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

November 1, 2016

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
Florida Power & Light Co.
Mail Stop: NT3/JW
15430 Endeavor Drive
Jupiter, FL 33478

SUBJECT: ST. LUCIE PLANT UNIT NO. 2 – INSERVICE INSPECTION PLAN FOURTH 10-YEAR INTERVAL RELIEF REQUEST NO. 11 (CAC NO. MF7456)

Dear Mr. Nazar:

By letter dated March 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16076A431), Florida Power and Light Company submitted a request for relief to the U.S. Nuclear Regulatory Commission (NRC), proposing an alternative to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized-Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," requirements at St. Lucie Plant Unit No. 2 (St. Lucie 2). The request for relief, Relief Request (RR) No. 11 (RR-11), pertains to examination requirements for reactor vessel closure head (RVCH) nozzles and partial-penetration welds fabricated with primary water stress corrosion cracking-resistant materials at St. Lucie 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.55a(z)(1), the licensee proposed an alternative examination interval on the basis that the alternative provides an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the ASME Code required examinations of the RVCH nozzles and partial-penetration welds in question can be extended to a 20-year interval and still provide an acceptable level of quality and safety. Therefore, the NRC staff authorizes the use of the St. Lucie 2 inspection deferment proposed in RR-11 for the fourth 10-year Inservice Inspection interval, which commenced 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 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 Perry H. Buckberg at 301-415-1383 or Perry.Buckberg@nrc.gov.

Sincerely,

Jeanne A. Dion, Acting 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
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 11 REGARDING

EXAMINATION OF REACTOR CLOSURE VESSEL HEAD PENETRATIONS AND WELDS

ST. LUCIE PLANT UNIT NO. 2

FLORIDA POWER AND LIGHT COMPANY

DOCKET NUMBER 50-335

1.0 INTRODUCTION

By letter dated March 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16076A431), Florida Power & Light Company (FPL, the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV Code) Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized-Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," for St. Lucie Unit No. 2 (St. Lucie 2). The FPL submittal included Dominion Engineering Inc., Technical Note TN-5696-00-02, Revision 0 (ADAMS Accession No. ML16076A432) as Attachment No 3.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the licensee requested relief from the examination frequency of ASME B&PV Code Case N-729-1, on the basis that the alternative examination frequency proposed by Relief Request No. 11 (RR-11) provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

As required by 10 CFR 50.55a(g)(4), throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components that are classified as ASME B&PV Code, Class 1, 2, and 3 components must meet the requirements, except design and access provisions and preservice examination requirements, as set forth in Section XI of the ASME B&PV Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the conditions listed therein. Additionally, pursuant to 10 CFR 50.55a(g)(6)(ii)(D), all licensees of pressurized water reactors must augment their inservice inspection [ISI] program with ASME Code Case N-729-1, subject to conditions specified in paragraphs [10 CFR 50.55](g)(6)(ii)(D)(2) through (6).

Pursuant to 10 CFR 50.55a(z)(1), alternatives to the requirements of paragraphs (b) through (h) of this section may be used, when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. A proposed alternative must be submitted prior to implementation, and the licensee or applicant demonstrates 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.

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 proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

The affected components are ASME B&PV Code Class 1, Reactor Vessel Closure Head (RVCH) Penetration Nozzles fabricated from ASME SB-167, Alloy UNS N06690 (Alloy 690). The penetration nozzle J-groove welds are fabricated from ASME SFA-5.14, filler metal - ERNiCrFe-7 (UNS N06052) and ASME SFA-5.11, welding electrode - ENiCrFe-7 (UNS W86152), also referred as 52/152 weld materials. The examination requirements of these components are delineated in ASME Code Case N-729-1, Table 1, Item B4.40. The original St. Lucie 2 RVCH penetration nozzles were manufactured from ASME SB-167, Alloy UNS N06600 (Alloy 600) and used J-grove welds fabricated from ASME SFA-5.14, filler metal - ERNiCr-3 (UNS N06082) and ASME SFA-5.11, welding electrode – EniCrFe-3 (UNS W86182), also referred to as 82/182 weld materials. The original RVCH was replaced with the new primary water stress corrosion cracking (PWSCC) resistant materials and placed in operation in January 2008.

3.2 Inservice Inspection Interval

The current St. Lucie 2 ISI interval is the fourth 10-year ISI interval, which started on August 8, 2013, and will end on August 7, 2023. The proposed alternative would defer the required examinations, which are currently scheduled for the spring 2017 refueling outage. The licensee proposes to complete the required examination during the spring 2023 refueling outage, prior to the end of the fourth 10-year ISI interval at St. Lucie 2. The ASME B&PV Code of Record for the current ISI interval at St. Lucie 2, is the 2007 Edition with the 2008 Addenda of Section XI.

3.3 ASME Code and Regulatory Requirements

Section 50.55a(g)(6)(ii)(D) of 10 CFR requires, in part, that licensees shall augment their ISI program in accordance with ASME Code Case N-729-1, subject to the conditions specified in paragraphs (2) through (6) of 10 CFR 50.55a(g)(6)(ii)(D). ASME Code Case N-729-1, Table 1, Inspection Item B4.40, requires volumetric or surface examinations be performed within one inspection interval (nominally 10-calendar years) of its inservice date for a replaced RVCH with PWSCC resistant nozzles and weld materials. Therefore, the required volumetric or surface examinations for St. Lucie 2 replacement RVCH nozzles and welds shall be completed by January 2018 in order to fulfill the requirements of ASME Code Case N-729-1. The required examinations are currently scheduled for the spring 2017 refueling outage.

3.4 Proposed Alternative

The licensee's proposed alternative would delay the required inspections by a period of approximately 5.5 years. FPL proposes to complete the required inspections in accordance with ASME Code Case N-729-1 during the St. Lucie 2 refueling outage scheduled for spring 2023. The proposed inspections will occur during the facility's fourth 10-year ISI interval, which is currently scheduled to end on August 7, 2023.

3.5 Licensee's Basis for Use of the Proposed Alternative

The licensee's basis for use of the proposed alternative relied on multiple topics of consideration including the concept that the specified inspection interval used in ASME Code Case N-729-1, is primarily based, in part, on Alloy 600/82/182 corrosion data and was intended to be conservative and subject to reassessment when additional laboratory and actual plant operating experience became available for PWSCC resistant materials (i.e., Alloy 690/52/152). Another topic of consideration addressed previous examinations performed on the St. Lucie 2 replacement RVCH, and available service data from operating plants with similar replacement heads, where volumetric and surface examinations have satisfactorily been completed in accordance with ASME Code Case N-729-1, with no recordable indications of PWSCC. An additional topic of consideration addressed supplemental materials and fabrication requirements used during the fabrication of the St. Lucie 2 replacement RVCH, which further increased its resistance to PWSCC.

In addressing the basis for use of the proposed alternative, the licensee asserts that the inspection intervals contained in ASME Code Case N-729-1 for heads with Alloy 690/52/152 materials are based in part on data from Alloy 600/82/182 materials. The volumetric or surface inspection interval of RVCH penetrations fabricated from PWSCC resistant materials was conservatively established as once each inspection interval (10-calendar years). This inspection interval was partly based on PWSCC crack growth rates and data contained in "Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (MRP-55)," dated July 18, 2002, and "Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115)," dated November 2004 (both documents are available at www.epri.com).

The licensee used the parameters defined by ASME Code Case N-729-1 to calculate the factor of improvement (FOI) for its replacement RVCH with Alloy 690/52/152 materials needed to support extending the inspection interval to 15.5-calendar years (FOI value of 7.35). The licensee stated that the PWSCC crack growth rates for Alloy 690/52/152 materials are significantly lower than those of Alloy 600/82/182 materials, therefore, they merit a much longer inspection interval than required by ASME Code Case N-729-1. The licensee bases this assertion on laboratory test data for Alloy 690/52/152 data presented in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," dated February 2014 (ADAMS Accession No. ML14283A046). The licensee asserted that this FOI value of 7.35, is supported by its analyses of Alloy 690/152/52 material crack growth rate data presented in MRP-375, and data from two NRC contractors. Specifically, data from the Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL) summary report, when assessed relative to the crack growth rate behavior of Alloy 600/82/182 materials (ADAMS Accession No. ML14322A587). The licensee further stated that comparing cold worked Alloy 690 material data against non-cold worked Alloy 600 data would add further conservatism to the FOI it is using for its Alloy 690/52/152 materials.

In addressing its second basis for use of the proposed alternative, the licensee stated that preservice volumetric examinations of the replacement RVCH partial penetration nozzles were performed prior to entering service in 2008 with no recordable indication. In addition, bare metal visual examinations were performed in accordance with ASME Code Case N-729-1, Table 1, item B4.30, in 2012. These visual examinations were performed by qualified examiners Visual Testing (VT-2), using on the outer surface of the RVCH including the annulus area of the penetration nozzles. These examinations did not reveal any indications of nozzle leakage (e.g., boric acid deposits) on nozzle surfaces or near nozzle penetrations. The VT 2 examinations and acceptance criteria as required by ASME Code Case N-729-1, Table 1, item B4.30 are not affected by RR-11, the VT-2 examinations will continue to be performed on a frequency of every third refueling outage or every 5-calendar years, whichever is less. Additionally, the licensee highlighted the available service history of replacement RVCH with Alloy 690/52/152 nozzles, which have been successfully examined in accordance with ASME Code Case N-729-1, with no indications of PWSCC.

In addressing its third basis for use of the proposed alternative, the licensee asserted that they imposed supplement requirements on the materials and fabrication procedures used for the St. Lucie 2 replacement RVCH penetration nozzles. Specifically, these supplemental requirements included thermal treatment, prohibition of cold straightening after thermal treatment, and reduced impurities, which resulted in the replacement Alloy 690/52/152 RVCH nozzles being more PWSCC resistant than the standard Alloy 690/52/152 materials used in MRP-375. Since the available data for standard Alloy 690/52/152 is bounded by the FOI of 7.3, these supplemental enhancements provide further conservatism.

In conclusion, FPL stated that its analysis demonstrates that the proposed alternative inspection interval provides a significant nuclear safety improvement, when comparing the current St. Lucie 2 RVCH with the PWSCC resistant materials to an RVCH fabricated with Alloy 600/82/182 materials examined to the current requirements. As such, the licensee found that the proposed one-time revised volumetric/surface examination frequency will provide an acceptable level of quality and safety, and requested to extend the inspection frequency of the RVCH nozzles and associated J-grove welds at St. Lucie 2 from the required 10 years to 15.5 years.

3.6 NRC Staff Evaluation

In evaluating the technical sufficiency of the licensee's proposed one-time extension of the St. Lucie 2 RVCH nozzle and J-grove weld volumetric/surface examination interval to 15.5 years, the NRC staff considered each of the aspects of the licensee's basis for use of the proposed alternative. The NRC staff found that the technical basis included by the licensee, provided sufficient information for the NRC staff to review the proposed alternative.

Due to issues with PWSCC, many domestic and foreign PWR plants have replaced RVCHs with Alloy 600/82/182 nozzles and welds with RVCHs containing PWSCC resistant Alloy 690/52/152 nozzles and welds. The inspection frequencies developed in Code Case N-729-1 for RVCH penetration nozzles using Alloy 690/52/152 were developed based, in part, on crack growth rate equations documented in MRP-55 and MRP-115 for Alloy 600/82/182 materials. The licensee's primary technical basis is that the available crack growth rate data for the new, more crack-resistant materials (i.e., Alloy 690/52/152) justifies a longer inspection interval, and demonstrates a significant FOI versus the older Alloy 600/82/182 materials. Based on its proposed inspection interval of 15.5-calendar year, the licensee determined that it needed an FOI of 7.35. The staff independently verified that the licensee's requested inspection interval of 15.5 years implies an FOI of 7.35, using St. Lucie 2 upper head operating temperature of 602.6 °F.

In evaluating the licensee's first technical basis for use of the proposed alternative, the NRC staff notes FPL's use of MRP-375. This document, in part, summarizes numerous Alloy 690/52/152 crack growth rate data from various sources to develop factors of improvement for the crack growth rate equations provided in MRP-55 and MRP-115. While the NRC staff finds the licensee's assertions and/or interpretations to be reasonable, MRP-375 is not an NRC-approved document. Additionally, NRC staff has not validated all of the data reported in MRP-375. Therefore, the NRC staff does not consider it appropriate to use all of the data from this document to review the licensee's relief request. A more detailed review of the data provided in MRP-375 will be performed by an international group of experts as part of an Alloy 690 Expert Panel. It is expected that this group will complete its review sometime in 2017. In the interim, the NRC staff review relied upon Alloy 690/52/152 crack growth rate data from the PNNL and ANL summary report. The licensee also used the PNNL and ANL data for Alloy 690 test samples with up to 22-percent cold work, which were generally consistent with the overall data presented in MRP-375, and also support the licensee's use of FOI value of 7.35.

The PNNL and ANL data summary report also includes crack growth rate data up to approximately 20 percent cold work based on the observation of local strains in welds and weld dilution zone data. However, the NRC staff did not consider the weld dilution zone data in its assessment. This is because the limited weld dilution zone data that is currently available has shown higher crack growth rates than are commonly observed for Alloy 690/52/152 material. The high crack growth rates in weld dilution zones may be due to the reduced chromium present in these areas. The NRC staff chose to exclude the weld dilution zone data from this analysis due to the limited number of data points available, the variability in the results, and the limited area of continuous weld dilution for potential flaws to grow through. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a J-groove weld along the low alloy steel head interface. It is not fully apparent to the NRC staff how accelerated crack growth in a very small weld dilution zone area would result in a significantly increased probability of leakage or component failure as a result of the requested inspection interval extension. Exclusion of these data may be reevaluated as additional data become available, a better understanding of the existing data is obtained, or if a longer extension of the inspection interval is requested in the future. Therefore, the NRC staff finds that the impact of these weld dilution zone crack growth rates on the change in volumetric inspection frequency, as requested by the licensee's proposed alternative, is not considered to be relevant for this specific relief request, and finds that the licensee's use of an FOI value of 7.35 is justified and bounded by the relevant available data included in the PNNL and ANL data summary report.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds the absence of leak indications found during a bare metal visual examination for leakage through the St. Lucie 2 replacement RVCH nozzle and J-groove welds is a reasonable means to demonstrate the absence of through-wall degradation at these locations prior to the time the examinations were conducted. The NRC staff finds that continued performance of future bare metal visual examinations in accordance with the code case is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Finally, the NRC staff finds that the proposed alternative frequency for the volumetric/surface examinations, in conjunction with the required VT-2 examinations, will provide reasonable assurance of the continued structural integrity of St. Lucie 2 replacement RVCH nozzles and associated J-groove welds, based on the currently available demonstrated service history of replacement RVCH Alloy 690/52/152 nozzles that have been in service for approximately 22 years with no reports of PWSCC.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC staff found that the licensee provided further technical justification to support its proposed alternate examination interval. Specifically, the staff recognizes that using supplemental material and fabrication requirements that included reduced impurities, controlled grain size and microstructure, and limited cold work during fabrication can significantly reduce the susceptibility of nickel alloys to PWSCC. Therefore, the NRC staff found that the FPL analysis provided sufficient technical justification and supports the concept that extending the volumetric/surface inspection interval for St. Lucie 2 replacement RVCH to 15.5-calendar years does not pose a higher risk than that associated with an RVCH with Alloy 600/82/182 nozzles and associated J-groove welds, inspected at intervals as specified in 10 CFR 50.55a(g)(6)(ii)(D). Hence, the NRC staff found the licensee's technical basis to be acceptable.

Therefore, the NRC finds that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(z)(1).

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the alternative method proposed by FPL in RR-11 will provide an acceptable level of quality and safety for the volumetric/surface examination frequency requirements of the St. Lucie 2 RVCH. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the one-time use of RR-11 at St. Lucie 2 for the duration up to, and including the spring 2023 refueling outage that will occur in the fourth 10-year ISI interval.

All other requirements of the ASME B&PV Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) 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 Contributor: Roger Kalikian

Date: November 1, 2016

If you have any questions, please contact Perry H. Buckberg at 301-415-1383 or Perry.Buckberg@nrc.gov.

Sincerely,

/RA/

Jeanne A. Dion, Acting 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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