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

June 8, 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 INSERVICE INSPECTION INTERVAL RELIEF REQUEST NO. 2, REVISION 0 (TAC NO. MF4538)

Dear Mr. Nazar:

By letter dated August 1, 2014, as supplemented by letter dated March 5, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14224A010 and ML15076A193, respectively), Florida Power and Light (FPL, the licensee) submitted Relief Request No. 2 (RR-2) to the U.S. Nuclear Regulatory Commission (NRC), proposing an alternative to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI requirements at the St. Lucie Plant, Unit No. 2 (SL-2).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(ii), the licensee requested to use an alternative on the basis that compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The relief request pertains to the repair and replacement of the Alloy 600 small bore nozzles on the hot leg reactor coolant system piping and the pressurizer, and the Alloy 600 heater sleeves on the pressurizer.

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, relief requests that were previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1), and relief requests that were 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, that FPL has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC authorizes the use of RR-2 at SL-2 for the duration of the fourth 10-year inservice inspection interval, scheduled to end on August 7, 2023.

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

Please direct any inquiries to Allison Dietrich at 301-415-2846 or Allison.Dietrich@nrc.gov.

Sincerely,

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

RELIEF REQUEST NO. 2, REVISON 0

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated August 1, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14224A010), as supplemented by letter dated March 5, 2015 (ADAMS Accession No. ML15076A193), Florida Power and Light Company (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) at the St. Lucie Plant, Unit No. 2 (SL-2).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(2), the licensee proposed alternative repair techniques for Alloy 600 small bore nozzles and heater sleeves on the hot leg piping and the pressurizer, 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

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, relief requests 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 relief requests 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(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 the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of the 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

Enclosure

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 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.

3.0 TECHNICAL EVALUATION

3.1 Component Affected

The components affected are ASME Code Class 1 pressure boundary piping and the pressurizer vessel. They include small bore Alloy 600 nozzles that are typically welded with Alloy 82/182 weld metals to the interior surface of the hot leg reactor coolant system (RCS) piping and the pressurizer, and small bore Alloy 600 heater sleeves that are typically welded with Alloy 82/182 weld metals to the interior surface of the pressurizer during original fabrication.

In Table 1 of the attachment to the March 5, 2015, letter, the licensee provided an updated list of the affected components with information detailing the date of repair, method of repair, materials used, and reason for repair.

3.2 Applicable Code Edition and Addenda

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

3.3 Duration of Relief Request

The licensee submitted RR-2 for the fourth 10-year ISI interval which commenced on August 8, 2013, and will end on August 7, 2023.

3.4 ASME Code Requirement

The ASME Code, Section XI, IWB-3132.2, "Acceptance by Repair/Replacement Activity," states that a component whose volumetric or surface examination reveals flaws that exceed the acceptance standards of Table IWB-3410-1 is unacceptable for continued service, until the additional examination requirements of IWB-2430 are satisfied and the component is corrected by repair or replacement to the extent necessary to meet the standards of IWB-3000.

3.5 Background

This request is related to several prior relief requests dated January 8, 2003; November 21, 2003; and January 4, 2006 (ADAMS Accession Nos. ML030100006, ML033290288, and ML060090296, respectively), which are related to the degradation of Alloy 600/82/182 components and associated repairs. In these requests, the licensee proposed alternative repairs ("half nozzle" and "sleeve" repairs) to the IWB-3132.2 requirement. The NRC evaluated these requests and authorized the licensee's proposed alternatives for the previous 10-year ISI interval on May 26, 2006 (ADAMS Accession No. ML061290056).

The licensee stated that industry experience has shown that Alloy 600 nozzle and heater sleeve base metals and Alloy 82/182 attachment weld metals are susceptible to PWSCC. The Alloy 690 base materials and Alloy 52/152 weld metals are resistant to PWSCC.

The licensee stated that the NRC-approved topical report (TR) WCAP-15973-NP-A, Revision 0, "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Programs" (ADAMS Accession No. ML050700431), accompanied by the licensee's response to conditions the NRC staff placed for use of the TR, provides the technical basis for RR-2. The licensee performed plant-specific engineering evaluations and discussed its findings to each of the NRC staff's conditions delineated in Sections 4.1, 4.2, and 4.3 of the NRC staff's safety evaluation (SE) of the TR.

The licensee stated that in the "half nozzle" technique, the Alloy 600 nozzle or heater sleeve is cut outboard of the Alloy 82/182 partial penetration weld between the nozzle and the pipe wall, between the nozzle and the pressurizer wall, or between the heater sleeve and the pressurizer wall approximately midwall. The cut sections of the Alloy 600 nozzle or heater sleeve are replaced with a short section ("half nozzle") of Alloy 690 nozzle or heater sleeve, which is then welded to the outside surfaces of the pipe wall or the pressurizer wall with Alloy 52/152 weld metals. With these repair techniques, the new welds on the exterior surface of the piping or the pressurizer become the new pressure boundary. The remainder of the Alloy 600 nozzle or heater sleeve, including Alloy 82/182 partial penetration welds, remains in place without correction.

The licensee stated that in the "sleeve" technique, the entire Alloy 600 nozzle is removed by machining and the bore diameter is slightly enlarged. Subsequently, the Alloy 690 sleeve is inserted into the bore and rolled into place. The end of the sleeve at the interior surface of the pipe or the pressurizer is either roll expanded or welded to the interior surface of the pipe or the pressurizer, essentially eliminating corrosion of the carbon steel by stopping the replenishment of borated solution in contact with the carbon steel. The Alloy 690 nozzle is inserted into the Alloy 690 sleeve and the nozzle and sleeve are welded to the exterior of the pipe or the pressurizer with Alloy 52/152 weld metals. With these repair techniques, the new welds on the exterior surface of the piping or the pressurizer become the pressure boundary.

In a letter dated May 26, 2006 (ADAMS Accession No. ML061290056), the NRC authorized the licensee's proposed repair, replacement, or mitigation of Alloy 600 small bore nozzles on the hot leg piping and the pressurizer using either the "half nozzle" or "sleeve" technique for the third 10-year ISI interval. RR-2 is similar to the previously authorized relief request. Table 1 of the attachment to the March 5, 2015, letter shows an updated list of all previous repairs by "half nozzle" or "sleeve" technique that have left remnants of the original Alloy 600 nozzle and heater sleeve base metals and Alloy 82/182 attachment weld metals in place.

The licensee stated that the remnant Alloy 600/82/182 materials (nozzle and heater sleeve base metal and weld metal) will not receive additional examination. The new pressure boundary welds (Alloy 52/152), located on the exterior surface of the RCS piping and the pressurizer, will be examined in accordance with the applicable requirements of Sections III and XI of the ASME Code. There has not been any indication of degradation detected in these previously repaired nozzles and heater sleeves during ISI.

The licensee stated in its March 5, 2015, letter that one pressurizer nozzle and five hot leg RCS piping nozzles were repaired in 1989 with Alloy 600 nozzles and Alloy 82/182 pad welds. These six nozzles were subsequently replaced by Alloy 690 nozzles and welded with Alloy 52/152 attachment metal welds to the existing Alloy 82/182 pads. The licensee identified these six nozzles with Alloy 82/182 pads in Table 1 of March 5, 2015, letter. These six repairs have been receiving a bare metal visual examination each refueling outage in accordance with the SL-2 RCS leak test procedures and ASME Code Case N-722-1, "Additional Examinations for PWR [Pressurized-Water Reactor] Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials, Section XI, Division 1." There has not been any indication of degradation detected to date in these six repairs.

3.6 Proposed Alternative

The licensee proposed an alternative to IWB-3132.2. The licensee's proposed alternative is to use either the "half-nozzle" or the "sleeve" technique to repair, replace, or mitigate cracked or uncracked Alloy 600 small bore nozzles and heater sleeves with components that are made of Alloy 690 material.

3.7 Basis for Use

In its request, the licensee indicated that the basis for its request is consistent with the TR. The licensee noted that this report provides a basis for the evaluation and continued use of "half nozzle" and "sleeve" repairs. The licensee further noted that when using the TR, licensees are required to perform plant-specific analyses. These analyses include general corrosion assessment, thermal fatigue crack growth assessment, and stress corrosion crack growth assessment. A brief description of these analyses follows.

For general corrosion assessment, licensees are required to:

- Calculate the minimum acceptable wall thinning thickness for the ferritic vessel or piping that will adjoin to the mechanical nozzle seal assembly (MNSA) repair or "half nozzle" repair.

- Calculate the overall general corrosion rate for the ferritic materials based on the calculation methods in the TR, the general corrosion rates listed in the TR for normal operations, startup conditions (including hot standby conditions), cold shutdown conditions, and the respective plant-specific times (in percentage of total plant life) at each of the operating modes.

- Track the time at cold shutdown conditions to determine whether this time does not exceed the assumptions made in the analysis. If these assumptions are exceeded, the licensees shall provide a revised analysis to the NRC and provide a discussion on whether volumetric inspection of the area is required.

- Calculate the amount of general corrosion based thinning for the vessels or piping over the life of the plant as based on the overall general corrosion rate calculated in Step 2 and the thickness of the ferritic vessel or piping that will adjoin to the MNSA repair or "half nozzle" repair.

- Determine whether the vessel or piping is acceptable over the remaining life of the plant by comparing the worst case remaining wall thickness to the minimum acceptable wall thickness for the vessel or pipe.

For thermal fatigue crack growth assessment, licensees are required to:

- Demonstrate that the geometry of the leaking penetration is bounded by the corresponding penetration reported in Calculation Report CN-CI-02-71, Revision 1, "Summary of Fatigue Crack Growth Evaluation Associated with Small Diameter Nozzles in CEOG [Combustion Engineering Owners Group] Plants."

- Demonstrate that the plant-specific pressure and temperature profiles in the pressurizer water space for the limiting curves (cooldown curves) do not exceed the analyzed profiles shown in Figure 6-2(a) of Calculation Report CN-CI-02-71, Revision 01, as stated in Section 3.2.3 of the NRC SE of the TR.

- Demonstrate that the plant-specific Charpy upper shelf energy (USE) data shows a USE value of at least 70 foot pounds (ft-lb) to bound the USE value used in the analysis. If the plant-specific Charpy USE data does not exist and the licensee plans to use Charpy USE data from other plants' pressurizers and hot leg piping, then justification (e.g., based on statistical or lower bound analysis) has to be provided.

For stress corrosion crack growth assessment, licensees are required to:

- Conduct appropriate plant chemistry reviews and demonstrate that a sufficient level of hydrogen overpressure has been implemented for the RCS, and that the contaminant concentrations in the reactor coolant have been typically maintained at levels below 10 parts per billion (ppb) for dissolved oxygen, 150 ppb for halide ions, and 150 ppb for sulfate ions.

- Review their plant-specific RCS coolant chemistry histories over the last two operating cycles for their plants, and confirm that these conditions have been met over the last two operating cycles.

Additional details concerning the evaluations may be found in the NRC staff evaluation portion of this document (Section 3.9).

3.8 Basis for Hardship

The licensee stated that as a result of the PWSCC susceptibility of Alloy 600 base metals and Alloy 82/182 weld metals, cracks have developed in these materials, resulting in leakage of the reactor coolant. The leak path from the flaw through the weld and base materials cannot be determined. To remove PWSCC cracks and susceptible materials entirely, the licensee would have to enter into a confined area to access the internal surface of the hot leg piping or the pressurizer to grind out the entire original Alloy 82/182 attachment weld metals and Alloy 600 nozzle and heater sleeve base metals. These activities would expose involved personnel to a high radiation dose, as well as safety hazards. In an April 23, 2003, letter (ADAMS Accession No. ML031140498), the licensee discussed details of the radiological and occupational safety hazards associated with such repairs, replacements, or mitigations associated with the complete removal of unacceptable flaws.

3.9 NRC Staff Evaluation

In accordance with 10 CFR 50.55a(z)(2), the NRC staff evaluated 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.

In evaluating the licensee's proposed alternative, the NRC staff assessed the safety significance of leaving remnants of flaws and PWSCC-susceptible materials in service after repair, as well as potentially exposing the ferritic portions of the affected piping to borated water as a result of these types of repair techniques. In support of RR-2, the licensee used the methods and analysis delineated in the NRC-approved TR WCAP-15973-NP-A, Revision 0. The TR provides the technical basis for repairs by the "half nozzle" and the MNSA techniques. In its SE of the TR dated January 12, 2005, the NRC staff stated that the TR's methods and analysis are acceptable for use by licensees, provided that the conditions imposed by the NRC staff for use of the TR are met. These conditions are delineated in Sections 4.1, 4.2, and 4.3 of the NRC staff's SE of the TR. In evaluating RR-2, the NRC staff focused on whether the plant-specific engineering evaluations provided by the licensee met the conditions imposed by the NRC staff for use of the TR, as discussed below.

3.9.1 Conditions in Section 4.1, "General Corrosion Assessment"

- Condition 4.1.1 requires the licensee to calculate the minimum acceptable wall thinning thickness for the ferritic vessel or piping that will adjoin to the MNSA repair or "half nozzle" repair.

  Section 2.4 of the TR discusses methods for calculating acceptable wall thinning thickness (i.e., limiting allowable diameter or bore hole size) for ferritic vessel or piping after consideration of material loss during service due to corrosion. The NRC staff finds the licensee's results documented in Tables 2A and 2B of RR-2 for the hot leg piping and pressurizer acceptable because (a) the limiting allowable diameter for the "half nozzle" and "sleeve" repairs is based on bounding Combustion Engineering (CE) designed PWRs that do not exceed the ASME Code limits as documented in the TR, and (b) the licensee's calculated plant-specific maximum bore hole diameter after repair with consideration of material loss due to corrosion does not exceed the limiting allowable diameter.

  Therefore, the NRC staff determines that the licensee satisfied Condition 4.1.1 because the licensee calculated minimum wall thinning thickness and determined that the ASME Code limits were not exceeded.

- Condition 4.1.2 requires the licensee to calculate the overall general corrosion rate for the ferritic materials based on the calculation methods in the TR, the general corrosion rates listed in the TR for normal operations, startup conditions (including hot standby conditions), cold shutdown conditions, and the respective plant-specific times (in percentage of total plant life) at each of the operating modes.

Section 2.3 of the TR discusses the method for calculating the overall corrosion rate for ferritic materials. The method in Section 2.3 of the TR is based on exposure of carbon or low Alloy steels to bulk solutions of boric acid. The licensee's calculated plant-specific overall corrosion rate for ferritic materials is 1.34 mils per year (mpy). For this calculation, the licensee used the plant-specific data from the date the first repair was performed by the "half nozzle" technique (i.e., from January 1, 1995, to February 28, 2014), and the TR specified corrosion rate for ferritic materials at each of the operating modes and temperature conditions. The plant-specific data is the percentage of total plant time spent at each of the temperature conditions. SL-2 spent 90.5 percent of the time at high temperature, 2.0 percent at intermediate temperature, and 7.5 percent at low temperature from January 1, 1995, to February 28, 2014. The TR specified corrosion rate (i.e., corrosion rate for CE plants obtained from laboratory testing data) for ferritic materials for each temperature condition is 0.4 mpy for high temperature conditions, 19.0 mpy for intermediate temperature conditions, and 8.0 mpy for low temperature conditions. Therefore, the NRC staff finds that the licensee's plant-specific overall corrosion rate of 1.34 mpy for ferritic materials is acceptable and bounded by the overall allowable corrosion rate of 1.53 mpy for CE PWR plants.

The NRC staff notes that the "sleeve" repair technique is not discussed in the TR. However, repairs using the "sleeve" technique inherently result in crevice or gap geometry that is comparable or even tighter than the crevice or gap geometry in the "half nozzle" technique. In RR-2, the licensee described the "sleeve" technique as follows: The entire nozzle is removed by machining and the bore diameter is slightly enlarged; an Alloy 690 sleeve is sized and inserted into the nozzle bore to provide initial tight fit with maximum clearance gap of 0.010 inch; and then the sleeve is rolled to plastically deform to provide a tighter metal to metal fit with the nozzle bore.

The NRC staff notes that the corrosion determination methods discussed in Sections 2.3 and 2.5 of the TR for repairs by the "half nozzle" and MNSA techniques are also applicable to the "sleeve" technique, because the geometry and size of the crevice or gap resulting from the "sleeve" technique is comparable or even tighter than the crevice or gap resulting from the "half nozzle" or MNSA technique. The corrosion determination method in Section 2.5 of the TR is based on exposure of carbon or low alloy steels to solutions of boric acid confined in a tight crevice or gap where the volume of the solutions is such that the solutions cannot be replenished or refreshed. Thus, the corrosion rate of ferritic materials in a tight crevice or gap is considerably reduced as compared to the exposure to bulk solution of boric acid, as discussed in Section 2.3 of the TR. As the tight crevice or gap is packed with corrosion products, the steel becomes isolated from borated coolant, and the rate of corrosion is reduced. The licensee's estimate of lifetime diametrical loss or increase in hole diameter for repairs by the "sleeve" technique is 0.025 inch, and is, therefore, bounded by the diametrical loss of the most limiting nozzle (i.e., 0.17 inch as documented in Section 2.4 of the TR).

Therefore, the NRC staff determines that the licensee satisfied Condition 4.1.2, and that the licensee's overall corrosion calculations for the "half nozzle" and "sleeve" repairs are acceptable, because the licensee followed the TR methods, and the overall corrosion rate or increase in hole diameter does not exceed the allowable limit.

- Condition 4.1.3 requires the licensee to track the time at cold shutdown conditions to determine whether this time exceeds the assumptions made in the analysis. If these assumptions are exceeded, the licensee shall provide a revised analysis to the NRC, and provide a discussion on whether volumetric inspection of the area is required.

  The percentage of total time the plant spent at low temperature conditions or cold shutdown was 7.5 percent from January 1, 1995, to February 28, 2014. The NRC staff determines that the total plant time of 7.5 percent at cold shutdown is bounded by the analysis assumption of 10 percent of time, as documented in Section 2.3 of the TR.

  Furthermore, the NRC staff confirms that, hypothetically, if the licensee were to not track the plant operating conditions for the entire fourth 10-year ISI interval (i.e., between August 8, 2013, and August 7, 2023), the total calculated diametrical loss or increase in hole diameter due to exposure of ferritic steels to bulk solutions of boric acid would be 0.146 inch, which is bounded by diametrical loss of most limiting nozzle (i.e., 0.17 inch as documented in Section 2.4 of the TR). The NRC staff also confirms that the lifetime diametrical loss or increase in hole diameter due to exposure of carbon or low alloy steels to solutions of boric acid confined in a tight crevice where the volume of the solution is such that the solution cannot be replenished or refreshed (e.g., crevice from repair by the "sleeve" technique), is 0.025 inch, which is bounded by diametrical loss of most limiting nozzle (i.e., 0.17 inch as documented in Section 2.4 of the TR).

  Therefore, the NRC staff determines that the licensee satisfied Condition 4.1.3, because the licensee has tracked the time the plant has spent at cold shutdown conditions, and the assumptions made in the analysis are not exceeded.

- Condition 4.1.4 requires the licensee to calculate the amount of general corrosion-based thinning for the vessels or piping over the life of the plant, as based on the overall general corrosion rate calculated in Step 2 and the thickness of the ferritic vessel or piping that will adjoin to the MNSA repair or "half nozzle" repair.

  For the first "half nozzle" repair, the amount of diametrical loss is calculated to be 131.3 mils or 0.1313 inch for the period from 1994 to the plant license renewal expiration date of April 6, 2043. The diametrical loss of all other repairs performed by the "half nozzle" technique, as shown in Table 1 of RR-2, is bounded by 131.3 mils, or 0.1313 inch since they will have fewer years of service; thus, the material loss will be less. The NRC staff finds the licensee's calculation acceptable, because the licensee used the overall general corrosion rate in Condition 4.1.2 to determine the lifetime diametrical loss of the "half nozzle" repairs.

  Therefore, the NRC staff determines that the licensee satisfied Condition 4.1.4 for the "half nozzle" repairs, because the licensee calculated the amount of general corrosion-based thinning for the vessels or piping over the life of the plant, and the limiting allowable diameter for "half nozzle" repairs in Table 2A of RR-2 is not exceeded.

  The first "sleeve" repair is expected to have 54 years of service from 1989 to the plant license renewal expiration date of April 6, 2043. According to Section 2.5 of the TR, the lifetime diametrical loss of a "sleeve" repair is 25 mils, or 0.025 inch. Therefore, the NRC staff finds the licensee's calculation acceptable, because the licensee used

Section 2.5 of the TR, and the diametrical loss of all "sleeve" repairs in Table 1 of its March 5, 2015, letter is bounded by 25 mils, or 0.025 inch.

Therefore, the NRC staff determines that the licensee satisfied Condition 4.1.4 for the "sleeve" repairs, because the licensee calculated the amount of general corrosion-based thinning for the vessels or piping over the life of the plant, and the limiting allowable diameters for "sleeve" repairs in Table 2B of RR-2 are not exceeded.

- Condition 4.1.5 requires the licensee to determine whether the vessel or piping is acceptable over the remaining life of the plant by comparing the worst case remaining wall thickness to the minimum acceptable wall thickness for the vessel or pipe.

  From a review of Table 2A of RR-2, the NRC staff finds that the "half nozzle" repairs are acceptable for the remaining life of the plant, because the computed worst case diameter (i.e., described in Table 2A as the repair bore diameter after 49 years) is less than the limiting allowable diameter. Therefore, the NRC staff determines that the licensee satisfied Condition 4.1.5 for the "half nozzle" repairs.

  From review of Table 2B of RR-2, the NRC staff finds that the "sleeve" repairs are acceptable for the remaining life of the plant, because the computed worst case diameter (i.e., described in Table 2B as the repair bore diameter after 54 years) is less than the limiting allowable diameter. Therefore, the NRC staff determines that the licensee satisfied Condition 4.1.5 for the "sleeve" repairs.

Therefore, the NRC staff determines that as a result of meeting all conditions of Section 4.1, there is reasonable assurance that the potential of general corrosion of ferritic materials in the crevice region inherent to "half nozzle" or "sleeve" repair is very low.

3.9.2 Conditions in Section 4.2, "Thermal Fatigue Crack Growth Assessment"

- Condition 4.2.1 requires the licensee to demonstrate that the geometry of the leaking penetration is bounded by the corresponding penetration reported in Calculation Report CN-CI-02-71, Revision 1, "Summary of Fatigue Crack Growth Evaluation Associated with Small Diameter Nozzles in CEOG Plants."

  The licensee stated that the geometry of small diameter penetrations under consideration is bounded by the geometry of corresponding small diameter penetrations used in Calculation Report CN-CI-02-71, Revision 1. From review of the NRC SE report documented in NUREG-1779, "Safety Evaluation Report Related to the License Renewal of St. Lucie Nuclear Plant, Units 1 and 2," and letters dated August 25, 2005; April 12, 2006; and April 23, 2003 (ADAMS Accession Nos. ML052420490, ML061040289, and ML031140498, respectively), the NRC staff finds that the geometry of the SL-2 small diameter penetrations listed in Table 1 of its March 5, 2015, letter is bounded by the geometry of corresponding small diameter penetrations in Calculation Report CN-CI-02-71, Revision 1.

  Therefore, the NRC staff determines that the licensee satisfied Condition 4.2.1, because the geometry of the penetration listed in Table 1 of its March 5, 2015, letter is bounded by the corresponding geometry of penetration reported in Calculation Report CN-CI-02-71.

- Condition 4.2.2 requires the licensee to demonstrate that the plant-specific pressure and temperature profiles in the pressurizer water space for the limiting curves (cooldown curves) do not exceed the analyzed profiles shown in Figure 6-2(a) of Calculation Report CN-CI-02-71, Revision 01, as stated in Section 3.2.3 of the NRC SE of the TR.

  The NRC staff notes that the licensee has a plant procedure that administratively controls cooldown of the pressurizer water space. By its plant procedure, the licensee has limited the cooldown rate of pressurizer water temperature to a maximum rate of 75 degrees Fahrenheit (°F) per hour for normal operation. In addition, the licensee has recorded the pressurizer water temperature during cooldown until the water temperature is less than 120 °F. The NRC staff finds that the plant-specific pressurizer water pressure and temperature profiles meet the analyzed profiles documented in Figure 6-2(a) of Calculation Report CN-CI-02-71.

  Therefore, the NRC staff determines that the licensee satisfied Condition 4.2.2, because the SL-2 pressure and temperature profiles in the pressurizer water space for the limiting curves (cooldown curves) did not exceed the analyzed profiles in Figure 6-2(a) of Calculation Report CN-CI-02-71

- Condition 4.2.3 requires the licensee to demonstrate that the plant-specific Charpy USE data shows a USE value of at least 70 ft-lb to bound the USE value used in the analysis. If the plant-specific Charpy USE data does not exist, and the licensee plans to use Charpy USE data from other plants' pressurizers and hot leg piping, then justification (e.g., based on statistical or lower bound analysis) has to be provided.

  From a review of the Charpy test data documented in Table 3 of its March 5, 2015, letter, the NRC staff notes that the licensee used ten plates (SA-533 Grade B, Class 1, Alloy steel), including two pressurizer lower shell plates, one pressurizer upper head plate, one pressurizer bottom head plate, and six reactor vessel plates, to demonstrate plant-specific Charpy USE value of at least 70 ft-lb for pressurizer shell. The pressurizer vessel plates and the reactor vessel plates are all expected to be very similar in fabrication and chemistry, and have received similar heat treatment. The licensee used two plates (SA-516 Grade 70 carbon steel) to demonstrate plant-specific Charpy USE value of at least 70 ft-lb for hot leg piping. The Charpy USE value is the absorbed energy at 100-percent shear. The 100-percent shear state is obtained by performing Charpy tests at progressively higher temperatures. As the testing temperature increases, the absorbed energy increases, as well as the percent shear. The NRC staff confirms that with increasing test temperature, the above plates are expected to exhibit a USE value of at least 70 ft-lb at 100-percent shear.

  Therefore, the NRC staff determines that the licensee satisfied Condition 4.2.3 because the licensee demonstrated that a USE value of at least 70 ft-lb is reached and bounded by the analysis USE value.

Therefore, the NRC staff determines that as a result of meeting all conditions of Section 4.2, a reasonable assurance exists that the potential for crack growth by thermal fatigue is very low.

3.9.3 Conditions in Section 4.3, "Stress Corrosion Crack Growth Assessment"

- Condition 4.3.1 requires the licensee to conduct appropriate plant chemistry reviews and demonstrate that a sufficient level of hydrogen overpressure has been implemented for the RCS, and that the contaminant concentrations in the reactor coolant have been typically maintained at levels below 10 ppb for dissolved oxygen, 150 ppb for halide ions, and 150 ppb for sulfate ions.

  The licensee has reviewed the SL-2 RCS chemistry data from 2005 to 2014 and determined that:

  - The hydrogen concentration has never been below 15 cubic centimeters per kilogram (cc/kg) during periods of time in which the reactor has been critical, and has typically been maintained between 25 cc/kg and 50 cc/kg;

  - The dissolved oxygen has never exceeded 10 ppb, and has typically been maintained at less than 5 ppb; and

  - The concentration of fluoride, chloride, and sulfate have never exceeded 150 ppb, and has typically been maintained at less than 5 ppb.

  Therefore, the NRC staff determines that the licensee satisfied Condition 4.2.3, because the licensee has conducted appropriate plant chemistry reviews, and the results showed that the contaminant concentrations in the reactor coolant were maintained at the required levels (i.e., the dissolved oxygen below 10 ppb, halide ions below 150 ppb, sulfate ions below 150 ppb, and hydrogen between 25 cc/kg and 50 cc/kg).

- Condition 4.3.2 requires that during the outage in which the "half nozzle" repair is scheduled to be implemented, licensees adopting the TR's stress corrosion crack growth arguments will need to review their plant-specific RCS coolant chemistry histories over the last two operating cycles for their plants and confirm that these conditions have been met over the last two operating cycles.

  The NRC staff determines that the licensee satisfied Condition 4.3.2, because the licensee reviewed the RCS coolant chemistry history from the last two operating cycles and determined that the contaminant concentrations met the required limits and there were no water chemistry transients during the last two operating cycles.

Therefore, the NRC staff determines that as a result of meeting all conditions of Section 4.3, a reasonable assurance exists that the potential for crack growth by the stress corrosion cracking is very low.

In summary, the NRC staff determines that the licensee's proposed alternatives in RR-2 are acceptable, because the plant-specific engineering evaluations provided by the licensee satisfactorily meet the conditions imposed by the NRC staff for use of the TR. The NRC staff also determines that there is a reasonable assurance that the remnants of original nozzles and satisfactorily meet the conditions imposed by the NRC staff for use of the TR. The NRC staff also determines that there is a reasonable assurance that the remnants of original nozzles and heater sleeves, including the flaws and PWSCC materials left in service, will not impact the structural integrity of the primary pressure boundary.

3.9.4 Evaluation of Hardship

The NRC staff found that requiring the licensee to comply with the ASME Code (i.e., IWB-3132.2 and IWA-4000 of Section XI to remove entirely the detected leaking PWSCC flaws and susceptible materials, or mitigate preemptively nonleaking PWSCC flaws and susceptible materials, under consideration) would result in hardship. The nozzles, heater sleeves, and partial penetration J-groove attachment welds under consideration are made of Alloy 600/82/182 materials susceptible to PWSCC. To conduct the ASME Code repair and replacement activities on these nozzles and heater sleeves would require removing the cracks and the PWSCC-susceptible materials entirely. This would necessitate that the licensee work in a confined and high radiological area associated with the hot leg RCS piping and the pressurizer, which would expose the involved personnel to high radiation doses, as well as safety hazards. The radiation dose rates and descriptions of the safety hazards associated with the repair and replacement activities are documented in the licensee's April 23, 2003, letter (ADAMS Accession No. ML031140498). Based on the provided dose rates and descriptions of safety hazards, the NRC staff determines that complying with the specified requirements of 10 CFR 50.55a(g) constitutes hardship.

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 subject small bore nozzles and heater sleeves, and that 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, RR-2 is authorized for use pursuant to 10 CFR 50.55a(z)(2) for SL-2 for the duration of the fourth 10-year ISI interval, scheduled to end on August 7, 2023.

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

Principal Contributor: A. Rezai

Date: June 8, 2015

Please direct any inquiries to Allison Dietrich at 301-415-2846 or Allison.Dietrich@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

cc w/enclosure: Distribution via Listserv

DISTRIBUTION:

PUBLIC	LPL2-2 r/f	RidsNrrDorlLpl2-2
RidsNrrDeEpnb	RidsACRS_MailCTR	RidsNrrLABClayton
RidsNrrPMStLucie	RidsRgn2MailCenter	MWaters, OEDO
ADietrich, NRR	LRonewicz, NRR	ARezai, NRR

ADAMS Accession No.: ML15131A395 *by e-mail

OFFICE	LPL3-1/PM	LPL2-2/PM	LPL2-2/LAiT	LPL2-2/LA	DE/EPNB/BC*	LPL2-2/BC
NAME	ADietrich	FSaba	LRonewicz	BClayton	DAlley	SHelton
DATE	5/18/15	5/21/15	5/14/15	5/14/15	5/1/15	6/8/15

OFFICIAL RECORD COPY