UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

June 9, 2016

Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE PLANT - NRC DESIGN BASES INSPECTION (PROGRAMS) REPORT, 05000335/2016010 AND 05000389/2016010

Dear Mr. Nazar:

On, April 29, 2016, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your St. Lucie Plant, Units 1 and 2, and discussed the results of this inspection with you, Mr. Robert Coffey, and other members of your staff. Additional inspection results were discussed with Mr. Rich Wright and other members of your staff on June 2, 2016. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented three findings of very low safety significance (Green) in this report. The findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the St. Lucie Plant.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II; and the NRC Resident Inspector at the St. Lucie Plant.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 50-335, 50-389
License Nos. DPR-67, NPF-16

Enclosure:
Inspection Report 05000335/2016010 and 05000389/2016010,
w/Attachment: Supplementary Information

cc: Distribution via ListServ

accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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ADAMS: ☒ Yes ACCESSION NUMBER:_______________ ☒ SUNSI REVIEW COMPLETE ☒ FORM 665 ATTACHED

OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS	RII:DRS		
SIGNATURE	**TNF1**	**MCG9**	**SXL5**	**JHB1**	**MXL6 FOR**		
NAME	T. Fanelli	M. Greenleaf	S. Herrick	J. Bartley	L. Suggs		
DATE	6 /9 / 16	6 /9 / 16	6 /9/ 16	6 /9/ 16	6 / 8/ 16		
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	050000335, 05000389
License Nos.:	DPR-67, NPF-16
Report Nos.:	05000335/2016010, 05000389/2016010
Licensee:	Florida Power & Light Company (FP&L)
Facility:	St. Lucie Plant, Units 1 and 2
Location:	6501 South Ocean Drive Jensen Beach, FL 34957
Dates:	April 25 – April 29, 2016
Inspectors:	Theodore N. Fanelli, Senior Reactor Inspector (Lead) Sandra Herrick, Reactor Inspector Michael C. Greenleaf, Reactor Inspector
Approved by:	Jonathan H. Bartley, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

IR 05000335/2016-010 and 05000389/2016-010; 04/25/2016 – 04/29/2016; St. Lucie Plant, Units 1 and 2 Design Bases Inspection (Programs).

Three Nuclear Regulatory Commission (NRC) inspectors from Region II conducted this inspection. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, or Red) and determined using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” (SDP) dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, “Aspects Within the Cross-Cutting Areas,” dated December 4, 2014. All violations of NRC requirements were dispositioned in accordance with the NRC’s Enforcement Policy dated February 4, 2015. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 5.

NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green: The inspectors identified a green non-cited violation of Technical Specification (TS) 3.3.3.1 for failing to take the required TS actions after identifying a condition adverse to quality that affected the operability of the containment high range radiation monitors (CHRRMs) (RD-26-40 and RD-26-41). The licensee declared the CHRRMs for both Unit 1 and Unit 2 inoperable and identified alternate methods for assessing emergency action levels, performing core damage assessment and dose assessment. The licensee entered these issues in the corrective action program for resolution as AR2128751 and AR2135780.

  The performance deficiency was determined to be more than minor because it was associated with the Emergency Response Organization Performance attribute of the Emergency Preparedness Cornerstone and adversely affected the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding was evaluated using IMC 0609, Appendix B, “Emergency Preparedness Significance Determination Process.” The finding is of very low safety significance (Green) because the finding affected an EAL that was rendered ineffective such that any Site Area Emergency would not be declared for a particular off-normal event, but because of other EALs, an appropriate declaration could be made in a degraded manner (e.g., delayed). This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance. (Section 1R21.b.1)

- Green: The inspectors identified three examples of a green non-cited violation of Title 10 Code of Federal Regulations (CFR) Part 50.49.e.(5) “aging” for the licensee’s failure to assure conformance with the qualification procedures and methods specified in IEEE 323-1974 “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations” as amended by RG 1.89 “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.” In response to this issue, the licensee’s immediate corrective actions included an immediate determination of operability, in which the licensee concluded that that for the specific examples documented in this violation, the affected components were operable. The licensee entered these issues in the corrective action program for resolution as AR2128753, AR02128366, AR2128755, and AR2135777.

The three performance deficiencies were determined to be more than minor because they were associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, with time in service, significant aging degradation of SSCs increases the likelihood these SSCs could unpredictably fail when called upon to perform their designed safety function. The team used IMC 0609 Attachment 4, “Initial Characterization of Findings,” issued June 19, 2012, and IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the findings were a deficiency affecting the design of a mitigating structure, system, or component (SSC), and the SSC maintained their operability or functionality. This finding was assigned a cross-cutting aspect of H.6 Design Margins in the Human Performance Area because the finding was indicative of current licensee performance and the licensee did not operate and maintain equipment within design margins and margins were not carefully guarded and were changed without a systematic and rigorous process (WP.2). (Section 1R21.b.2)

- <u>Green</u>: The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to verify, justify, and document an activation energy used to determine the thermal lifespan of safety related cable insulation. In response to this issue, the licensee’s immediate corrective actions included an immediate determination of operability, in which the licensee concluded that affected components remained operable. The licensee entered this issue in the corrective action program for resolution as AR2128756.

  The performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, using incorrect activation energies provided erroneous environmental qualification of Class 1E components, which affected the reliability of the acoustic monitor when called upon. The team used IMC 0609 Attachment 4, “Initial Characterization of Findings,” issued June 19, 2012, and IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the findings were a deficiency affecting the design of a mitigating structure, system, or component (SSC), and the SSC maintained their operability or functionality. This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance. (Section 1R21.b.3)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 <u>Component Design Bases Inspection (71111.21N)</u>

a. <u>Inspection Scope</u>

The inspection team performed a pilot inspection conducted as outlined in NRC Inspection Procedure (IP) 71111.21N, Attachment 1, and "Environmental Qualification under 10 CFR 50.49 Programs, Processes, and Procedures." The team assessed Plant St. Lucie's implementation of the site environmental qualification program as required by 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants." The team evaluated whether Plant St. Lucie staff properly maintained the environmental qualification of electrical equipment important to safety throughout plant life, established and maintained required environmental qualification documentation records, and implemented an effective corrective action program to identify and correct environmental qualification related deficiencies.

The inspection included review of environmental qualification program procedures, component environmental qualification files, environmental qualification test records, equipment maintenance and operating history, maintenance and operating procedures, vendor documents, design documents, and calculations. The team interviewed program owners, engineers, maintenance staff, and warehouse staff. The team performed in-plant walkdowns (where accessible) to verify equipment was installed as described in Plant St. Lucie's environmental qualification component documentation files; and that the components were installed in their tested configuration. Additionally, the team performed in-plant walkdowns to determine whether equipment surrounding the environmental qualification component could fail in a manner that could prevent the safety function of the components, and to verify that components located in areas susceptible to a high energy line break were properly evaluated for operation in a harsh environment. The team reviewed and inspected the storage of replacement parts and associated procurement records to verify environmental qualification parts approved for installation in the plant were properly identified and controlled, and that storage and environmental conditions did not adversely affect the components' qualified lives. Documents reviewed for this inspection are listed in the Attachment.

The inspection procedure requires the team to select six to ten components to assess the adequacy of the environmental qualification program. The team selected six components for this inspection. Component samples selected for this inspection are listed below:

- U1 PORV and Block Valves
- U1 FE1402A PORV and Block Valve tail pipe flow sensor (Endevco accelerometer)
- U2 HPSI (High Pressure Safety Injection) pump

- U2 V3495 solenoid valve associated with recirculation to RWST for the Containment Spray PP 2A, Low Pressure Safety Injection, and High Pressure Safety Injection PP 2A.

- U1 V3480 motor operated valve for loop 1A SDC Return

- U2 Radiation Monitors RD-26-40

b. Findings

1. Failure to Comply with TS requirements for Containment High-Range Radiation Monitors (CHRRM)

Introduction: The NRC identified a Green non-cited violation (NCV) of Technical Specification (TS) 3.3.3, “Monitoring Instrumentation,” for failing to address TS required actions due to a known condition adverse to quality, associated with coaxial cabling for the CHRRMs, that affected the operability and reliability of the CHRRMs (RD-26-40 and RD-26-41).

Description: In 1997 and 1998, NRC Information Notice (IN) 97-45 and its Supplement 1 notified the licensee that during certain design basis accidents (DBAs), the CHRRMs are subject to erratic behavior and possible failure from the damage to the cabling system due to high environmental temperatures and steam. The high temperatures and steam could cause thermally induced currents (TIC), turbulences, vibrations, and moisture intrusion. The NRC distributed IN 97-45 in response to issues discovered at San Onofre Nuclear Generating Station (SONGS). To determine the severity of this condition, Southern California Edison and EPRI performed testing on un-aged cables of similar compositions and documented effects of moisture intrusion and TIC. The testing on un-aged cables resulted in blistering of cable jackets, water intrusion, erratic signals, and in some cases complete failure.

In 1997, the licensee issued a condition report (CR 97-0400), documenting their evaluation of the IN and the susceptibility of their CHRRMs to these phenomena. The evaluation determined that the phenomena described in the IN affected their CHRRMs and that they were in nonconformance. The licensee determined that an operability assessment and root cause analysis were required. The operability report contained in the CR identified that the TIC phenomenon could produce large false readings in the range of 6,000 R/h at the onset of a DBA, leading to the possibility of an over-classification of the event. The inspectors noted that the operability report determined that moisture intrusion would not affect the CHRRMs based on a lower DBA temperature for a LOCA. The inspectors determined that the report did not adequately consider the highest temperatures possible for DBAs (i.e. steam line breaks) and its effect on moisture intrusion or the other effects outlined in the IN such as airflow turbulence and electronic noise etc. At the time, the licensee determined that the potential for these errors did not pose an operability concern. In addition to the CR in 1997, the licensee generated a plant management action item (PMAI 97-04-116) to address the issue. The PMAI and CR were closed under the provision that the licensee would use the system as is until the Nuclear Utility Group on Equipment Qualification (NUGEQ) had determined a solution to what they considered an important industry concern. As of the date of this inspection, no further actions were taken by the licensee to resolve this issue.

Technical Specification limited condition for operation 3.3.3.1 required, for each unit, that "with one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6." For Unit 1 Table 3.3-6 ACTION 15, for Unit 2 Table 3.3-6 ACTION 27 stated, "with the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s),and
2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status."

The bases for both units TS 3.3.3.1 "Radiation Monitoring Instrumentation" specified that "the OPERABILITY of the radiation monitoring channels ensures that:

(1) the radiation levels are continually measured in the areas served by the individual channels; and
(2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and
(3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

This capability is consistent with the recommendations of Regulatory Guide 1.97, 'Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,' December 1980 and NUREG-0737, 'Clarification of TMI Action Plan Requirements,' November 1980."

The inspectors determined that during DBAs that caused environmental transients in containment, the CHRRMs were not capable of performing their specified safety function because of the various phenomena identified by the IN and licensee CR. This was due to system capability being degraded to a point where it could not perform with a reasonable expectation of reliability, availability, and capability. The inspectors determined that the cabling system for the CHRRMs was not in conformance with the qualification requirements in 10 CFR 50.49. The inspectors also noted, based upon licensee administrative procedure EN-AA-203-1001, "Operability Determinations/Functionality Assessments," Attachment 4, that the CHRRMs should be classified as inoperable as the phenomena discussed represent new failure modes that will cause a failure of the CHRRMs while operating in their design basis environment. Further, the inspectors noted that the licensee failed to take the required TS actions based on the determination in the 1997 CR and the licensee failed to correct the condition adverse to quality once the nonconformance was identified.

Analysis: The inspectors determined that the licensee's failure to take the required actions for TS 3.3.3.1 was a performance deficiency since the CHRRMs failure could impair licensee efforts to support emergency response activities. The performance deficiency was determined to be more than minor because it was associated with the Emergency Response Organization Performance attribute of the Emergency Preparedness Cornerstone and adversely affected the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding was evaluated using IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process." The finding is of very low safety significance (Green) because the finding affected an EAL that was rendered ineffective such that any Site Area Emergency would not be declared for a particular off-normal event, but because of other EALs, an appropriate declaration could be made in a degraded manner (e.g., delayed). This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance.

Enforcement: Technical Specification 3.3.3.1, "Radiation Monitoring, Limiting Condition for Operation," required, that "with the number of Containment Area – Hi Range Area Monitors OPERABLE Channels less than one channel, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or: 1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and, 2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status." Contrary to the above, since 1997, the licensee failed to restore at least one channel of CHRRM to operable status within 72 hours or initiate the preplanned alternate method of monitoring the appropriate parameter(s), and prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days. For corrective actions, the licensee declared the CHRRMs for both Unit 1 and Unit 2 inoperable and identified alternate methods for assessing emergency action levels, performing core damage assessment and dose assessment. Because this issue is of very low safety significance (Green) and has been entered into the licensee's corrective action program as AR2128751 and AR2135780.this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. This violation is identified as NCV 05000335/2016010-01, 05000389/2016010-01, Failure to comply with TS 3.3.3.1 requirements for CHRRMs.

2. Failure to Implement Qualification Procedures and Methods in Accordance with IEEE 323-1974

Introduction: The inspectors identified three examples of a Green NCV of Title 10 Code of Federal Regulations (CFR) Part 50.49.e.(5) "aging" for the licensee's failure to assure conformance with the qualification procedures and methods specified in the Institute of Electrical and Electronics Engineers (IEEE) 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," as amended by Regulatory Guide (RG) 1.89, "Environmental Qualification (EQ) of Certain Electric Equipment Important to Safety for Nuclear Power Plants."

Description: In 1983 the NRC issued 10 CFR 50.49 “Environmental qualification of electric equipment Important to safety for nuclear power plants,” which was documented in the Federal Register volume 48, number 15. Subsequently in 1984, the NRC revised RG 1.89, which amended and endorsed IEEE 323-1974. The RG in Section D. “Implementation,” specified that conformance with the amended IEEE standard was the acceptable method to the NRC staff for all operating reactors to comply with 10 CFR 50.49 unless an alternate method was submitted to the NRC for approval. The licensee did not provide an acceptable alternate method as specified in RG 1.89 therefore; the inspectors assessed whether the licensee’s EQ program conformed to the amended IEEE 323-1974 and thus complied with 10 CFR 50.49.

Standard IEEE 323-1974 Section 5, “Principles of Qualification,” established that the “principles and procedures for demonstrating the qualification of Class IE equipment include:

(1) Assurance that the severity of the qualification methods equal or exceed the maximum anticipated service requirements and conditions,
(2) Assurance that any extrapolation or inference be justified by allowances for known potential failure modes and the mechanism leading to them,
(3) On-going qualification testing of installed equipment whose qualified life is less than the design life of the equipment,
(4) Documentation files which provide the basis for qualification,
(5) Qualification test data as required for ongoing qualification testing,
(6) Qualification of any interfaces associated with Class 1E equipment.”

Standard 323-1974 Section 6, “Qualification Procedures and Method,” detailed the minimum procedures and methods necessary to meet the principles established in section 5 described above.

Example 1: The licensee used the Arrhenius mathematical model based on previous accelerated thermal aging tests to extrapolate a new thermal life for previously qualified components in accordance with 10 CFR 50.49.e(5), where the thermal qualified life was the time dependent variable. As one example, the licensee documented a Class 1E switch phenolic material having a 71,980-year useful of life at 104˚F inside containment. The inspectors noted that the analysis did not discuss the validation of the input and output data or the uncertainties and confidence limits for the extrapolated thermal life as discussed in quality standards (see Document Pack No. 8770-A-451-14.0). Standard IEEE 323-1974 Section 6.5.4, “Determination of Qualification,” specified that “the qualified life shall be based upon the known limits of extrapolation of the time dependent environmental effects if an accelerated aging test was used to determine the mathematical model.” The inspectors determined that the licensee did not base their thermal qualified life on the known limits of extrapolation of the Arrhenius model as specified in IEEE 323-1974. Judgements about the appearance of the output data, whether it appeared excessive (as above) or if it appeared reasonable, did not verify whether adequate margins existed as required by 10 CFR 50.49.e(8). The verification of margins required bounding the analysis with the known limits of extrapolation to support the analytical assumptions and conclusions as specified by IEEE 323 and required by 10 CFR 50.49.f. Without adequate verified margin, there is no reasonable expectation that the components can perform their safety function during DBAs. Many different quality standards discuss the known limits of extrapolation related to the Arrhenius model, and

IEEE 323-1974 specifically identified IEEE 101-1972, “IEEE Guide for the Statistical Analysis of Thermal Life Test Data,” as one of those known quality standards (ref NUREG 588 page II-46, resolution to comment 84). Further, the subcomponent-level service life projections for environmental qualification must prove available margins for harsh environments and the operational capability must be demonstrated with tests that included operational aging and a design basis tests (LOCA/MSLB as applicable). In addition, the acceleration for the aging part of the test must be based on the activation energy of the most limiting (most aging sensitive) element of the component, which may vary with service temperature and operational configuration (ref NUREG 588 page II-47). In every sample assessed by the inspectors, the licensee based qualified life, for equipment covered under 10 CFR 50.49, on these extrapolations that did not consider the known limits of extrapolation, the required margins, and demonstrate qualified life with supporting design bases qualification tests. Therefore, the environmental qualification for these components have not been established in that the licensee failed to ensure the reliability, capability, and availability of Class 1E components during DBAs.

Example 2: The licensee’s program for performing EQ analysis did not account for non-thermal aging effects that occur under normal service life conditions as required by 10 CFR 50.49.e(5) for the six samples examined for ongoing qualification. Standard IEEE 323-1974 Section 5 specified the need to ascertain non-thermal aging effects. It stated “The principles and procedures for demonstrating the qualification of Class 1E equipment include; assurance that the severity of the qualification methods equal or exceed the maximum anticipated service requirements and conditions, and assurance that any extrapolation or inference be justified by allowances for known potential failure modes and the mechanism leading to them.” Standard IEEE 323-1974 Section 6.5.3.1, “Failure Modes,” stated, in part, “the modes of failure produced under intensified or accelerated -environmental or other influences shall be the same as those predicted under the required service conditions.” The inspectors identified significant non-thermal degradation mechanisms in and around the observed samples that may become apparent when the equipment is subjected to DBAs (ref NUREG 588 page II-47, resolution to comment 88). These included vibration, electrical transients, high humidity, corrosive environments (i.e. salt air) and normal handling of equipment. In addition, the inspectors noted that, for original qualification, the tests demonstrated that the components were susceptible to non-thermal degradation mechanisms. In the original qualification documents, (see Document Pack No. 8770-A-451-22.0), the tests qualified the cables for 7.8 years and simulated thermal-cycling, low-level vibration, high humidity, and transient voltages in addition to the thermal aging. After the 7.8 years of qualified life, during the LOCA test the qualification records identified that Brand Rex coaxial cable insulation and the BNC connectors Kel-F insulation deteriorated to a hardened and brittle compound compromising the jackets and insulations due to these aging mechanisms (ref NUREG 588 page II-48). For additional qualified life, the licensee did not examine additional exposure to the non-thermal aging mechanisms identified in the original reports or the ones identified in the immediate area around the observed components by the inspectors. Additional exposure to these aging mechanisms could have detrimental effects on these components during DBAs. The inspectors determined that the licensee did not consider non-thermal aging factors and provide proof that the degradation affects were acceptable, which did not ensure component reliability, capability, and availability during DBAs.

Example 3: The licensee's ongoing qualification program did not re-qualify or replace subcomponents whose qualified life was less than that of the assemblies it was a part of as specified by IEEE 323-1974 Sections 5, "Principles of Qualification," and 6.6, "On-Going Qualification." Section 5 specified, in part, "principles and procedures for demonstrating the qualification of Class 1E equipment include on-going qualification testing of installed equipment whose qualified life is less than the design life of the equipment." Section 6.6 specified, in part, that "should on-going qualification methods demonstrate that the qualified life is less than the required life, a periodic replacement plan shall be instituted." Target Rock specifically limited the qualified life of certain subcomponents of the solenoid-operated valve V3495 to 20 years (e.g. the solenoid coil, limit switches, and elastomer parts). The licensee extended the qualified life of the valve assemblies from 40 to 60 years but failed to consider the components that had a shorter life than the life of the valve assemblies even though Target Rock identified them in the qualification report. The inspectors determined that operating these safety significant valves beyond the qualified life of essential subcomponents did not ensure their reliability, capability, and availability during design basis accidents.

Analysis: The failure to implement qualification procedures and methods, as described in the three examples, in accordance with IEEE 323-1974 were performance deficiencies. The three performance deficiencies were determined to be more than minor because they were associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, with time in service, significant aging degradation of SSCs increases the likelihood these SSCs could unpredictably fail when called upon to perform their designed safety function. The team used IMC 0609 Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, and IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the findings were a deficiency affecting the design of a mitigating structure, system, or component (SSC), and the SSC maintained their operability or functionality. This finding was assigned a cross-cutting aspect of H.6 Design Margins in the Human Performance Area because the finding was indicative of current licensee performance and the licensee did not operate and maintain equipment within design margins and margins were not carefully guarded and were changed without a systematic and rigorous processes (WP.2).

Enforcement: Title 10 CFR Part 50.49(e)(5), Aging, stated, in part, "Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life." Contrary to the above, since February 18, 2013, the licensee failed to consider all significant types of degradation which can have an effect on the functional capability of the equipment and replace or refurbish equipment at the end of its designated life unless ongoing qualification demonstrated that the item had additional life. Specifically, the licensee did not adequately consider the impact of the known limits of Arrhenius extrapolation, non-Arrhenius aging mechanisms, and subcomponents with a shorter life than the required life. In response to this issue, the licensee's immediate corrective actions included an immediate determination of operability, in which the licensee concluded that that for the specific examples documented in this violation, the affected components were operable. The licensee entered these issues in the corrective action program for resolution as AR2128753, AR02128366, AR2128755, and AR2135777. This violation is being treated as an NCV consistent with section 2.3.2 of the Enforcement Policy. This violation is identified as NCV 05000335/2016010-02 and 05000389/2016010-02, Failure to Implement Qualification Procedures and Methods in Accordance with IEEE 323-1974.

3. Failure to Define, Justify, and Document Activation Energies

- Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to verify, justify, and document an activation energy used to determine the thermal lifespan of safety related cable insulation.

  Description: The licensee obtained a letter dated July 18, 1986, from a vendor to the Tennessee Valley Authority (TVA) addressing the activation energy specific to tests performed on TVA ordered cabling (it was 1.47eV). The letter did not state that it was applicable to the cables used at St. Lucie. In addition, the letter specified that the test equipment used could not be calibrated specifically for the activation energy determination method for which the test equipment was credited in the letter. The licensee's original qualification documentation for their TEC 2273-C2 cabling components, which are used for the pressurizer power operated relief valve acoustic monitor, stated the activation energy was 0.65 eV and later it was stated to be 1.07eV. Based on the TVA letter, the licensee changed the qualified activation energy from 1.07eV to 1.47eV. Subsequently, the licensee used this activation energy (1.47eV) to determine the qualified life of Class 1E components without performing verification and validation that it was appropriate to use in their design. Regulatory Guide 1.89, in regulatory position C.5.c, stated, that "the basis for the aging acceleration rate and activation energies used during qualification should be defined, justified, and documented." Originally, the service life of the cables in question was determined to be ~7.8 years at 0.65eV then ~29 years at 1.07eV and the cables have now been in use well beyond these qualified lives. The inspectors determined that the licensee used incorrect activation energies and failed to define, justify, and document it. Using safety related cables beyond their correct qualified life did not ensure the reliability, capability, and availability of the acoustic monitoring system during design basis accidents.

  Analysis: The failures to define, justify, and document the basis for activation energies used during ongoing qualification and how it was applicable to Plant Saint Lucie, as specified by RG 1.89 was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Design Control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, using incorrect activation energies provided erroneous environmental qualification of Class 1E components, which affected the reliability of the acoustic monitor when called upon. The team used IMC 0609 Attachment 4, "Initial Characterization of Findings," issued June 19. 2012 and IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the findings were a deficiency affecting the design of a mitigating structure, system, or component (SSC), and the SSC maintained their operability or functionality. This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," stated, in part, that "design control measures shall provide for verifying or checking the adequacy of design." Licensee "Equipment Qualification Documentation Package Drawing Number 8770-A-451-22.0," Revision 6, stated, in part, "Brand-Rex letter, dated July 18, 1986 (Section 3.3.8), indicates that a newly calculated activation energy for Hypalon is 1.47eV." Contrary to the above, since September 1986, the licensee's design control measures failed to provide for verifying or checking the adequacy of design of the TEC Valve Flow Monitoring System cable assemblies. Specifically, the licensee failed to verify or check the adequacy of activation energies for TEC-2273-C2 cabling systems before using the newly calculated activation energies in ongoing qualifications. In response to this issue, the licensee's immediate corrective actions included an immediate determination of operability, in which the licensee concluded that affected components remained operable. The licensee entered this into their corrective action programs as AR2128756. This violation is being treated as an NCV consistent with section 2.3.2 of the Enforcement Policy. This violation is identified as NCV 05000335/2016010-03 and 05000389/2016010-03, Failure to Define, Justify, and Document Activation Energies.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On April 29, 2016, the team presented the inspection results to you and other members of the licensee's staff. On June 2, 2016, a re-exit meeting was conducted via teleconference to present the final inspection results to Mr. Rich Wright and other members of the licensee's staff. Proprietary information that was reviewed during the inspection was returned to the licensee or destroyed in accordance with prescribed controls.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
G. Arntson, Principal Nuclear Engineer, ERRT
L. Berry, Principal Nuclear Engineer, Licensing
S. Catron, Nuclear Fleet Licensing CFAM
D. Cecchett, Principal Nuclear Engineer, Licensing
R. Coffey, Nuclear Plant General Manager
S. Cornell, Principal Nuclear Engineer, Mechanical Design
C. Costanzo, Site Vice President
K. Fleischer, Consultant
W. Laframboise, Nuclear Engineering Site Manager
E. McCool, Nuclear Fleet Engineering
W. Parks, Nuclear Operations Site Director
M. Snyder, Nuclear Licensing Manager
A. Terezakis, Nuclear Operations Supervisor
M. Wolaver, Engineering Manager

NRC personnel:
T. Morrissey, Senior Resident Inspector, Division of Reactor Projects, St. Lucie Resident Office
R. Reyes, Resident Inspector, Division of Reactor Projects, St. Lucie Resident Office

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened & Closed

05000335, 389/2016010-01	NCV	Failure to comply with TS requirements for CHRRMs (Section 1R21.b.1)
05000335, 389/2016010-02	NCV	Failure to Implement Qualification Procedures and Methods in Accordance with IEEE 323-1974 (Section 1R21.b.2)
05000335, 389/2016010-03	NCV	Failure to Define, Justify, and Document Activation (Section 1R21.b.3)

LIST OF DOCUMENTS REVIEWED

Condition Reports Written as a Result of the Inspection
02128612, EQ doc-pac Did Not Specify Most Limiting Component
02128751, Failed to resolve a CAQ from a 1997 Information Notice for CHRRM (Radiation Detector 26-40.
02128663, EQ CDBI Identified CHRRM Indication Error During DBA
02128756, Failure to define and justify activation energy for Pressurizer Acoustic monitors.
02128755, Failure to use non-Arrhenius methodology for aging mechanics.
02128753, Failure to consistently use Quality Standards IEEE 1 and 101 when applying Arrhenius model to validate input and output calculations.
02128570, During NRC EQ inspection doc-pac 2998-A-451-35.6 R13 showed typographical error on page 79 of 90.
02128366, During the NRC EQ inspection, it was identified that the most limiting component was not properly selected in the EQ Doc pack for the following valves.
02128301, 2A HPSI MOTOR POST-EPU TEMP INCREASE / Post-EPU temperature increase in the Unit 2 ECCS Room has not been incorporated into the EQ Doc Pac 4.8 for the 2A HPSI Pump motor.

Procedures
0-PME-80.07, Preventative Maintenance of Environmentally Qualified Limitorque Motor
0-PMI-99.03, Instrument and Control Department EQ Maintenance Tabs, Rev. 7
1-EOP-3, Loss of Coolant Accident, Rev. 31
1-PMI-99.01, I&C Department EQ Maintenance Instructions, Rev. 1
2-OSP-03.01A, 2A HPSI Pump Safeguards Full Flow Test, Rev. 14, completed 10/11/2015
2-OSP-03.01B, 2B HPSI Pump Safeguards Full Flow Test, Rev. 14, completed 3/9/2014
2-PMI-99.01, Instrumentation and Control Department EQ Maintenance Instructions, Rev. 4
EN-AA-203-1001, Operability Determinations / Functionality Assessments, Rev. 20
OP-01-0010125A, Surveillance Data Sheets, Data Sheet 24, Valve Testing Procedures, Rev. 165, completed 12/31/2015
Operated Valve Actuators, Rev. 9
QI-07-NSC-1, Warehouse Receipt, Storage, Issuance and Maintenance, Rev. 16

Drawings
8770-A-451-1000, St. Lucie Plant Unit No 1 Equipment Qualification Documentation Package, Rev. 12
8770-B-327 SH 83, Charge Converter Wiring, Rev.13
St. Lucie Plant Unit No 2 Equipment Qualification Documentation Package

Calculations
0113, Post LOCA Reactor Auxiliary Building ECCS Pump Area Temperature Transient Analysis, Rev. 0
2998-B-327 SH443 R8, Containment High Range Rad Monitors Detector No RD-26-40, RD-26 41, Rev. 8
CN-OA-08-46, LOCA Containment Pressure/Temperature Analysis for St. Lucie Unit 1 EPU, Rev. 1
CN-OA-08-39, St. Lucie Unit 2 EPU Containment LOCA Pressure and Temperature Analysis, Rev. 2
CN-OA-09-16, Lucie Unit 2 Containment Main Steam Line Break Analysis for EPU, Rev. 2

Self-Assessment Reports
SAQH 2106593 Quick Hit/Department Assessment Report 2106593, dated 03/22/2016

Work Orders

38022804 01	38022804 22	40200509 01
38022804 08	38022805 01	40282737 01
38022804 21	40178791 08	38022804 01
38022804 22	40200509 01	38022804 08
38022805 01	40282737 01	38022804 21
40178791 08	38022804 01	38022804 22
40200509 01	38022804 08	38022805 01
40282737 01	38022804 21	40178791 08
38022804 01	38022804 22	40200509 01
38022804 08	38022805 01	40282737 01
38022804 21	40178791 08	

Miscellaneous Documents
2998-15891, Digital High Range Radiation Monitors, Rev. 2
2998-23779, Instruction Leaflet for Medium and Large Induction Motors, Rev. 0
2998-5779, CE-15 I/M Sol Valves V-3495 & 3496 Target Rock, Rev. 1
2998-6738, HPSI Pumps, R8
2998-A-451-16.2, 3M Corporation Splices, Rev. 8
2998-A-451-33.0, United Controls International Silicone Tape P/N UCI-003XS, Rev. 2
2998-A-451-35.6, Equipment Qualification Documentation Package Target Rock Solenoid
2998-A-451-4.4, General Electric HPSI Pump Motor, Rev. 5
2998-A-451-4.8, Homewood Energy Services HPSI Pump Motor, Rev. 0
2998-A-451-6.3, Equipment Qualification Document Package The Rockbestos Company
2998-A-451-8.3, Equipment Qualification Documentation Package General Atomic Radiation
600198, "Test of Limitorque Valve Operator to Meet General Requirements of an Electric Valve
8770-6251, Velan Valves and Maintenance Instructions (PI-1184-N), R28
8770-8387, I/M Power Operated Relief Valves V-1402 & V-1404, R5
8770-A-451-14.0, Limitorque Corporation Motor Operators, Rev. 16
8770-A-451-22.0, St Lucie Plant Unit No 1 Equipment Qualification Documentation Package, Rev. 6
8770-A-451-22.0, St Lucie Plant Unit No 1 Equipment Qualification Documentation Package, Rev. 8
8770-9605, TEC Model 914 Instruction Manual, Rev. 1
Actuator in Nuclear Reactor Containment Environment," Jan 2, 1969.
B-0058, "Limitorque Valve Actuator Qualification for Nuclear power Station Service"
E-254-960, Test Report Class 1E Design Qualifications Testing of Analog High Range Radiation
Electrical Cable, Rev. 6
FPL L-92-288 from GR Madden to NRC, Generic Letter 90-06 Supplemental Response, dated 10-9-92
GEK 50401, Topical Report IEEE 323 Class 1E Horizontal Induction Motor Model 5N811052C57 Horizontal Class B Insulated, dated October, 1977
GL 90-06, Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low-Temperature Over Pressure Protection for Light-Water Reactors" Pursuant to 10 CFR 50.54(f)
HES-QA-PAC-01, Homewood Energy Services (HES), HPSI Pump Motor QA Package, Rev. 0
Monitor (RD-23, RP-2C, RP-23, and RP-20-01, Rev. 1

NKO-2H08-WW, Qualification Report of Homewood Energy Form Wound Class H Insulation System for Medium and Low Voltage Motors, dated 3/21/2008
NP-1558 Review of Equipment Aging Theory and Technology, Rev. A
Requirements of IEEE 323-1974, 344-1975 and 382-1972 Standards, Rev. G
Rock Corporation 1” Solenoid Valve; Model 77CC-001 (Modified Per SK 4017) per
Rock Corporation Model 86Z561-001 Solenoid Operated Globe Valves in Accordance with Standard Case IV Conditions (Modified) IEE 382 – 1980, Rev. 0
TR-4519, Qualification Extension Analysis Report for Environmental Qualification of the Target
TRC-2375-G, Qualification Test Report Aging, Seismic, and Accident Simulation Test of Target Valves, Rev. 13