FPL.

August 27, 2015

L-2015-218
10 CFR 50.4
10 CFR 50.55a

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Re: St. Lucie Unit 1
Docket No. 50-335
Inservice Inspection Plan
Fourth Ten-Year Interval Unit 1 Relief Request No. 10, Revision 0

Pursuant to 10CFR50.55a(z)(2), Florida Power & Light (FPL) requests relief from the examination requirements of the ASME Code, Section XI, 2001 Edition through 2003 Addenda, for the subject CEDM welds. FPL requests that this relief be approved prior to start of the fall 2017 St. Lucie Unit 1 refueling outage that is currently scheduled to begin on September 26, 2017.

The attachment to this letter provides FPL's justification for this relief request.

Please contact Ken Frehafer at (772) 467-7748 if there are any questions about this submittal.

Sincerely,

ES Katzman

Eric S. Katzman
Licensing Manager
St. Lucie Plant

Attachment
ESK/KWF

cc: USNRC Regional Administrator, Region II
USNRC Senior Resident Inspector, St. Lucie Units 1 and 2

A047
NRR

Relief Request
In Accordance with 10 CFR50.55a(z)(2)

-- Hardship or Unusual Difficulty Without Compensating Increase in Level of Quality or Safety.--

1. ASME Code Component(s) Affected

Class 1, Category B-O, Item Number B14.10, Pressure Retaining Welds in Control Rod Housings.

2. Applicable Code Edition and Addenda

The Code of record for St. Lucie Unit 1 is the 2001 Edition through 2003 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" as modified by 10CFR50.55a.

3. Applicable Code Requirement

Exam Cat.	Item No.	Examination Requirements
B-O	B14.10	Volumetric or surface examination of 10% peripheral CRD housings

4. Reason for Request

Due to the configuration of the control rod drives, it is not practical to meet the examination requirements of the ASME Code, Section XI, 2001 Edition through 2003 Addenda. Relief is requested in accordance with 10 CFR50.55a(z)(2).

This Relief Request has been written to address CEDM housing welds that are inaccessible for examination and contains a proposed alternative that provides an acceptable level of quality and safety.

Removal and disassembly of the CEDM Coil Stack Assembly and Seismic Shroud Assembly Seismic Shroud Assembly to facilitate examination is not possible without significant work, significant radiation exposure, and/or potential damage to the plant. An estimate of the time required for disassembly, examination, and re-assembly of the CEDM for the performance of the examinations is estimated to result in 3.36 rem for the performance of the examination of 1 CEDM, welds 1 through 4.

The St. Lucie Unit 1 replacement reactor vessel head contains sixty-nine (69) control element drive mechanisms (CEDM). Of the sixty-nine (69) CEDMs, there are twenty-eight (28) periphery CEDMs. Each of the twenty-eight (28) periphery CEDM pressure housing assemblies contains 5 full penetration butt welds that require a surface or volumetric examination.

It is not possible to perform the surface or volumetric examinations of 4 of these welds in CEDM housings without significant reactor vessel head disassembly. The 2 upper welds (CEDM 1 & 2) and 2 motor housing welds (CEDM 3 & 4) are covered by the shroud assembly which 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. Only the lower weld (CEDM 5) is accessible for examination.

Direct access to the CEDM 1 – 4 welds is not obtainable without extensive disassembly of the CEDM Coil Stack Assembly (drive motor coils) and Seismic Shroud Assembly.

- Unlike piping weld inspections that only require insulation removal, inspection of CEDM welds 1 through 4 requires significant disassembly of permanent plant equipment.

- Direct Access to the CEDM 1 – 4 welds is not obtainable without extensive disassembly of the CEDM Coil Stack Assembly and Seismic Shroud Assembly that covers the CEDM pressure and motor housings. These activities would be considered a High Risk since they involve disassembly of sensitive electrical components that position control rods that function to control reactivity and also have a safe shutdown function. There is also the risk of damage to components during disassembly and restoration as well as alignment and post maintenance testing that could severely impact the plant with an extended off-line condition to properly obtain long lead replacement parts, if required.

- Access inside the Reactor Vessel Head (RVH) service structure that surrounds all 69 CEDMs in Unit 1 acts as part of the seismic supported RVH duct work, is required to be removed in order to perform inspections on CEDM welds 2, 3 & 4. 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 (PMT) 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 PMT failure is increased as a result of handling these sensitive components.

Component Description and Function

The internal mechanisms of the CEDM are magnetic jack drives (latch mechanisms) used to vertically position the control element assemblies (CEA) in the core. Each CEDM is capable of withdrawing, inserting, holding, and tripping the CEA from any point within its travel in response to operating signals. Each CEDM is connected to its associated CEA by an extension shaft. Each CEDM housing is surrounded by a seismic shroud on the outside, designed to limit horizontal movement. The seismic shroud completely covers CEDM welds 1 & 2.

Each CEDM has a 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 utilized in engaging and driving the notched drive shaft. The electromagnetic coils of the coil stack assembly completely cover CEDM welds 3 & 4.

A reed switch position transmitter (RSPT) assembly is positioned so as to utilize the permanent magnet in the top of each extension shaft. The permanent magnet actuates the reed switches one at a time as it passes by them. This signal, which gives the position of the CEA, is transmitted by the RSPT via cabling. The RSPT is contained in the seismic shroud.

The weld metal and the adjoining base metals for the CEDM welds are listed in the table below:

CEDM Weld #	Component /Spec	Component/Spec	Weld material	Comment
1	Upper Pressure Housing (UPH) Upper End Fitting, SA479-Tp316	UPH-Tube, SA-213 Tp316	IN316L & ER316L	GTAW process, Consumable insert & bare wire
2	UPH-Tube, SA-213 Tp316	UPH Lower End Fitting, SA479-Tp316	IN316L & ER316L	GTAW process, Consumable insert & bare wire
3	Motor Housing Upper End Fitting, SA182-F348	Motor Housing, SA182-F403 Mod with CC N-2	ERNiCrFe-7A (UNS N06054)	GTAW process, bare wire
4	Motor Housing. SA182-F403 Mod with CC N-2	Motor Housing Lower End Fitting, SB166-Alloy N06690	ERNiCrFe-7A (UNS N06054)	GTAW process, bare wire
5	Reactor Vessel Head Penetration Adapter, (SB166-Alloy N06690)	Reactor Vessel Head Penetration Tube, (SB167-Alloy N06690)	ERNiCrFe-7 (UNS N06052) or ENiCrFe-7 (UNS W86152)	GTAW process or SMAW process

The CEDM upper pressure housing sub-components and welds 1 and 2 are 316/316L austenitic stainless and are the same grade as previously used at St. Lucie Unit 1 without any service related degradation. The CEDM motor housing is similar to the original CEDM motor housing except the alloy 600 material and weld material has been replaced with alloy 690 and its compatible alloy 52/152 weld material which has superior resistance to PWSCC degradation.

The welding process for the all the CEDM housings was the gas tungsten arc welding (GTAW) process. All welds are V-groove welds made from the component outside diameter. The SA182-F403 motor housing has a ERNiCrFe-7A weld build up or butter on each end followed by a post weld heat treatment (PWHT) that meets ASME Section III NB-4622.1-1 as well as Code Case N-2 from the original Construction Code. The weld buttered ends of the motor housing received a liquid penetrant test (PT) inspection before the end fittings were attached. No welding to the F403 material was allowed after PWHT. The reactor vessel head penetration to head adapter weld # 5 is also made primarily with the GTAW process but the option to use the SMAW process was available. The specific weld and base materials are listed in the table above. All pressure boundary housing welds received the required PT and radiography (RT) inspections. All weld procedures were qualified in accordance with the ASME Section IX Code.

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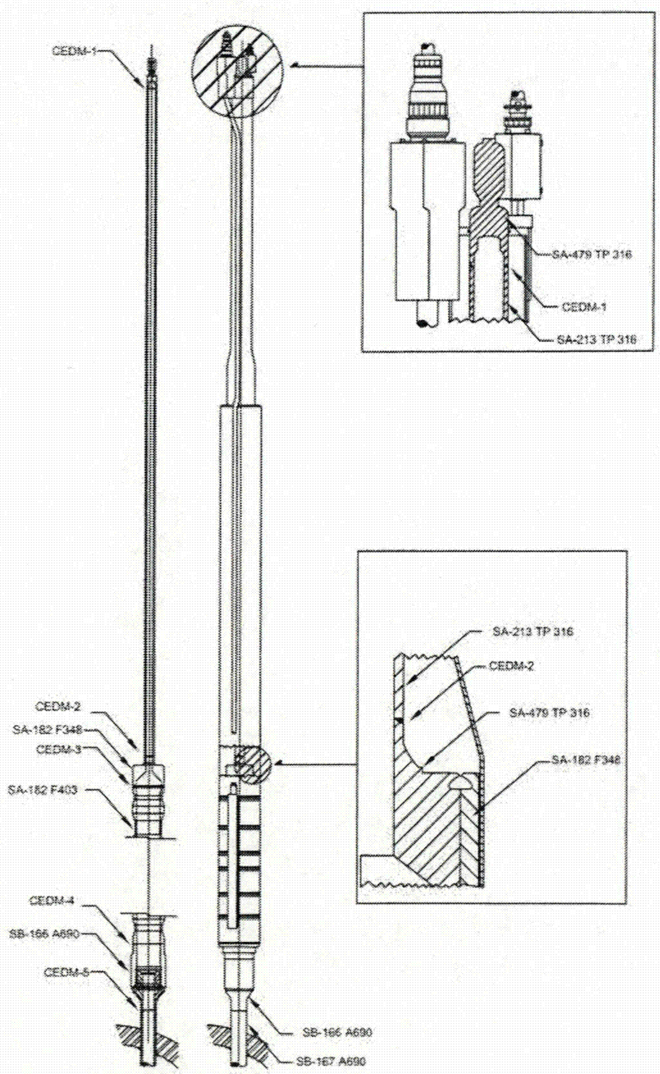
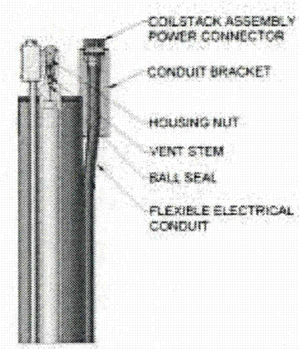
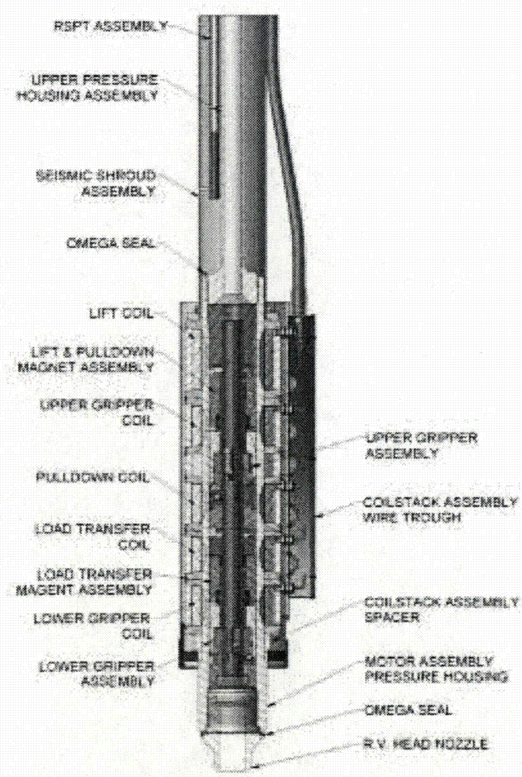
Below are illustrations and descriptions of the key components to show the level of effort required to gain access to CEDM 1-4 welds.

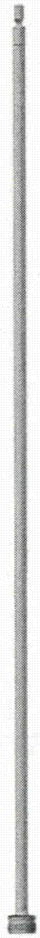




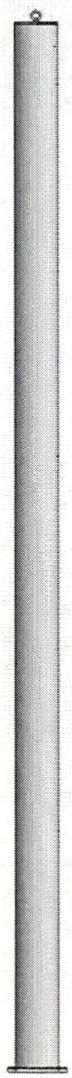


- **Upper Pressure Housing Assembly –**

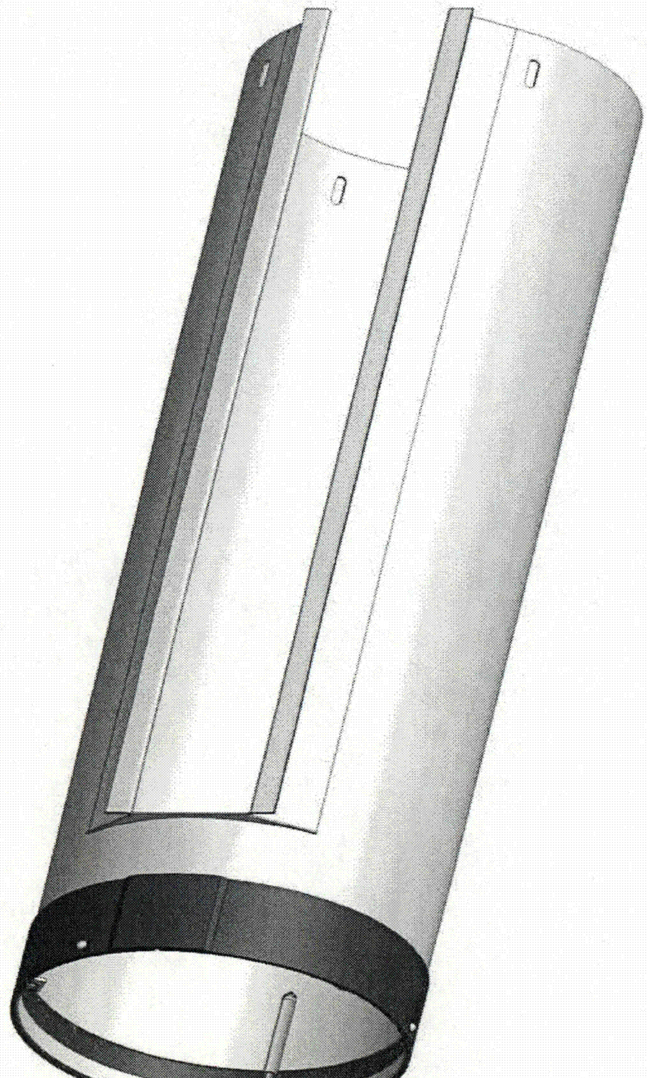
  The upper pressure housing assembly forms the upper portion of the pressure boundary and is designed and built to the ASME Boiler and Pressure Vessel Code Section III requirements for Class 1 Appurtenance for 2500 psia and 650°F. The height of the upper pressure housing is designed to allow for the complete withdrawal of the extension shaft assembly without striking the top closure plug. This is accomplished by providing clearance between the top of the pressure housing and the top of the extension shaft assembly when the latch (latch (gripper))s are engaged with the lowest lobe on the drive shaft. The upper end of the housing is designed to allow venting (if required) of the mechanism after filling the system and prior to a hydrostatic test or system operation. The seal utilizes a 440 stainless steel ball seating on a 316 stainless steel seat. A housing nut is installed to act as a closure plug. The housing nut has an Omega seal that can be welded in the event of seat leakage through the mechanical seal or the EPR O-Ring.



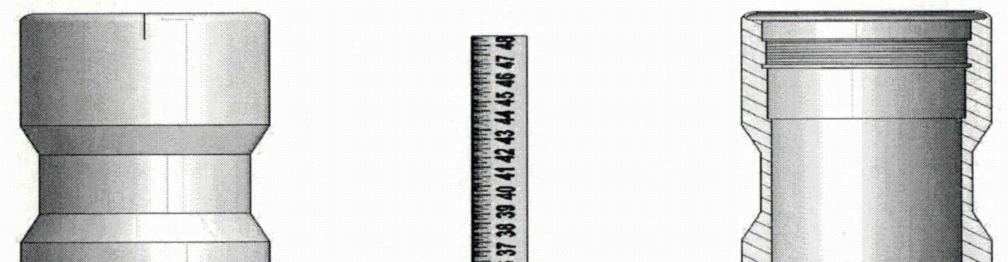
- **Seismic Shroud Assembly -** A stainless steel shroud extends from the top of the coil stack assembly to the top of the upper pressure housing. The shroud acts as a lift rig for the removal and installation of the coil stack from the upper end of the mechanism. Two shoulder eyebolts are installed in the top plate of this assembly to facilitate removal. The shroud is bolted to the coil stack.



- **Cooling Shroud Assembly -** The coil stack is surrounded by a sheet metal cooling shroud producing an annulus through which air can flow through. Air flowing from top to bottom of the coil stack will remove operating coil heat and some internal CEDM heat to maintain the coils at a temperature of less than 350°. The cooling shroud, an integral part of the CEDM, seals its lower end against the plenum flow plate by a spring-loaded rubber collar.



- **The CEDM Motor Housing Assembly** - The motor housing assembly is designed and built to the ASME Boiler and Pressure Vessel Code Section III requirements for Class 1 Appurtenances operating at 2500 psia and 650ºF. The motor housing along with the upper pressure housing are NPT stamped as an assembly. The motor housing assembly contains the motor assembly and forms the lower portion of the CEDM pressure boundary. The motor housing assembly is fabricated from Type 403 stainless steel with upper end fittings of 348 stainless steel and lower end fittings of Alloy 690. The 403 stainless steel section is required for both its magnetic and strength properties. A lower ledge supports and positions the coil stack assembly and the shroud assembly. The end fittings use Acme threads for strength and Omega seals to provide the water pressure boundary.



5. Proposed Alternative and Basis for Use

Proposed Alternative

The subject welds will continue to receive VT-2 examination as required by Examination Category B-P with the Reactor Coolant Pressure Boundary (RCPB) system leakage test conducted prior to startup from each refueling outage.

FPL proposes performing nondestructive examination of 15 CEDM weld #5 locations (as opposed to examining 15 welds on 3 CEDM's) of the periphery to align with the ASME Category B-O, B14.10 requirement of 10% of the periphery as an alternative providing an acceptable level of quality and safety. The periodic percentage requirements (Table IWB-2412-1) for B-O do not apply per IWB-2412 (a)(3) which allows deferral to the end of the interval. Therefore, examination of the accessible welds on the periphery CEDMs combined with the periodic system leakage tests provides an acceptable level of quality and safety for identification of degradation.

Basis for Use

CEDM Welds Nos. 1 and 2 are austenitic stainless steel weld material and CEDM Welds Nos. 3, 4, and 5 are alloy 690 weld material. These materials are resistant to stress corrosion cracking (SCC) mechanisms in the controlled Reactor Coolant System (RCS) environment.

For SCC to occur in the CEDM housing welds, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. While residual stresses are always present as a result of welding, the ID stresses are minimized since all welding is performed from the component outside diameter and the small diameter precludes the possibility for inside diameter repairs. The alloy 690 weld material used in CEDM Weld Nos. 3, 4 and 5 are highly resistant to SCC mechanisms based on years of replacement steam generator and replacement reactor vessel head penetration performance with the alloy 690 weld and base metals. The 316L austenitic stainless steel weld material in CEDM Weld Nos.1 and 2 are also resistant to SCC in controlled RCS conditions based on the years of operating experience without SCC in primary loop piping welds. The RCS chemistry is controlled to reduce oxygen to < 5 ppb during normal operation. Contaminants known to increase the susceptibility of austenitic stainless steels are also strictly controlled in the RCS environment by Technical Specifications. The low temperature of the CEDM column also tends to decrease the susceptibility to SCC mechanisms (i.e, The CEDM Weld No. 1 has been measured to be below 135°F during operation).

Since 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. Therefore the consequence of a guillotine break is a highly unlikely event.

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 St. Lucie Unit 1 Technical Specification (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 St. Lucie Unit 1 surveillance procedure requires the inventory balance be performed once every 24 hours since it provides the best and earliest trend of an increase in RCS leakage. The procedure methods use the recommendations and guidance in WCAP-16423-NP (Adams ML070310084) and WCAP-16465-NP (Adams ML070310082). The leak rate calculated using water balance inventory method is the most sensitive of the methods available with the leak rate calculated to the nearest 0.01 gallons-per-minute (gpm).

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

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 (μ) 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 ($\mu + 2\sigma$).

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 ($\mu + 3\sigma$).

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

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

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

In addition, the St. Lucie Unit 1 Technical Specification was revised to the extent practical to meet the improvements of NRC approved revision 3 to Technical Specification Task Force (TSTF) Standard Technical Specification (STS) 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 (Ref. St. Lucie Letter L-2011-073 dated March 11, 2011, ADAMS ML110871284).

The NRC concluded in the safety evaluation that the changes to the St. Lucie Unit 1 Technical Specifications were acceptable and that "The proposed actions for inoperable RCS leakage detection instrumentation maintain sufficient continuity, redundancy, and diversity of leakage detection capability that an extremely low probability of undetected leakage leading to pipe rupture is maintained. This extremely low probability of pipe rupture continues to satisfy the basis for acceptability of LBB in GDC 4." (NRC Issuance of Amendments regarding TSTF-513 Revision 3, dated 3-30-2012, ML12052A221).

There have been no reported CEDM/CRDM housing pressure boundary welds above the reactor vessel head of the same type of welded mag jack CEDM/CRDM assemblies. There is operating experience with the large diameter rotating motor type design CEDM at two plants. One plant has developed repeated cracking in the CEDM large diameter motor housings that have resulted in complete replacement of the bolted and flanged CEDMs multiple times. Another unit with the same type of large diameter CEDMs has only 2 failures of spare CEDM housings associated with an internal weld overlay and welded orifice flow restrictor to limit heat loss. The high rate of cracking at the one unit with large diameter rotating motor type design CEDMs and the near absence of degradation on other PWR units, suggests that the plant with multiple failures is an outlier to the PWR industry.
10 CFR 50.55a(g)(4) recognizes that throughout the service life of a nuclear power facility, components which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements set forth in the ASME Code to the extent practical within the limitations of design, geometry and materials of construction of the components.

The St. Lucie Unit 1 replacement reactor vessel head contains sixty nine (69) control element drive (CEDM) mechanisms. The periphery consists of twenty eight (28) CEDMs. Prior to assembly, preservice surface examinations of all 5 welds on the twenty eight (28) periphery CEDMs was performed. In addition a volumetric preservice examination was performed of the CEDM welds prior to the assembly utilizing equipment, procedures, and personnel qualified in accordance with ASME Section XI, Appendix VIII. No indications were identified.

Florida Power & Light replaced the complete Reactor Vessel Head (RVH) assembly including CEDMs during the SL1-20 (2005) refueling outage (3RD Period of the previous interval). The purpose of the replacement of the component was to replace those materials that have been proven to have high susceptibility to primary water stress corrosion cracking (PWSCC). Replacement of the RV head and CEDMs provided all new material and components that form the RCS pressure boundary. Additionally, after installation of the replacement reactor vessel head, FPL performed bare metal visual inspection in (2010 and 2015) in accordance with the requirements of ASME Code Case N-729-1 as modified by 10CFR50.55a of the entire head surface and the CEDM to reactor vessel head interface for evidence of leakage. No evidence of leakage was observed.

CEDM weld No.1 is the only weld potentially not in contact with coolant during operation. 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 volume. Further, during start up there is control rod drop testing which 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 was forced into solution and exchanged with the bulk RCS coolant.

If a postulated through wall crack were to occur in an area of the CEDM upper pressure housing (CEDM Weld No. 1) that is air filled, the less dense air would escape more easily than RCS, removing the gas volume and bringing the through wall crack in contact with RCS coolant. Although these welds are inaccessible for PT, there are VT-2 examinations in the area of the RV head and CEDMs.

The replacement motor housing assembly welds (CEDM 3 & 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 B&PV Code, Section III, 1998 Edition through the 2000 Addenda for Class 1 appurtenances.

Combustion Engineering designed latch driven CEDMs with butt welds have had excellent service performance history. A review of operational experience in the INPO database did not identify failures of CEDM butt welds of the design described above. Although some failures were noted in the thin ligament specially designed seal welds (NB-4360) that are used on the threaded joints, they do not receive the same level of NDE 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 applicable to the conditions that the CEDM butt weld RCS environment.

Personnel and system engineers perform walk downs of the reactor head after shutdown and during startup looking for leakage or other abnormal conditions.

The required VT-2 examination is performed of all CEDMs from the 62' containment elevation and looking down from the platform above the CEDM housings since there is no permanent ladder down into the upper cavity and the temporary access for the outage is removed during upper head re-assembly and outage de-mob for mode change with the Reactor Coolant System at NOP/NOT with a 4 hour hold at conditions prior to beginning the examination following reactor vessel re-assembly. The examination is performed by VT-2 qualified personnel.

FPL performs a visual examination at the beginning of each outage, prior to reactor vessel head disassembly. This examination is performed from the upper cavity elevation utilizing the inspection ports surrounding the vessel head. The inspection is also performed from the incore instrument (ICI) column access doors inside the reactor vessel head shroud during disassembly for evidence of leakage as well as all the accessible CEDMs. In addition to general recording requirements, the procedure requires "evidence of leakage or indeterminate inspections shall be indicated with an asterisk and recorded." Any evidence of leakage is required to be entered in the corrective action program and dispositioned. While this examination does not require VT-2 qualified personnel, typically the personnel utilized are VT-2 qualified.

The class 1 system leakage test VT-2 examination is performed every outage of all CEDMs from the 62' containment elevation and looking down from the platform above the CEDM housings since there is no permanent ladder down into the upper cavity and the temporary access for the outage is removed during upper head re-assembly and outage de-mob for mode change with the Reactor Coolant System at NOP/NOT with a 4 hour hold at conditions prior to beginning the examination following reactor vessel re-assembly. The examination is performed by VT-2 qualified personnel. Pressure boundary through wall leakage is unacceptable and shall be corrected prior to returning the system to service. The source of leakage shall be determined and quantified, if possible, and entered into the corrective action program.

6. Duration of Proposed Alternative

This relief request is applicable to the St. Lucie Unit 1 Fourth Inservice Inspection Interval which began February 11, 2008 and ends February 10, 2018.

7. Precedent

St. Lucie Plant, Unit No. 2 – Relief Request Number 14 – Alternative from examination requirements for the welds in the control element drive mechanism housing (TAC No. MF4341).