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

July 10, 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 – INSERVICE INSPECTION PLAN FOURTH 10-YEAR INTERVAL RELIEF REQUEST NO. 13 (CAC NO. MF9320)

Dear Mr. Nazar:

By letter dated February 24, 2017 (Agencywide Documents Access and Management System Accession No. ML17055A618), Florida Power and Light Company (FPL) requested relief from the examination requirements of Table IWB-2500-1 of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, 2001 Edition through 2003 Addenda regarding the examinations of welds in the control element drive mechanism (CEDM) housing on top of the reactor vessel closure head at St. Lucie Plant Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.55a(z)(2), FPL requested to use the proposed alternative in Relief Request No. 13 on the basis that compliance with the specified ASME requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that FPL has addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) and that the proposed alternative to the ASME Code required examinations of welds in the St. Lucie Unit 1 CEDM housing provides reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(z)(2), the NRC authorizes the use of Relief Request No. 13 at St. Lucie Unit 1 for the fourth 10-Year inservice inspection interval, which ends on February 10, 2018.

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 Perry H. Buckberg at 301-415-1383 or Perry.Buckberg@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
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 13

EXAMINATION OF WELDS IN CONTROL ELEMENT DRIVE MECHANISM HOUSING

ST. LUCIE PLANT UNIT NO. 1

FLORIDA POWER & LIGHT

DOCKET NUMBER 50-335

1.0 INTRODUCTION

By letter dated February 24, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 17055A618), Florida Power and Light (FPL) requested relief from the examination requirements of Table IWB-2500-1 of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, 2001 Edition through 2003 Addenda regarding the examinations of welds in the control element drive mechanism (CEDM) housing on top of the reactor vessel closure head at St. Lucie Plant Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), FPL requested to use the proposed alternative in Relief Request No. 13 on the basis that compliance with the specified ASME requirements 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), 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 [ISI] of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry and materials of construction of the components.

Section 50.55a(g)(6)(ii)(D) of 10 CFR, *Reactor vessel head inspections,* requires licensees of pressurized water reactors (PWRs) to augment their ISI of the reactor vessel closure head with ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," with conditions.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if FPL demonstrates (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Enclosure

Based on the above, and subject to the following technical evaluation, the NRC staff finds that it has the regulatory authority to authorize an alternative proposed by FPL.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

The affected components are the pressure retaining welds in the CEDM housing. The welds are classified as ASME Code Class 1, Examination Category B-O, Item No. B14.10, in accordance with the ASME Code, Section XI, Table IWB-2500-1.

The St. Lucie Unit 1 replacement reactor vessel head (RVH) contains 69 CEDM housings. Of the 69 CEDM housings, 28 are periphery CEDM housings. Each of the 28 periphery CEDM pressure housings contains five full penetration butt welds (Nos. 1 through 5). CEDM weld Nos. 1 and 2 are austenitic stainless steel weld material (IN316L and ER316L). CEDM weld Nos. 3, 4, and 5 are alloy 690 weld material (Alloy 52/52M/152).

3.2 Applicable Code Edition and Addenda

The code of record is the ASME Code, Section XI, 2001 Edition through 2003 Addenda.

3.3 Applicable Code Requirement

ASME Code, Section XI, Table IWB-2500-1, Examination Category B-O, Item No. B14.10, requires volumetric or surface examination of 10 percent peripheral CEDM housing welds.

3.4 Reason for Request

FPL stated that because of the configuration, it is not practical to perform the required examinations of welds Nos. 1 through 4 in CEDM housings in accordance with Examination Category B-O, Item No. B14.10 in Table IWB-2500-1 of the ASME Code, Section XI. During refueling outage SL1-27 in the fall of 2016, FPL identified that CEDM weld No. 5 is only accessible at four inspection ports. Each of the four inspection ports provides access to two CEDM No. 5 welds, allowing a total of eight CEDM No. 5 welds for examination. The proposed Relief Request addresses CEDM housing weld Nos. 1 through 4 that are inaccessible for examination.

FPL explained that removal and disassembly of the CEDM coil stack assembly and seismic shroud assembly to perform the surface or volumetric examinations of CEDM housing weld Nos. 1 through 4 in is not possible without significant work, significant radiation exposure, and/or potential damage to the plant. The two upper welds (CEDM Nos. 1 and 2) and two motor housing welds (CEDM Nos. 3 and 4) are covered by the shroud assembly that provides seismic support and houses the sensitive rod position indicator coils. The shroud assembly also serves as the support for the CEDM coil stack assembly, which consists of electromagnetic coils mounted on the outside of the CEDM motor housing assembly. The coils supply magnetic force to actuate the mechanical latches used in engaging and driving the notched drive shaft. The electromagnetic coils of the coil stack assembly completely cover CEDM weld Nos. 3 and 4. Only the lower weld (CEDM No. 5) is accessible for examination. FPL estimated that the time required for disassembly, examination, and re-assembly of the CEDM housing would result in a radiological dose of 3.36 rem for the examination of welds Nos. 1 through 4 on one CEDM housing.

FPL explained that disassembly of sensitive CEDM electrical components would be considered high risk because they position control rods that control reactor reactivity and have a safe shutdown function. Safety-related components may be damaged during disassembly and restoration. Alignment and post-maintenance testing of the components may severely impact the plant with an extended off-line condition to obtain long-lead replacement parts, if required.

To examine CEDM welds Nos. 2, 3 and 4, FPL stated that it would need to remove the seismic-supported RVH duct work for access inside the RVH service structure that surrounds all 69 CEDMs. To gain access to these welds would require service structure modifications, a significant level of service structure disassembly or removal entirely. Any modification would be a significant level of effort as the service structure is an engineered structure.

Removal and reinstallation of the CEDM coil stacks and sensitive control rod position indicators would also require post-maintenance testing that could not be performed until reactor vessel reassembly. Although testing of the CEDM assembly is a normal part of start-up testing, the possibility of a post-maintenance testing failure is increased as a result of handling these sensitive components.

3.5 Proposed Alternative

In lieu of examining the subject CEDM welds in accordance with Examination Category B-O in Table IWB-2500-1 of the ASME Code, Section XI, FPL proposed that (1) it will perform the VT-2 visual examination on subject welds (Nos. 1 through 5) as part of the system leakage test performed during each refueling outage in accordance with Examination Category B-P of Table WB-2500-1, and (2) it will perform surface or volumetric examination on eight accessible CEDM weld No. 5 locations in accordance with Examination Category B-O in Table IWB-2500-1 of the ASME Code, Section XI. The CEDM weld No. 5 location is only accessible for inspection through four inspection ports; two welds are accessible through each inspection port, for a total of eight welds.

FPL noted that the periodic percentage requirements in Table IWB-2412-1 of the ASME Code, Section XI, do not apply to Examination Category B-O weld (i.e., CEDM weld Nos. 1 through 5) per subparagraph IWB-2412(a)(3) of the ASME Code, Section XI, which allows deferral to the end of the interval.

3.6 Basis for Use

FPL replaced the original RVH assembly, including CEDMs, with all new components during the 2005 refueling outage (SL1-20). Before deploying the new, replacement RVH, FPL performed preservice surface examinations of all five CEDM housing welds on the 28 periphery CEDMs. FPL also performed a preservice volumetric examination of the CEDM welds using equipment, procedures, and personnel qualified in accordance with ASME Section XI, Appendix VIII. No indications were identified. FPL has not performed surface and/or volumetric examinations of weld No. 5 on any of the CEDM housings during the current fourth10-year ISI interval (February 11, 2008, to February 10, 2018).

FPL noted that for stress corrosion cracking (SCC) to occur in the CEDM housing welds, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment.

Stresses

FPL stated that while residual stresses are always present as a result of welding, the inside diameter stresses in CEDM housing welds are minimized because all welding is made from the component outside diameter that precludes the possibility for inside diameter repairs.

Material Specifications

FPL stated that the CEDM upper pressure housing sub-components and weld Nos. 1 and 2 are made with 316/316L austenitic stainless and are the same grade as previously used at St. Lucie Unit 1 without any service related degradation. FPL further stated that the 316L austenitic stainless steel weld material in CEDM weld Nos. 1 and 2 are resistant to SCC in controlled reactor coolant system (RCS) conditions based on the years of operating experience without SCC in primary loop piping welds. The original weld Nos. 1 and 2 had been in service for approximately 29 years from the commercial service in 1976 through the 2005 reactor vessel closure head replacement. FPL stated that the replacement CEDM motor housing is similar to the original CEDM motor housing except the original alloy 600 material and alloy 82/182 weld material has been replaced with alloy 690 and its compatible alloy 52/152 weld material (i.e., weld Nos. 3, 4, and 5), which has superior resistance to primary water stress corrosion cracking (PWSCC) degradation.

Environment

To minimize corrosion, FPL's Chemistry Control Program controls RCS to reduce coolant oxygen to less than or equal to 100 parts per billion (ppb) steady state and a normal value of less than 5 ppb during normal operation. Contaminants known to increase the susceptibility of austenitic stainless steels are also strictly controlled in the RCS coolant environment by the Chemistry Control Program and Technical Specifications (TSs). The low temperature of the CEDM column also tends to decrease the susceptibility to SCC mechanisms. The low temperature of the CEDM column near Weld No. 1, where any trapped air could potentially exist, also tends to decrease the susceptibility to SCC mechanisms. FPL measured the temperature at CEDM Weld No. 1 to be below 140 °F during operation at St. Lucie Unit 2, which bounds the temperature of weld No. 1 at St. Lucie Unit 1. FPL concluded that the conditions for SCC to occur in the CEDM housing are extremely unlikely.

CEDM weld No.1 is the only weld potentially not in contact with coolant during operation. However, trapped air that contains oxygen may exist in the vicinity of weld No. 1. FPL stated that as the RCS pressure increases during start-up, the trapped volume of air is squeezed until the remaining volume is reduced to a fraction of its original size. Further, during start-up there is control rod drop testing that results in a rapid exchange of RCS coolant with the coolant in the CEDM column to further reduce the air volume. Eventually, the gas pocket would be expected to nearly disappear during plant operations as the gas is forced into solution and exchanged with the bulk RCS coolant. Although removing trapped oxygen prior to initial start-up by venting each CEDM is not regularly performed, the combination of mechanical agitation with rod drop, reducing the trapped air volume with operating pressure, and dissolution of trapped air, removes trapped oxygen that could cause weld degradation in the CEDM near weld No. 1.

Weld design and Welding Process

FPL stated that the all CEDM housing welds were fabricated with the gas tungsten arc welding (GTAW) process. All are V-groove welds and are made from the component outside diameter. The stainless steel SA182-F403 motor housing has a nickel-based Alloy 52M (ERNiCrFe-7A) weld build up or 'butter' on each end followed by a post-weld heat treatment (PWHT) that meets the requirements of the ASME Code, Section III, NB-4622.1-1 and Code Case N-2 from the original Construction Code. The weld buttered ends of the motor housing received a liquid penetrant test (PT) before the end fittings were attached. No welding to the stainless steel F403 material was allowed after PWHT. The RVH penetration to head adapter weld No. 5 is also made primarily with the GTAW process but the option to use the shielded metal arc welding process was available. All pressure boundary housing welds received the required PT and radiography (RT) inspections. All weld procedures were qualified in accordance with the ASME Code, Section IX.

The replacement motor housing assembly welds (CEDM Nos. 3 and 4) use a modern narrow groove weld joint geometry so that less weld material is present. The replacement weld joint designs are in compliance with the design, fabrication, inspection, and testing requirements of the ASME Code, Section III, 1998 Edition through the 2000 Addenda for Class 1 components.

RCS Leakage Detection

According to FPL, because SCC is a time dependent degradation mechanism, if a postulated through wall CEDM housing leak were to occur in these ductile materials, there would be time for detection prior to a 360° circumferential break to occur. St. Lucie Unit 1 has several methods for early detection of RCS leakage by operators such that detection would occur prior to a guillotine break. FPL concluded that the consequence of a guillotine break is a highly unlikely event.

FPL stated that the primary method for quantifying and characterizing RCS identified and unidentified leakage is by means of a reactor coolant water inventory balance. The inventory balance is performed as required by TS 4.4.6.2.c at least once every 72 hours except when operating in the shutdown cooling mode (not required to be performed until 12 hours after establishment of steady state operation). However, the plant surveillance procedure requires the inventory balance be performed once every 24 hours. The procedure uses the guidance in Westinghouse report WCAP-16423-NP (ADAMS Accession No. ML070310084) and WCAP-16465-NP (ADAMS Accession No. ML070310082). FPL stated that the water balance inventory method can calculate the leak rate to the nearest 0.01 gallons per minute (gpm).

In its request, FPL stated that Action levels on the absolute value of Unidentified RCS Inventory Balance from surveillance data are as follows:

Action Level 1
- An adverse trend over time is observed
- Seven day rolling average of UNIDENTIFIED leak rate is greater than 0.1 gpm.
- Nine consecutive RCS UNIDENTIFIED leak rates greater than the baseline mean value.

Action Level 2
- Two consecutive UNIDENTIFIED leak rates greater than 0.15 gpm.
- Two of three consecutive UNIDENTIFIED leak rates greater than the baseline mean plus two times the standard deviation.

Action Level 3
• One UNIDENTIFIED leak rate greater than 0.30 gpm.
• One UNIDENTIFIED leak rates greater than the base line mean value plus three times the standard deviation.

These Action Levels trigger condition reporting, various investigations of leakage up to and including containment entry to identify the source of the leakage.

FPL noted that RCS leakage can also be detected by three other 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 TS required monitoring (TS 4.4.6.2 a and 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 that is dependent on the percentage of failed fuel. FPL stated that the containment atmosphere radiation monitors can detect a 1 gpm leak rate within 1 hour using design basis reactor water activity assuming 0.1 percent failed fuel. The containment sump monitoring system can detect a leak rate of 1 gpm.

St. Lucie Unit 1 TSs were revised to meet Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler-513 to define new time limit for restoring inoperable RCS leakage detection instrumentation to operable status and to establish alternate methods of monitoring RCS leakage when one or more required systems are inoperable (ADAMS Accession No. ML110871284). By letter dated March 30, 2012, NRC approved TSTF-513 Revision 3 (ADAMS Accession No. ML12052A221).

<u>Operating Experience</u>

FPL noted that Combustion Engineering designed latch driven CEDMs with butt welds such as at St. Lucie Unit 1 have had excellent service performance history. FPL did not find failures of CEDM butt welds of the design described above in the Institute of Nuclear Power Operation's database. Although some failures were noted in the thin ligament specially designed seal welds that are used on the threaded joints, they do not receive the same level of nondestructive examination that the affected CEDM butt welds receive. In addition, the seal welds have a unique configuration of trapped or occluded water chemistry that is not a factor in the St. Lucie Unit 1 CEDM butt weld RCS environment.

FPL has explained that there have been no reported degradation in CEDM housing pressure boundary welds above the RVH that are the same type as the St. Lucie Unit 1 CEDM assemblies. Two nuclear plants have experienced degradation with their larger diameter rotating motor CEDM housings. One plant has developed repeated cracking in the large diameter motor CEDM housings that have resulted in complete replacement of the bolted and flanged CEDMs multiple times. Another plant with the same type of large diameter CEDM housings has only two failures of spare CEDM housings associated with an internal weld overlay and welded orifice flow restrictor to limit heat loss. The operating experience of these two plants suggests that the plant with multiple failures is an outlier to the PWR industry.

Inservice Examinations

After installation of the replacement RVH in 2005, FPL performed bare metal visual inspection of the entire head surface and the CEDM to RVH interface in 2010 and 2015 in accordance with the requirements of ASME Code Case N-729-1 as modified by 10 CFR 50.55a. FPL did not observe any leakage.

FPL stated that it performs walk downs of the RVH after shutdown and during start-up looking for leakage or other abnormal conditions. FPL performs the required VT-2 visual examination of all CEDMs from the 62-foot containment elevation. FPL looks down from the platform above the CEDM housings because no permanent ladder is available in the upper cavity and the temporary access is removed prior to beginning the examination following reactor vessel re-assembly.

FPL performs a visual examination at the beginning of each outage, prior to RVH disassembly. FPL performs the visual examination from the upper cavity elevation using the inspection ports surrounding the RVH. The inspection is also performed from the incore instrument column access doors inside the RVH shroud during disassembly for evidence of leakage as well as all the accessible CEDMs. FPL stated that any evidence of leakage is required to be entered in the corrective action program and dispositioned.

3.7 Duration of Proposed Alternative

FPL stated that this relief request is applicable to the fourth ISI Interval that began February 11, 2008, and ends February 10, 2018. An interval extension is being used to complete the fourth ISI Interval as allowed by IWA-2430(c)(1). The fifth ISI will start on February 10, 2018. Credit for these examinations will only be applied to the fourth ISI Interval.

3.8 NRC Staff Evaluation

The NRC staff evaluates the proposed alternative in terms of stresses, material specifications, environment, weld design and welding process, and operating experience. Based on this evaluation, the NRC staff determines the potential for degradation for the subject welds. As a defense-in-depth consideration, the NRC staff evaluates the adequacy of proposed examinations and effectiveness of the RCS leakage detection capability in case leakage does occur in the subject welds. Finally, the NRC staff reviews the validity of the licensee's hardship justification.

Stresses

The NRC staff recognizes that while residual stresses are always present in the welds as a result of welding, the residual stresses at the inside diameter of CEDM housing welds should be minimized because all five welds were made from the outside surface of the housing. The weld design minimizes the repairs on the inside diameter surface of the welds that, in turn, minimizes associated residual stresses. In terms of operating stresses caused by applied loading, the NRC staff determines that the licensee applied appropriate loads from pressure, temperature, and seismic effect. The NRC staff notes that the welds are designed to satisfy the allowable stresses in the Construction Code. Therefore, the NRC staff finds that the total stresses at the subject welds are within the design allowable.

In addition, the NRC staff notes that weld No. 1 is located at the upper part of the upper pressure housing and, therefore, supports minimal loads. This means that the stresses in weld No. 1 from the applied load would be small. With low stresses and non-corrosive environment, the potential for SCC would be minimal at weld No. 1 as well.

Material Specifications

The NRC staff finds that stainless steel and alloy 52/52M/152 metals used for the CEDM welds minimizes SCC because they are less susceptible to SCC than alloy 600/82/182 metals based on laboratory testing and industry operating experience.

Environment

The NRC staff finds that FPL controls the RCS coolant chemistry by limiting oxygen to a steady state limit of less than or equal to 100 ppb and a normal value of less than 5 ppb during normal operation. Contaminants known to increase the susceptibility of austenitic stainless steels are also strictly controlled in the RCS coolant environment by the chemistry control program and TSs. The NRC staff determines that the environment will not likely cause SCC in all five welds.

The NRC staff notes that even though CEDM weld No.1 is the only weld potentially not in contact with coolant during operation, trapped air that contains oxygen may exist in the vicinity of weld No. 1. The NRC staff notes that trapped air may contain excessive oxygen that may cause corrosive environment in the vicinity of weld No.1. As stated above, the trapped air may be reduced and oxygen may be removed during start-up which would reduce the likelihood of SCC in weld No. 1. The NRC staff notes that the probability of SCC is reduced at low temperatures based on laboratory tests. As stated above, the temperature at the weld No. 1 location is below 140 °F during operation, which is low enough to decrease susceptibility to SSC mechanisms. Therefore, the NRC staff finds that SCC would not likely occur at weld No. 1

Weld Design and Welding Process

The NRC staff notes that all CEDM housing welds were fabricated with GTAW process that has rigorous welding requirements to minimize fabrication defects. All are V-groove welds and are made from the component outside diameter.

The NRC staff finds that the CEDM housing weld joint designs are in compliance with the design, fabrication, inspection, and testing requirements of the ASME Code, Section III, 1998 Edition through the 2000 Addenda for Class 1 components. The stainless steel SA182-F403 motor housing has a nickel-based Alloy 52M (ERNiCrFe-7A) weld build up or butter on each end followed by a PWHT that meets the requirements of the ASME Code, Section III, NB-4622.1-1 and Code Case N-2 from the original Construction Code. The weld buttered ends of the motor housing received a liquid PT before the end fittings were attached. No welding to the stainless steel F403 material was allowed after PWHT. The RVH penetration to head adapter weld No. 5 is also made primarily with the GTAW process but the option to use the shielded metal arc welding process was available. All pressure boundary housing welds received the required PT and RT inspections. All weld procedures were qualified in accordance with the ASME Code, Section IX.

The replacement motor housing assembly welds (CEDM Nos. 3 and 4) use a modern narrow groove weld joint geometry so that less weld material is present. The NRC staff notes that the subject welds are designed with a V-groove configuration that are made from the outside diameter of the housing. The V-groove weld would have minimum deposit at the inside surface of the housing. This reduces the chance of SCC because the weld would be limited in contact with RCS coolant on the inside surface of the housing. In addition, because the weld area is limited on the inner diameter surface of the housing, the welds were not required to be repaired at the inside diameter of the housing during construction. This reduces the chance of crack initiation at the inside diameter surface of the housing.

The NRC staff finds that SCC will not likely occur in the CEDM housing welds because FPL has used appropriate weld materials, properly controlled RCS coolant chemistry, maintained only allowable stresses on the housing, and used a weld configuration that minimizes weld repairs.

Operating Experience

In terms of industry-wide operating experience, NRC staff identified the same two nuclear plants that have had degraded welds in the control rod drive mechanism (CRDM) housings that FPL has also identified as discussed above. Plant A has had degraded welds in CRDM housings since the 1980s. Plant A replaced CRDM housings in the 1990s and 2001 to eliminate weld degradations. Flaws were detected in the welds of the CRDM housing in 2001 and 2012. The flaws were caused by transgranular SCC. The major contributors to degradation in the affected Plant A welds are; (a) significant cold work was identified on the inside surface of the weld, (b) the CRDM rubbing against extra weld buildup at the inside surface used for alignment and, (c) possible elevated oxygen and chloride level in the environment at the degraded weld location.

Plant B detected flaws in welds in two spare CRDM housings in 1990. The flaws were caused by transgranular SCC. One of the degraded welds was also used for alignment. The failure analysis showed that the oxygen content in the spare CRDM housings was in the range of 300 to 1300 parts per million because the affected CRDM housings were not vented and are dead legs. Dead legs are thought to retain higher oxygen and chloride levels from refueling outages than the areas of the RCS that are exposed to active flow. As a comparison, the bulk primary RCS coolant has an oxygen content of 5 ppb at Plant B. The key contributors to the degraded weld could be elevated oxygen and chloride level in the environment at the degraded weld location and the impact of the pressure and/or temperature cycles.

The NRC staff did not identify degradation in CEDM housing welds in domestic nuclear plants that use the same type of the CEDM housing design used at St. Lucie Unit 1. Weld Nos. 1 through 4 at St. Lucie Unit 1 do not have weld buildup at the inside surface of the housing and they are not used to provide alignment for the CEDM internals as are the affected welds in Plants A and B discussed above. Therefore, potential cracking resulting from cold work (CEDM internals banging on the welds) or wear (CEDM internals rubbing) on these welds at St. Lucie unit 1 is not a concern.

The diameter of weld Nos. 1 and 2 at St. Lucie unit 1 is smaller than the affected welds in the CRDM housing used in Plants A and B. Weld Nos. 3 and 4 are made with the narrow V-groove design. A weld with a narrow groove design and a smaller diameter housing contains less weld metal than the normal groove weld with a large diameter housing. This minimizes the chance of fabrication defects. In addition, the V-groove weld design limits the exposure of the weld material on the inside surface of the housing. This minimizes the contact between the weld and primary water environment inside the housing, thereby reducing susceptibility to SCC.

As stated above, the plant-specific operating experience at St. Lucie Unit 1 has shown that no leakage has been detected in the subject welds since the RVH was replaced in 2005. In addition, the NRC staff notes that FPL did not find any degradation in the welds that were made of stainless steel and Alloy 82/182 weld metal in the original RVH with about 24 years of operating experience.

The NRC staff determined that the contributors to cracking in CRDM housing in Plants A and B are not applicable to the CEDM housing welds at St. Lucie Unit 1 because of the differences in the design of welds and housing between St. Lucie Unit 1 and Plants A and B.

Potential for Degradation

The NRC staff finds that (a) the stresses in the CEDM weld Nos. 1 through 4 are within the ASME Code allowable, (b) the weld material is less susceptible to SCC, (c) the RCS coolant environment is controlled to reduce corrosion, (d) the plant-specific operating experience of the CEDM housings has been favorable because FPL has not found any leakage from the subject welds and, (e) a guillotine break at the subject welds would not be likely because the welds are fabricated with either stainless steel or alloy 690 material that have sufficient fracture toughness to resist catastrophic failure. Based on the above, the NRC staff finds that the potential for SCC in weld Nos. 1 through 4 is small.

Examinations

The NRC staff finds it acceptable that FPL performed preservice surface and volumetric examinations of all welds in all CEDM housings prior to placing the replacement RVH in service based on the method of the ASME Code, Section XI, Appendix VIII. For inservice examination, the NRC staff finds it acceptable that FPL has performed (a) VT-2 examinations as part of the system leakage test during every refueling outage in accordance with IWA-5000 and IWB-5000 of the ASME Code, Section XI, (b) a visual examination of the RVH through the inspection port holes at the beginning of each refueling outage, and (c) periodic bare metal visual examinations of the RVH in accordance with ASME Code Case N-729-1 as conditioned in 10 CFR 50.55a(g)(6)(ii)(D) in 2010 and 2015.

RCS Leakage Detection Capability

The NRC staff determines that as a defense-in-depth measure, if a through-wall leak does develop in the CEDM housing welds, FPL would be able to detect the leak based on the following methods: (1) reactor coolant water inventory calculations, (2) the reactor cavity (containment) sump inlet flow monitoring system, (3) the containment atmosphere radiation gas monitoring system, and (4) the containment atmosphere radiation particulate monitoring system. In addition, FPL has implemented stringent administrative leakage controls such that it will take corrective actions before the leak rate reaches the TS leakage limit of 1 gpm. The NRC staff notes that the corrective actions performed early will reduce the likelihood of CEDM housing failures. Therefore, the NRC staff finds that the diverse RCS leakage detection systems and stringent administrative leakage controls will alert plant operators of any potential leakage from the CEDM housings readily such that the operator can take prompt corrective actions.

Hardship Justification

The NRC staff finds that the design of the CEDM housing restricts the accessibility of weld Nos. 1 through 4. To perform the required examination, FPL would need to disassemble the

CEDM housing, such as removing the seismic shroud, cooling shroud, coil stack assembly, electrical cables, and other components. The removal and reinstallation of these assemblies would expose plant personnel to a radiological dose and is a hardship to FPL. In addition, the assembly and disassembly of the CEDM housing may introduce human errors that may cause adverse effects on the operation of the CEDM. Therefore, the NRC staff has determined that it is a hardship and unusually difficult for FPL to perform either surface or volumetric examinations of CEDM weld Nos. 1 through 4 in accordance with the ASME Code, Section XI.

In summary, the NRC staff finds that the likelihood of failure for CEDM weld Nos. 1 through 4 is remote because the potential for degradation is small, the welds will be examined periodically to monitor for structural integrity, and any potential leakage from the welds will be promptly detected by the RCS leakage detection systems.

4.0 CONCLUSION

As set forth above, the NRC staff determined that FPL has demonstrated reasonable assurance of structural integrity and leak tightness of CEDM weld Nos. 1 through 4. The NRC staff also determined that compliance with the specified ASME requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that FPL 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 Relief Request No. 13 at St. Lucie Unit 1 for the fourth ISI Interval that began on February 11, 2008 and ends on February 10, 2018.

All other requirements of ASME Code, Section XI, for which relief has not been specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: John Tsao

Date: July 10, 2017

SUBJECT: ST. LUCIE PLANT UNIT NO. 1 – INSERVICE INSPECTION PLAN FOURTH 10-YEAR INTERVAL RELIEF REQUEST NO. 13 (CAC NO. MF9320) DATED JULY 10, 2017

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