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

August 31, 2017

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
Florida Power & Light Company
Mail Stop: EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: ST. LUCIE PLANT UNIT NO. 1 – RELIEF FROM THE REQUIREMENTS OF THE ASME CODE REGARDING RELIEF REQUEST 12 FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (CAC NO. MF9273)

Dear Mr. Nazar:

By letter dated February 14, 2017 (Agencywide Documents Access and Management System Accession No. ML17045A357), Florida Power & Light Company (the licensee) submitted Relief Request 12 for the fourth 10-year inservice inspection inverval at St. Lucie Plant Unit No. 1. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the licensee requested that the U.S. Nuclear Regulatory Commission (NRC) authorize a proposed alternative to examination frequency requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 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," for the reactor vessel closure head.

The NRC staff reviewed the submittal and, as set forth in the enclosed safety evaluation, concludes that the licensee's proposed alternative to ASME Section XI, Code Case N-729-1 provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(z)(1), the NRC staff authorizes the licensee's proposed alternative in Relief Request 12 for the remainder of the duration of up to, and including, the Spring 2021 refueling outage in the fifth 10-year inservice inspection interval at St. Lucie Plant Unit No. 1.

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

If you have any questions, please contact Michael Wentzel at 301-415-6459 or Michael.Wentzel@nrc.gov.

Sincerely,

for

Undine Shoop, 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

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 12 REGARDING

EXAMINATION OF WELDS IN CONTROL ELEMENT DRIVE MECHANISM HOUSING

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated February 14, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17045A357), Florida Power & Light Company (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Relief Request 12 pertains to frequency of inservice inspection (ISI) of the reactor pressure vessel (RPV) closure head penetration tubes and vent pipe and their dissimilar metal (DM) attachment welds made of primary water stress corrosion cracking (PWSCC) resistant materials. This request is for the Saint Lucie Plant Unit No.1 (St. Lucie Unit 1).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the licensee proposed an alternative frequency of examination for the RPV closure head penetrations and their DM attachment welds on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), "Inservice inspection standards requirement for operating plants," throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code Class 1, 2, and 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME Code that become effective subsequent to editions specified in paragraphs (g)(2) and (3) of 10 CFR 50.55a, and that are incorporated by reference in paragraph (a)(1)(ii) of 10 CFR 50.55a, to the extent practical, within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(g)(4)(ii), "Applicable ISI Code: Successive 120-month intervals," inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in paragraph (a) of 10 CFR 50.55a 12 months before the start of the 120-month inspection interval (or the optional ASME Code

Cases listed in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1" (ADAMS Accession No. ML13339A689), when using Section XI, that are incorporated by reference in paragraph (a)(3)(ii) of 10 CFR 50.55a, subject to the conditions listed in paragraph (b) of 10 CFR 50.55a. However, a licensee whose ISI interval commences during the 12- through 18-month period after July 21, 2011, may delay the update of its Appendix VIII program by up to 18 months after July 21, 2011.

Pursuant to 10 CFR 50.55a(g)(6)(ii)(D), "Augmented ISI Requirements: Reactor vessel head inspections": "(1) All licensees of pressurized water reactors must augment their inservice inspection program with ASME Code Case N-729-1, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of [10 CFR 50.55a]."

Pursuant to 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, 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 and authorized prior to implementation. The applicant or 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 10 CFR 50.55a 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 <u>TECHNICAL EVALUATION</u>

3.1 <u>Background</u>

By letter dated December 23, 2014 (ADAMS Accession No. ML14339A163), the NRC approved a similar request for St. Lucie Unit 1, authorizing deferral of ISI of the RPV closure head penetrations and their DM attachment welds for 3 calendar years (i.e., from the Spring 2015 refueling outage in the fourth 10-year ISI interval to the March 2018 refueling outage in the fifth 10-year ISI interval). The original RPV closure head was replaced with a PWSCC-resistant head during the refueling outage that ended in December 2005.

3.2 <u>Component Affected</u>

ASME Code Class 1 RPV closure head penetration tubes and vent pipe and their partial penetration (J-groove) DM attachment welds made of PWSCC-resistant materials are affected. In accordance with 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," these DM welds are classified as Item No. B4.40 (Table 1 of Code Case N-729-1).

The licensee stated that the original Alloy 600 RPV closure head at St. Lucie Unit 1 was replaced with an Alloy 690 head during the refueling outage that ended in December 2005. The penetration tubes and vent pipe are made of Alloy 690 and welded to the low alloy steel RPV closure head with Alloy 52/152 weld materials. Alloy 690 base material and Alloy 52/152 weld materials have been known to be resistant to PWSCC. The RPV upper head normal operating temperature at St. Lucie Unit 1 is 602.6 degrees Fahrenheit (°F).

3.3 Applicable Code Edition and Addenda

The code of record for the fourth 10-year ISI interval is the 2001 Edition through 2003 Addenda of the ASME Code.

3.4 Duration of Relief Request

The licensee submitted this request for the remainder of the fourth 10-year ISI interval that began on February 11, 2008, and is scheduled to end on February 10, 2018, up to and including the Spring 2021 (PSL1-30) refueling outage in the fifth 10-year ISI interval that is scheduled to begin on February 11, 2018, and end on February 10, 2028.

The licensee stated that approval of this request permits the ISI of the RPV closure head penetrations and their DM attachment welds be deferred for an additional 3 calendar years from the Spring 2018 refueling outage to the Spring 2021 refueling outage (PSL1-30 refueling outage) in the fifth 10-year ISI interval. The PWSCC-resistant RPV closure head was installed during the refueling outage that ended in December 2005.

3.5 ASME Code Requirement

The regulations at 10 CFR 50.55a(g)(6)(ii)(D) mandate augmented inspection in accordance with ASME Code Case N-729-1 with conditions for the RPV head penetration nozzles and their associated partial penetration DM welds made of PWSCC-resistant materials. ASME Code Case N-729-1, Item No. B4.40 (Table 1), requires that the RPV closure head penetration nozzles and their associated partial penetration DM welds of PWSCC-resistant materials be subjected to volumetric and surface examinations during every 10-year ISI interval (nominally 10 calendar years).

3.6 Proposed Alternative

The licensee proposed an alternative frequency of examination. The proposed alternative is to perform the volumetric and surface examinations of the RPV closure head penetration tubes and vent pipe and their DM attachment welds not later than 15.5 calendar years from the date of installation of the replacement RPV head.

3.7 Basis for Use of Alternative

The licensee's justification for use of the proposed alternative is based primarily on three topics of consideration:

- The first topic addresses the concept that the inspection interval in Code Case N-729-1 is based on PWSCC crack growth rates for Alloy 600/82/182 materials.
- The second topic addresses a bare metal visual examination (VE) of the licensee's replacement RPV closure head performed according to Code Case N-729-1 (Item No. B4.30 in Table 1).
- The third topic addresses a plant-specific factor of improvement (FOI) analysis that the licensee conducted.

In addressing its first basis for use of the proposed alternative, the licensee asserts that the inspection intervals contained in Code Case N-729-1 for Alloy 600/82/182 are based on reinspection years equal to 2.25 and that this value is based on PWSCC crack growth rates as defined in the 75^{th} percentile curve contained in Electric Power Research Institute Materials Reliability Program (MRP)-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," July 18, 2002, and MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," November 2004. The licensee further asserts that PWSCC crack growth rates of Alloy 690/52/152 are significantly lower than those of Alloy 600/82/182, and therefore, merit a longer inspection interval. The licensee bases that assertion on (a) the lack of cracking in other Alloy 690 components such as steam generators and pressurizers in the approximately 20 years that Alloy 690 has been in service in these components; (b) the failure to observe cracking in inspections already performed in replacement heads (16 of 40 replacement heads have been examined, which include heads that operate at higher temperatures than the head under consideration); (c) the similarity of the inspected heads to the head under consideration regarding configuration, manufacturing, design, and operating conditions; and (d) laboratory test data for Alloy 690/52/152 as contained in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," February 2014.

In addressing its second basis for use of the proposed alternative, the licensee stated that the bare metal visual VE examination performed according to Code Case N-729-1 on the St. Lucie Unit 1 replacement RPV closure head in the 2010 (first ISI VE) and 2015 (second ISI VE) refueling outages showed no indication of leakage. The licensee will continue to perform a bare metal visual VE examination on the St. Lucie Unit 1 replacement RPV closure head in accordance with all requirements specified in Code Case N-729-1 as mandated by 10 CFR 50.55a(g)(6)(ii)(D) with conditions.

Furthermore, the licensee stated that the results from the preservice volumetric examinations performed on St. Lucie Unit 1 replacement head penetration nozzles and their J-groove attachment welds in 2005 showed no detectable defects.

In addressing its third basis for use of the proposed alternative, the licensee performed a plant-specific calculation of the required FOI in the crack growth rate of Alloy 690/52/152, as compared to the crack growth rate of Alloy 600/82/182. In this plant-specific FOI calculation, the licensee used the actual temperature of the upper head at St. Lucie Unit 1 and conservatively assumed that calendar years were equal to effective full power years. Based on this calculation, and as documented in the submittal, the licensee determined that an FOI of 7.35 was required to meet the proposed and desired inspection interval of 15.5 calendar years. The licensee then proposed that because the required FOI of 7.35 was smaller than the FOI of 20, which bounded most of the MRP-375 data for Alloy 690/52/152, the use of an FOI of 7.35 would not result in a reduction in safety, and therefore, was justified.

The licensee stated that its analysis showed significant margin to ensure the potential for PWSCC in Alloy 690 nozzles and the Alloy 52/152 attachment welds is remote. As such, the licensee found the technical basis sufficient to provide reasonable assurance of the structural integrity and leak-tightness of the St. Lucie Unit 1 replacement RPV closure head by extending the inspection frequency of the head from a maximum of 10 years to a new maximum of 15.5 calendar years.

3.8 NRC Staff Evaluation

The NRC staff has evaluated this relief request pursuant to 10 CFR 50.55a(z)(1). The NRC staff's evaluation focused on whether the proposed alternative provides an acceptable level of quality and safety.

In evaluating the technical sufficiency of the licensee's proposed alternative, the NRC staff considered each of the three aspects of the licensee's technical basis for use of the proposed alternative. The NRC staff notes that due to PWSCC concerns, many PWR plants in the United States and overseas have replaced their RPV closure heads containing Alloy 600/182/82 nozzles with heads containing Alloy 690/152/52 nozzles. Alloy 690/152/52 materials have been resistant to PWSCC. The inspection frequencies developed in Code Case N-729-1 for the RPV closure head containing Alloy 600/182/82 materials were based, in part, on those materials' crack growth rate equations documented in MRP-55 and MRP-115. The licensee's primary technical basis is to present crack growth rate data for Alloy 690/152/52 materials that are resistant to PWSCC and demonstrate an FOI of these materials versus Alloy 600/82/182 materials. This FOI would then provide the basis for extension of the ISI frequency requested by the licensee in its proposed alternative.

In evaluating the licensee's first technical basis for use of the proposed alternative, the NRC staff notes that the licensee uses MRP-375. This document, in part, summarizes numerous Alloy 690/152/52 crack growth rate data from various sources to develop FOI for the crack growth rate equations provided in MRP-55 and MRP-115. While the NRC staff finds that the licensee's assertions and interpretations are reasonable, MRP-375 is not an NRC-approved document. Furthermore, the data provided in MRP-375 is currently under detailed review by an international group of experts as part of an Alloy 690 Expert Panel.

In the interim, the NRC staff's review will rely upon Alloy 690/152/52 crack growth rate data from two NRC contractors: Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). This data is documented in a data summary report dated October 30, 2014 (ADAMS Accession No. ML14322A587). Furthermore, this data generally supports the contention that the crack growth rate of Alloy 690/52/152 is more crack resistant but differs from the MRP-375 data in some respects.

The PNNL and ANL data summary report 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 of the licensee's proposed alternative. 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 690/152/52 materials. 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 results, and the limited area of continuous weld dilution for 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 very small areas of weld dilution zone would result in a significantly increased probability of leakage or component failure during a relatively short extension of the required inspection interval. 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. 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 request.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds that the result of past bare metal VEs of the head under consideration is a reasonable means to demonstrate the absence of leakage through the nozzle or J-groove weld, or both, prior to the time the examination was conducted. The NRC staff also finds that performance of future bare metal VEs in accordance with Code Case N-729-1 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 frequency for bare metal VEs, in conjunction with the new frequency for the volumetric and surface examinations, is sufficient to provide additional assurance of the structural integrity of the RPV closure head.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC staff finds that the licensee's calculated FOI of 7.35 to support an extension of the Code Case N-729-1 inspection frequency of 2.25 reinspection years to 15.5 calendar years was acceptable by NRC staff calculation. The NRC staff also finds that the application of an FOI of 7.35 to the 75^{th} percentile curves in MRP-55 and MRP-115 bounded essentially all of the NRC data included in the PNNL and ANL data summary report. Therefore, the NRC staff finds that this analysis supports the concept that a volumetric inspection interval for the RPV closure head of not more than 15.5 calendar years does not pose a higher risk than that associated with an Alloy 600/182/82 RPV closure head inspected at intervals of 2.25 reinspection years. Hence, the NRC staff finds the licensee's technical basis to be acceptable.

Therefore, the NRC staff finds that the licensee has provided an adequate technical basis to demonstrate that its proposed alternative examination frequency (i.e., not exceeding 15.5 calendar years) would provide an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee's proposed alternative provides an acceptable 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)(1). Therefore, the NRC staff authorizes the use of the licensee's proposed alternative at St. Lucie Unit 1 for the duration of up to, and including, the Spring 2021 refueling outage in the fifth 10-year ISI interval.

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

Principal Contributor: Ali Rezai

Date: August 31, 2017

SUBJECT: ST. LUCIE PLANT UNIT NO. 1 – RELIEF REQUEST NO. 12 FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (CAC NO. MF9273) DATED AUGUST 31, 2017

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