded that the request to authorize the proposed alternative in RR-9 at SL-1 from the date of the issuance of this SE until the end of the fourth 10-year ISI interval (February 10, 2018) meets the regulatory requirements set forth in 10 CFR 50.55a(z). However, the NRC staff concluded that the licensee's request for authorization of the proposed alternative in RR-9 prior to the date of this SE does not meet the regulatory requirements set forth in 10 CFR 50.55a(z), and the NRC does not have the regulatory authority to authorize the proposed alternative at SL-1 from February 10, 2008, to the date of issuance of this SE.

The NRC staff also determined that no safety concerns exist regarding the components under consideration during the period between the beginning of the fourth 10-year ISI interval (February 11, 2008) and the date of this SE, during which this proposed alternative has not been authorized.

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the RPV bottom head and the associated piping. The NRC staff finds that complying with the specified ASME Code 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 the proposed alternative in RR-9 from the date of issuance of this SE up to the end of the fourth 10-year ISI interval (February 10, 2018).

However, the proposed alternative is NOT authorized at SL-1 for the fourth 10-year ISI interval prior to the date of issuance of this SE. The matter of non-compliance with the ASME Code requirements in the fourth 10-year ISI interval prior to the date of this SE has been forwarded to NRC Region II for review and action, as appropriate.

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

Principal Contributors: T. McLellan
A. Rezai

Date: September 18, 2015

the NRC does not have the regulatory authority to authorize the proposed alternative at SL-1 prior to the date of the enclosed safety evaluation (SE) for the fourth 10-year Inservice Inspection interval. Therefore, the proposed alternative in RR-9 is NOT authorized at SL-1 for the fourth 10-year Inservice Inspection interval from February 10, 2008, to the date of the enclosed SE. The matter of non-compliance with the ASME Code requirements in the fourth 10-year interval prior to the date of this SE has been forwarded to NRC Region II for review and action, as appropriate.

The NRC staff also determined that no safety concerns exist regarding the components under consideration during the period between the beginning of the fourth 10-year ISI interval (February 11, 2008) and the date of the enclosed SE, during which this proposed alternative has not been authorized.

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

If you have any questions, please contact Robert L. Gladney at 301-415-1022 or Robert.Gladney@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. 50-335

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Safety Evaluation

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