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

April 17, 2015

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

SUBJECT: ST. LUCIE PLANT- U.S. NUCLEAR REGULATORY COMMISSION EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS INSPECTION REPORT 05000335/2015007 AND 05000389/2015007

Dear Mr. Nazar:

On March 6, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Saint Lucie Plant, Units 1 and 2, and discussed the results of this inspection with Mr. R. Coffey and other members of your staff. Additional inspection results were discussed with Mr. E. Katzman of your staff on April 2, 2015. Inspectors documented the results of this inspection in the enclosed inspection report.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission’s rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The NRC inspectors documented two findings of very low safety significance (Green) in this report. These findings involved violations of NRC requirements; one of these violations was determined to be Severity Level IV under the traditional enforcement process. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the 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 Saint Lucie plant.

If you disagree with a cross-cutting aspect assignment 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 Saint 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 Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jonathan Bartley, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 05000335, 05000389
License Nos. DPR-67, NPF-16

Enclosure:
IR 05000335/2015007 and 05000389/2015007
w/Attachment: Supplementary Information

cc: Distribution via Listserv

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 Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

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Jonathan Bartley, Chief
Engineering Branch 1
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Docket Nos. 05000335, 05000389
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ADAMS: ☒ Yes ACCESSION NUMBER: **ML15110A111** ☒ SUNSI REVIEW COMPLETE ☒ FORM 665 ATTACHED

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NAME	G. OTTENBERG	M. RILEY	S. HERRICK	M. ORR	J. BARTLEY	E. STAMM	
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.	50-335, 50-389
License Nos.	DPR-67, NPF-16
Report Nos.	05000335/2015007, 05000389/2015007
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:	February 9, 2015 – March 6, 2015
Inspectors:	G. Ottenberg, Senior Reactor Inspector (Team Leader) M. Orr, Reactor Inspector, NRC Region I M. Riley, Reactor Inspector S. Herrick, Project Engineer (Trainee)
Approved by:	Jonathan H. Bartley, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

Inspection Report (IR) 05000335/2015007, 05000389/2015007 02/09/2015-03/06/2015; Saint Lucie Plant, Units 1 and 2; NRC Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications.

This report covers a two-week, on-site inspection by four regional inspectors. Two non-cited violations were identified. The significance of inspection findings is indicated by their color (Green, White, Yellow, Red) using the Nuclear Regulatory Commission (NRC) Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross Cutting Areas" dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated January 28, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Rev. 5, dated February 2014.

<u>NRC Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- <u>Severity Level IV</u>. An NRC-identified severity level IV (SL IV) non-cited violation (NCV) of 10 CFR 50.59(c)(2)(ii) and an associated finding of very low safety significance (Green) was identified for the licensee's failure to obtain a license amendment prior to implementing a change to the Unit 1 reactor protective system (RPS). The failure to obtain a license amendment for the change resulted in the implementation of a modification that did not conform with the licensee's current licensing basis. The licensee's failure to obtain NRC approval prior to implementing the change to the Unit 1 RPS was determined to impact the regulatory process because the change required NRC review and approval prior to implementation. The licensee entered this issue into their corrective action program as action requests (ARs) 2029652 and 2030820, planned to restore the RPS configuration into conformance, and performed a prompt operability determination which concluded that there was a reasonable expectation that the RPS channels remained operable and could perform their required design basis functions.

  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, the modification did not ensure the reliability of the RPS to respond to a design basis event because the design requirements for physical separation of RPS channels A and C were not met and resulted in a condition where revision or rework would be required to resolve the physical separation concerns. The team determined the finding to be of very low safety significance (Green) because the finding did not affect a single RPS trip signal to initiate a reactor scram and the function of other redundant trips or diverse methods of reactor shutdown, did not involve control manipulations that unintentionally added positive reactivity, and did not result in a mismanagement of reactivity by operators. The traditional enforcement violation was evaluated using the NRC Enforcement Policy dated January 28, 2013, and revised February 4, 2015. The inspectors determined the violation was SL IV per Section 6.1.d.2 because the associated finding was evaluated by the SDP as having very low

safety significance (i.e., Green). The inspectors determined the finding was indicative of present licensee performance and was associated with the cross-cutting aspect of change management, in the area of human performance, because the licensee did not use a systematic process for evaluating and implementing a change such that nuclear safety remained the overriding priority. [H.3] (Section 1R17)

Green. An NRC-identified non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control, was identified for the licensee's failure to assure that design basis assumptions for steam generator blowdown (SGBD) flow rate were translated into procedural guidance. Specifically, procedures 1-NOP-23.02 and 1-AOP-09.03 for Unit 1, and 2-NOP-23.02 and 2-AOP-09.03 for Unit 2, allowed SGBD flow rates significantly in excess of the assumed values in non-loss of coolant accident (LOCA) event analyses. The licensee entered the issue into their corrective action program as action requests (ARs) 2030177, 2031217, and 2031218. The licensee's immediate corrective actions included performing a functionality assessment of the SGBD systems for both units, which included; re-performing the event analyses, issuing an operations department night order to temporarily provide operators appropriate direction for limiting the SGBD system flow, and plans to update the analyses of record, plant procedures, and the UFSAR with new system limitations.

The performance deficiency was determined to be more than minor because it affected the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not ensure the capability of the secondary side heat removal systems to respond to design basis non-LOCA events because analysis assumptions were not translated into procedural limitations for the SGBD system. The inspectors determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component (SSC), and the SSC maintained its operability or functionality. The inspectors determined that the issue was indicative of present licensee performance because the analyses were performed in 2013. The finding was associated with the cross-cutting aspect of design margins, in the area of human performance, because the organization did not operate and maintain equipment within design margins. [H.6] (Section 1R17)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, Experiments and Permanent Plant Modifications (71111.17T)

a. Inspection Scope

Evaluations of Changes, Tests, and Experiments: The team reviewed seven safety evaluations performed pursuant to Title 10, *Code of Federal Regulations* (CFR) 50.59, "Changes, tests, and experiments," to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The team also reviewed 14 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. The team reviewed these documents to determine if:

- the changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- the safety issues requiring the changes, tests or experiments were resolved;
- the licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- the design and licensing basis documentation used to support the change was updated to reflect the change.

The team used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Rev. 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000.

Permanent Plant Modifications: The team reviewed nine permanent plant modifications that had been installed in the plant during the last three years. The modifications reviewed are listed below:

- Item Equivalency Evaluation PSL 353568, Valve, Solenoid, with Inconel 718 Upper Seat (Item 1K) and Ball (Item 1C), Rev. 0
- Engineering Change (EC) 235953, Unit 1 MV-07-2A/B Margin Improvement, Rev. 0
- PCM 10045 (EC246557), Power Uprate – Hydrogen Purge and Containment Pressure Control System, Rev. 7
- EC 250013, Unit 1 Containment Spray Pump Flow Limitation, Rev. 5
- EC 280987, Replace Willmar Relay (VMR/954) in the 2A EDG Control Panel, Rev. 0
- EC 281181, Update Breaker Settings in 480V MCC 2A5 for 2HVE-8A , Rev. 0
- EC 271287, Diesel Oil Storage Tank Operating Margin, Rev. 2
- EC 275227, Emergency Diesel Generator 1B1 Cooling Fan Blade Repair, Rev. 2
- Commercial Grade Dedication (CGD) 435722, 1/4 - 20 ASTM 108, Zinc Plated Channel Nut with Spring, Rev. 0

The modifications were selected based upon risk significance, safety significance, and complexity. The team reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;
- the changes were in accordance with the specified design requirements;
- the procedures and training plans affected by the modification had been adequately updated;
- the test documentation as required by the applicable test programs had been updated; and
- post-modification testing adequately verified system operability and/or functionality.

The team also used applicable industry standards to evaluate acceptability of the modifications and performed walkdowns of accessible portions of the modifications. Documents reviewed are listed in the Attachment.

b. Findings

1. Failure to Submit a License Amendment Request for Unit 1 RPS

Introduction: The inspectors identified a severity level IV (SL IV) non-cited violation (NCV) of 10 CFR 50.59, “Changes, Tests, and Experiments,” and associated Green finding for the licensee’s failure to obtain a license amendment prior to implementing a change to the Unit 1 reactor protective system (RPS). The failure to obtain a license amendment for the change resulted in the implementation of a modification that did not conform with the licensee’s current licensing basis as described in the Updated Final Safety Analysis Report (UFSAR) and IEEE 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations.”

Description: The RPS is designed to assure adequate protection of the fuel, fuel cladding, and reactor coolant pressure boundary during anticipated operational occurrences. The nuclear instrumentation consists of ten channels of instrumentation to monitor neutron flux, which includes four sets of wide range logarithmic channels, four sets of power range safety channels, and two sets of power range control channels. On February 2, 2015, the Unit 1 RPS Channel C bistables tripped due to a failure of linear power range detector #7. Technical Specification (TS) limiting conditions for operation (LCO) 3.3.1.1 requires an inoperable linear power range channel to be restored to an operable status or be placed in trip status after 48 hours. On February 4, 2015, the licensee performed modification EC 283213, “Substitute Power Range Control Channel 9 for Linear Power Range Channel MC,” to provide an alternate nuclear instrumentation system excore detector arrangement that substituted power range control channel CC1 (Channel 9) for linear power range channel C (Channel MC). The modification was done to restore RPS Channel C to an operable status. Implementation of this modification resulted in RPS Channel A and C cables being routed in the same cable raceways for part of their route. This change in cable routing resulted in a lack of physical separation between the redundant RPS channel cables and also resulted in a departure from the licensee’s current design and licensing basis as described in the UFSAR and IEEE 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations,” which represented a non-compliance with 10 CFR 50.55a(h)(2).

UFSAR Section 7.2.1 stated that the instrumentation, function, and operation of the RPS conform to the specific requirements in IEEE 279-1971. UFSAR Section 7.2.2.2,

"Conformance to IEEE-279," described how the specific requirements were satisfied for Section 4 of IEEE 279-1971. Section 4.6 of IEEE 279-1971, "Channel Independence," stated that "channels that provide signals for the same protective function shall be independent and physically separated to accomplish decoupling of the effects of unsafe environmental factors, electrical transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction." UFSAR Section 7.2.2.2.6, which described how the licensee conforms to Section 4.6 of IEEE 279-1971, stated, in part, that "the routing of cables from protective system transmitters is arranged so that the cables are separated from each other and from power cabling to minimize the likelihood of common event failures."

The licensee identified during the implementation of modification EC 283213 that the change did not meet the physical separation requirements for the RPS A and C cables as described in the UFSAR; however, the licensee determined that a 10 CFR 50.59 screening for EC 283213 was not necessary since the change was fully bounded by a screening completed in 2013. The inspectors noted that the modification completed in 2013 replaced RPS Channel B linear power range excore detector with control channel #2 (CC2); however, the modification did not result in two redundant RPS channel cables in the same raceway as was done in EC 283213. The inspectors determined that the 10 CFR 50.59 screen completed in 2013 did not fully bound EC 283213 and that a 10 CFR 50.59 screen was necessary according to the licensee's applicability determination form. The inspectors further concluded that not meeting the physical separation criteria in IEEE 279-1971 and the UFSAR for the RPS Channel A and C cables was an adverse effect which would have required a 10 CFR 50.59 evaluation.

Procedure EN-AA-203-1202, "10 CFR 50.59 Evaluations," Rev. 0, described the licensee's process for completing 10 CFR 50.59 evaluations. This procedure stated that 10 CFR 50.59 evaluations would be completed using the guidance contained in NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Rev. 1. Section 4.3.2 of NEI 96-07 stated, "although this criterion allows minimal increases, licensees must still meet applicable regulatory requirements and other acceptance criteria to which they are committed (such as contained in regulatory guides and nationally recognized industry consensus standards, e.g., the ASME B&PV Code and IEEE standards)." Example six of this section also stated that changes that reduce system/equipment redundancy, diversity, separation or independence constitute more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR and would require NRC approval prior to implementation. The inspectors determined that the licensee should have obtained a license amendment prior to implementing modification EC 283213.

The licensee entered this issue into their corrective action program (CAP) as action requests (ARs) 2029652 and 2030820 and performed a prompt operability determination to verify that the RPS channels could perform their required design basis functions. The licensee reviewed postulated internal missiles in the reactor auxiliary building, external hazards, Appendix R fires, environmental qualification, and electrical transients and determined there was a reasonable expectation that the RPS channels remained operable and could perform their required design basis functions.

<u>Analysis</u>: The licensee's failure to obtain a license amendment, as required by 10 CFR 50.59(c)(2), for a change that did not conform with the licensee's current licensing basis as described in the UFSAR and IEEE 279-1971, 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, the modification did not ensure the reliability of the RPS to respond to a design basis event because the design requirements for physical separation of RPS channels A and C were not met and resulted in a condition where revision or rework would be required to resolve the physical separation concerns. Additionally, the licensee's failure to obtain NRC approval prior to implementing the change to the Unit 1 RPS was determined to impact the regulatory process because the change required NRC review and approval prior to implementation. Specifically, 10 CFR 50.59(c)(2)(ii) required, in part, that the licensee obtain a license amendment prior to implementing a proposed change if the change would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

This violation is associated with a finding that has been evaluated by the SDP and communicated with an SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider the regulatory process impact. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

The finding was evaluated using IMC 0609, Att. 4, "Initial Characterization of Findings," issued June 19, 2012, for Mitigating Systems, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined to be of very low safety significance (Green) because the finding did not affect a single RPS trip signal to initiate a reactor scram and the function of other redundant trips or diverse methods of reactor shutdown, did not involve control manipulations that unintentionally added positive reactivity, and did not result in a mismanagement of reactivity by operator(s). The traditional enforcement violation was evaluated using the NRC Enforcement Policy dated January 28, 2013, and revised February 4, 2015. The inspectors determined the violation was SL IV per Section 6.1.d.2 because the associated finding was evaluated by the SDP as having very low safety significance (i.e., Green).

The inspectors determined the finding was indicative of present licensee performance and was associated with the cross-cutting aspect of change management, in the area of human performance, because the licensee did not use a systematic process for evaluating and implementing a change so that nuclear safety remained the overriding priority. Specifically, the licensee determined that a 10 CFR 50.59 screen was not needed for modification EC 283213, thus precluding a 10 CFR 50.59 evaluation and review of NEI 96-07 which would have directed the licensee to obtain a license amendment before implementing the modification. [H.3]

<u>Enforcement</u>: Title 10 CFR Part 50.59(c)(2)(ii) required, in part, that the licensee shall obtain a license amendment pursuant to 10 CFR Part 50.90 prior to implementing a proposed change if the change would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. Contrary to the above, since February 4, 2015, the licensee failed to obtain a license amendment prior to implementing a change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of the Unit 1 RPS. Specifically, the licensee reduced channel separation between RPS Channels A and C which resulted in a

more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety according to NEI 96-07 guidance. Additionally, 10 CFR Part 50.55a(h)(2), “Protection Systems,” required, in part, that for nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis. UFSAR Section 7.2.1 stated that the instrumentation, function, and operation of the RPS conform to the specific requirements in Section 4 of IEEE 279-1971. Section 4.6 of IEEE 279-1971 stated, in part, that channels that provide signals for the same protective function shall be independent and physically separated. Contrary to the above, since February 4, 2015, the licensee failed to provide physical separation between RPS Channel A and C cables. The lack of channel separation between RPS Channels A and C did not ensure the reliability of the channels to respond to a design basis event and resulted in a condition where revision or rework would be required to resolve the physical separation concerns. The licensee entered the issue into their CAP, planned to restore the RPS configuration into conformance, and performed an operability determination. The inspectors reviewed the licensee’s operability determination, which concluded that there was a reasonable expectation that the RPS channels remained operable and could perform their required design basis functions. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s CAP as ARs 2029652 and 2030820. (NCV 05000335/2015007-01, Failure to Submit a License Amendment Request for Unit 1 RPS)

2. <u>Failure to Establish Appropriate Procedural Limitations to Prevent Exceeding Non-LOCA Event Analysis Assumptions for Steam Generator Blowdown Flow Rate</u>

<u>Introduction</u>: The inspectors identified a Green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the licensee’s failure to assure that design basis assumptions for steam generator blowdown (SGBD) flow rate were translated into procedural guidance. Specifically, procedures, 1-NOP-23.02, and 1-AOP-09.03 for Unit 1, and 2-NOP-23.02 and 2-AOP-09.03 for Unit 2, allowed SGBD flow rates significantly in excess of the assumed values in non-loss of coolant accident (LOCA) event analyses described in chapters 10 and 15 of the station’s UFSAR.

<u>Description</u>: The inspectors reviewed the SGBD system normal operating procedures 1(2)-NOP-23.02 and the abnormal operating procedures for secondary side chemistry, 1(2)-AOP-09.03. The inspectors identified that the licensee did not include procedural limitations for the SGBD system that corresponded to the assumptions in event analyses described in chapters 10 and 15 of the Unit 1 and Unit 2 UFSARs. Specifically, the SGBD system procedures 1(2)-NOP-23.02 and 1(2)-AOP-09.03, allowed the SGBD systems to be operated at values up to maximum blowdown flow rate, which corresponded to a flow rate of approximately 120 gallons per minute (gpm) per steam generator. The team identified that the SGBD systems had been operated at the allowable flow rates during performance of these procedures. The SGBD system flow rate was an input to non-LOCA event analyses for events including a loss of normal feedwater (LONF), feedwater line break (FLB), and station blackout (SBO). These event analyses assumed SGBD system flow rates of 50 to 65 gpm per steam generator, depending on the specific event. Excessive SGBD flow rates during these events would affect the ability to remove core decay heat using the secondary side heat removal systems, because auxiliary feedwater flow to the steam generators could be diverted through the SGBD line which connects to the steam generator.

The inspectors noted that the analyses for the LONF, FLB, and SBO events were updated for extended power uprate and were issued on March 27, 2013, for Unit 1 and June 9, 2010, and November 5, 2013, for Unit 2. Following the analysis revision, 1(2)-NOP-23.02 and 1(2)-AOP-09.03 were not updated to restrict SGBD flow to within that assumed in the analyses. Following the inspector's identification of the issue, the licensee re-performed the event analyses, and confirmed acceptance criteria could still be met at the SGBD flow rates that were allowed by their procedures.

Analysis: The licensee's failure to assure that design basis assumptions in accident analyses were correctly translated into procedural guidance as required by 10 CFR Part 50, Appendix B, Criterion III, was a performance deficiency. The performance deficiency was determined to be more than minor because it affected the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not ensure the capability of the secondary side heat removal systems to respond to design basis non-LOCA events because analysis assumptions were not translated into procedural limitations for the SGBD system. The inspectors used IMC 0609, Att. 4, "Initial Characterization of Findings," issued June 19, 2012, for Mitigating Systems, and IMC 0609, App. 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 finding was a deficiency affecting the design or qualification of a mitigating SSC, and the SSC maintained its operability or functionality.

The inspectors determined that the issue was indicative of present licensee performance because analyses were performed in 2013. The finding was associated with the cross-cutting aspect of design margins, in the area of human performance, because the organization did not operate and maintain equipment within design margins. Specifically, the licensee operated the SGBD system in excess of that assumed in non-LOCA event analyses because procedural limits corresponding to the assumed flow rates were not established. [H.6]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into procedures and instructions. Contrary to the above, since March 27, 2013, the licensee failed to assure that applicable design bases were translated into procedures or instructions. Specifically, when the analyses for LONF, FLB, and SBO established the assumed values for SGBD flow on June 9, 2010, March 27, 2013, and November 5, 2013, procedural limits corresponding to design basis assumptions for SGBD system flow rates were not translated into system operating procedures for the SGBD system operation. This could have impacted the ability to remove core decay heat using the secondary side heat removal systems during certain events. The licensee's immediate corrective actions included performing a functionality assessment of the SGBD systems for both units, which included re-performing the event analyses, issuing an operations department night order to temporarily provide operators appropriate direction for limiting the SGBD system flow, and planning to update the analyses of record, plant procedures, and the UFSAR with new system limitations. This violation is being treated as an NCV consistent with section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as ARs 2030177, 2031217, and 2031218. (NCV 05000335/2015007-02 and 05000389/2015007-02, Failure to Establish Appropriate

Procedural Limitations to Prevent Exceeding Non-LOCA Event Analysis Assumptions for Steam Generator Blowdown Flow Rate)

3. (Opened) Unresolved Item (URI): Adequacy of 10 CFR 50.59 Screening Performed for Unit 1 SGBD Maximum Flow Evaluation Test

Introduction: An unresolved item (URI) was identified regarding the adequacy of a 10 CFR 50.59 screening that was completed for the performance of a test on the Unit 1 SGBD system. A violation of 10 CFR 50.59(d)(1) was identified for the licensee's failure to perform a full written 10 CFR 50.59 evaluation which provided the basis that the test or experiment did not require a license amendment. Specifically, the test introduced operating conditions that were inconsistent with the analyses described in the station's UFSAR, and a full 10 CFR 50.59 evaluation was not performed. The URI is being opened to provide for additional inspection of the licensee's past operability evaluation of the test conditions, and corresponding event re-analyses, to determine if the violation of 10 CFR 50.59 was more than minor.

Description: On November 11, 2011, the licensee performed a test using procedure 1-LOI-23.01, Steam Generator Blowdown Maximum Flow Evaluation Test, Rev. 1. During the test, SGBD flow was increased to 160 gpm on each steam generator. Prior to the performance of the test, a 10 CFR 50.59 screening was performed for the activity, which determined that the proposed activity did not involve a test or experiment not described in the UFSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the UFSAR. The inspectors determined that at the time the 10 CFR 50.59 screen was completed, Chapter 15 of the UFSAR identified that the assumed SGBD flow rate during the loss of normal feedwater event was 40 gpm per steam generator. Another event involving a loss of feedwater with no AFW flow, described in UFSAR Chapter 10, identified that the SGBD flow rate was assumed to be 35 gpm. The inspectors determined that the SGBD flow rate of 160 gpm allowed by 1-LOI-23.01 was inconsistent with the UFSAR analyses assumptions for the SGBD system. Following the inspectors identification of the discrepancy, the licensee planned to evaluate the test conditions to determine if analysis acceptance criteria could be met when the SGBD flow rate input was increased to values allowed during the test. Additional inspection of this re-analysis is needed to determine if the full 10 CFR 50.59 evaluation, had it been performed, would have concluded that a license amendment should have been pursued prior to implementing the activity. This issue will be identified as URI 05000335/2015007-03, Adequacy of 10 CFR 50.59 Screening Performed for Unit 1 SGBD Maximum Flow Evaluation Test.

4OA6 Meetings, Including Exit

On March 6, 2015, the team presented inspection results to Mr. R. Coffey and other members of the licensee's staff. Additional inspection results were discussed with Mr. E. Katzman of the licensee's staff on April 2, 2015. The team verified that no proprietary information was retained by the inspectors or documented in this report.

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee
L. Berry, Principal Nuclear Engineer
S. Cornell, Nuclear Staff Engineer
E. Katzman, Nuclear Licensing Manager
W. LaFramboise, Design Engineering Manager

NRC
T. Morrissey, Senior Resident Inspector, Saint Lucie Plant
R. Reyes, Resident Inspector, Saint Lucie Plant

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened and Closed

05000335/2015007-01	NCV	Failure to Submit a License Amendment Request for Unit 1 RPS [Section 1R17]
05000335 & 389/2015007-02	NCV	Failure to Establish Appropriate Procedural Limitations to Prevent Exceeding Non-LOCA Event Analysis Assumptions for Steam Generator Blowdown Flow Rate [Section 1R17]

Opened

05000335/2015007-03	URI	Adequacy of 10 CFR 50.59 Screening Performed for Unit 1 SGBD Maximum Flow Evaluation Test [Section 1R17]

Closed

None

Discussed

None

Updated

None

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations
10 CFR 50.59 Evaluation Form, EC 246487, Rev.1
10 CFR 50.59 Evaluation Form, EC 246548, Rev. 2
10 CFR 50.59 Evaluation Form, EC 246594, Rev. 3
10 CFR 50.59 Evaluation Form, EC 250014, Revs. 0 and 1
10 CFR 50.59 Evaluation Form, EC 250134, Rev. 0
10 CFR 50.59 Evaluation Form, EC 277297, Rev. 0
10 CFR 50.59 Evaluation Form, EC 277805, Rev. 0

10 CFR 50.59 Screenings
Screening Review Form, 1-LOI-27.01, Rev. 0
Screening Review Form, EC 249990, Rev. 0
Screening Review Form, EC 271176, Rev. 0
Screening Review Form, EC 274718, Rev. 0
Screening Review Form, EC 275025, Rev. 0
Screening Review Form, EC 277248, Rev. 0
Screening Review Form, EC 283213, Rev. 0
Screening Review Form, IEE EVAL 395011, Rev. 0
Screening Review Form, New Steam Generator Blowdown Temporary Procedure 1-LOI-23.01 Screening, dated 9/26/11
Screening Review Form, PCM 06068, Rev. 0
Screening Review Form, PCM 06123, Rev. 0
Screening Review Form, PCM 06142, Rev. 0
Screening Review Form, PCM 06159, Rev. 0
Screening Review Form, PCM 09071, Rev. 1

Licensing Bases Documents
Technical Specifications and Bases, Current
Updated Final Safety Analysis, Current and Amendment 24
Saint Lucie, Unit 1- Issuance of Amendment Regarding Reactor Coolant System Minimum Flow and Technical Specification Update, dated 8/18/99
Saint Lucie, Unit 1- Issuance of Amendment Regarding Updated Core Operating limits Report (COLR) Methodologies, dated 5/6/04

Calculations
129154-C-P-0088, Hydraulic Transient Analysis of the Intake Cooling Water System, Rev. 1
25486-000-30R-M60G-00001, Final Engineering Study of Unit 1 EDG Diesel Oil Storage Tanks, Rev. 0
25486-266-PHC-FDT-00002, Pipe and Pipe Support Analysis of Overflow Piping for Unit 2 Diesel Oil Storage Tanks for Oil Level Increase per EC 271287, Rev. 4
25486-266-CYC-0001-00001, Evaluation of the Diesel Oil Storage Tank Building Spill Containment, Rev. 2
25846-266-CYC-0001-00001, Evaluation of the Unit 2 Diesel Oil Storage Tank Building Spill Containment, Rev. A
25846-266-CYC-0001-00002, Seismic Assessment of Unit 2 Diesel Oil Storage Tanks for Oil Level Increase per EC 271287, Rev. 4

CN-MRCDA-09-65, PSL1 MSIP SI Fatigue Eval, Rev. 0
CN-OA-08-39, St. Lucie Unit 2 EPU Containment LOCA Pressure and Temperature Analysis, Rev. 2
F-MECH-CALC-018, Evaluation of the Calculations Made by Florida Power and Light for the Minimum and Maximum Torque Requirements of the Butterfly Motor Operated Valves in the Generic letter 89-10 Program at St. Lucie Unit 1, Rev. 12
JPN-PSL-SEIP-93-041, Emergency Diesel Generator Instrument Setpoint Evaluation, Rev. 5
JPN-PSL-SEMP-96-062, Evaluation of Potential Vortex Formation in the RWT, Rev. 0
PSL-1FJE-90-002, GL 89-10 MOV Cable Voltage Drop, St. Lucie Unit 1, Rev. 12
PSL-1FJF-06-154, St. Lucie 1 Cycle 21 PCM—Excore Detector Decalibration Factors, Rev. 0
PSL-1FJM-92-020, Generic Letter 89-10 Differential Pressure Calculation for St. Lucie Unit No. 1 HPSI and AFW Motor Operated Valves, Rev. 5
PSL-1FJM-96-009, Impact of Vortex Formation in RWT, Rev. 2
PSL-1FSE-05-002 CCN-14, Unit 1 125VDC System ETAP Model and Analysis, Rev. 1
PSL-1FSE-09-001 CCN-8, Unit 1 Electrical Coordination Study and Appendix R Evaluation, Rev. 0
PSL-2-FJE-90-0020, St. Lucie Unit 2 Emergency Diesel Generator 2A and 2B Electrical Loads, Rev. 10
PSL-2-FSE-02-002 CCN-1, Unit 2 MCC Control Circuit Voltage Drop Calculation, Rev. 0
PSL-2-FSE-03-010, Unit 2 Electrical System Computer Model (ETAP) Documentation, Rev. 3
PSL-2-FSE-08-001 CCN-9, Unit 2 Electrical Coordination Study, Rev. 0
PSL-2-FSM-10-009, EDG Fuel Oil Supply and Storage – Unit 2, Rev. 1
PSL-ENG-SEMS-07-010, Use of Ultra Low Sulfur Diesel (ULSD) Fuel Oil in the Emergency Diesel Generators, Rev. 2

Corrective Action Documents
AR 00428830
AR 00430053
AR 00435966
AR 00473280
AR 00485688
AR 00582916
AR 01616284
AR 01638429
AR 01644830
AR 01672533
AR 01688436
AR 01724025
AR 01724104
AR 01786230
AR 01989565
AR 02023290

Procedures
0010502, Hazardous Material Emergency Response, Rev. 35
1400065, Maintenance and Calibration of Plant Instrumentation and Control Equipment, Rev. 54
1-ADM-09.23, Time Critical Action Program, Rev. 3
1-AOP-09.03, Secondary Chemistry, Rev. 7

1-AOP-69.01, Inadvertent ESFAS Actuation, Rev. 7
1-EOP-99, Appendices/Figures/Tables/Data Sheets, Appendix N, Hydrogen Purge System Operation, Rev. 54
1-LOI-23.01, Steam Generator Blowdown Maximum Flow Evaluation Test, Rev. 1
1-LOI-27.01, Wood Flour Addition in Circulating Water Pump Suction Wells, Revs. 0 and 1
1-NOP-25.02, Hydrogen Purge System, Revs. 4, 5, 11, and 18
1-PMI-64.04C, Channel C Nuclear Instrument Beginning of Cycle Calibration, Rev. 3
1-PMI-64.04E, Detector 9 and Detector 10, Beginning of Cycle Calibration, Rev. 2
1-PMI-64.07, Excore Neutron Detectors Removal and Installation, Rev. 7
1-SMI-64.05C, Nuclear Instrumentation Channel Calibration Channel C, Rev. 2
1-SMI-64.06, Linear Power Range Safety and Control Channel Monthly Calibration, Rev. 9
1-SMI-64.10C, NI-007 Linear Power Range Safety Channel C Quarterly Calibration, Rev. 6
1-SMI-64.11, Linear Power Range Control Channel Calibration, Rev. 4
1-PMM-59.03, Emergency Diesel Generator 24 Month Preventive Maintenance, Rev. 2
2-AOP-09.03, Secondary Chemistry, Rev. 6
2-NOP-59.05, Emergency Diesel Generator Fuel Oil Receipt and Filtration, Rev. 12
2-OSP-59.01A, 2A Emergency Diesel Generator Monthly Surveillance, Rev. 28
2-OSP-59.12, 18 Month Surveillance for EDG Fuel Oil System Cross-Connect Capability, Rev. 4
2-PTP-57, TCW Heat Exchanger Performance Evaluation, Rev. 1
C-200, Offsite Dose Calculation Model (ODCM), Rev. 43
EN-AA-203-1201, 10 CFR Applicability and 10 CFR 50.59 Screening Reviews, Rev. 0
ENG-QI 2.1, 10CFR50.59 Applicability/Screening/Evaluation, Revs. 13 and 14
RM-AA-100, Records Management Program, Rev. 5

Completed Procedures:
1-NOP-23.02, Steam Generator Blowdown System Operation, Rev. 16 and 17, dated 10/21/11 and 11/30/11
1-NOP-25.03, Hydrogen Purge System (St. Lucie Unit 1), Rev. 18, dated 11/24/14
1-OSP-07.02A, 1A Containment Spray Pump Code Run, Rev. 12, dated 12/03/14
1-OSP-07.02A, 1A Containment Spray Pump Safeguards Full Flow Test, Rev. 13, dated 10/05/13
1-OSP-07.02A, 1B Containment Spray Pump Code Run, Rev. 12, dated 12/16/14
1-OSP-07.02B, 1B Containment Spray Pump Safeguards Full Flow Test, Rev. 13, dated 10/05/13
1-OSP-68.02, Local Leak Rate Test St. Lucie Unit 1, Rev. 25, dated 10/19/2013

Design Basis Documents
DBD-AFW-1, Auxiliary Feedwater System, Rev. 4
DBD-CNTMT-1, Containment System, Rev. 4
DBD-CS-1, Containment Spray System, Rev. 5
DBD-EDG-1, Emergency Diesel Generator System Design Basis Document, Rev. 5
DBD-RPS-1, Reactor Protection System, Rev. 4

Drawings
041550, Sht. 1, RCP Rotating Element, Rev. B
041655, Sht. 1, Reactor Coolant Pump, Rev. A
2998-11458, Atmospheric Dump Valve I-MV-08-18A, 18B, 19A, & 19B, Rev. 11

2998-18810, Solenoid operated Relief Valve V1474 & V1475, Rev. 3
2998-9233, Sht. 1A, Diesel Oil Storage Tank Field Notes and Fittings List, Rev. 4
2998-B-327, U2 Control Wiring Diagram Charging Pump 2A Bypass Valve, Sht. 196, Rev. 12
2998-G-079, Sht. 1, Flow Diagram Main Steam System, Rev. 44
2998-G-086, Sht. 1, Unit 2 Flow Diagram Miscellaneous Systems, Rev. 54
2998-G-089, Sht. 1A, Unit 2 Flow Diagram, Turbine Cooling Water System, Rev. 25
2998-G-089, Sht. 1B, Unit 2 Flow Diagram, Turbine Cooling Water System, Rev. 27
2998-G-089, Sht. 2, Unit 2 Flow Diagram, Turbine Cooling Water System, Rev. 17
2998-G-125, Sht. DO-AC-1, Unit 2 Diesel Oil Piping, Rev. 13
2998-G-169, Sht. 1, Unit 2 Diesel Oil Piping, Rev. 10
2998-G-683, Diesel Oil Stg Tank Foundations & Missile Protection Str – M, Rev. 9
3509-G-115, Sht. 1A, Flow Diagram Steam Generator Blowdown Process System, Rev. 26
8770-11744, 3" Gate Valves, I-V-25-11, -13, & -15, Rev. 2
8770-12801, PSL-1 EDG Fan Drive Mod, 12 and 16 Cylinder Assembly Drawing, Rev. 1
8770-5321, 150lb., Carbon Steel, Locking Device, Gate Valve, Rev. 2
8770-B-326, Sht. 299, Reactor Sump Valve MV-07-2A, Rev. 5
8770-B-327, Sht. 59, Excore Neutron Monitoring System, Rev. 4
8770-B-327, Sht. 485, Post Accident Hydrogen Purge Fan, HVE-7A, Rev. 12
8770-B-327, Sht. 486, Post Accident Hydrogen Purge Fan, HVE-7B, Rev. 9
8770-B-327, Sht. 488, Hydrogen Purge System Valve Outside Air Cooling Line FCV-25-10, Rev. 8
8770-G-080, Sht. 1, Flow Diagram Condensate System, Revs. 73 and 74
8770-G-082, Sht. 2, Flow Diagram Miscellaneous Sampling Systems, Rev 7
8770-G-086, Sht. 1, Flow Diagram Miscellaneous Systems, Rev. 53
8770-G-088, Sht. 2, Flow Diagram Containment Spray and Refueling Water Systems, Rev. 59
8770-G-862, HVAC – Air Flow Diagram, Rev. 38
8770-G-863, Sht. 2, HVAC Refrigerant Piping, Rev. 1
8770-G-870, Sht. 3, HVAC – Reactor Auxiliary Building, Rev. 24
8770-G-878, Sht. 1, HVAC – Control Diagrams, Rev. 38
8770-B-327, U1 Control Wiring Diagram Out-of-Core Neutron Detectors No. 1, 5, & 9, Sht. 60, Rev. 16
8770-B-327, U1 Control Wiring Diagram Out-of-Core Neutron Detectors No. 3 & 7, Sht. 62, Rev. 14
8770-B-328, U1 Cable and Conduit List Installation Notes, Sht. 18, Rev. 8
8770-G-078, Sht. 111C, Unit 1 Flow Diagram, Reactor Coolant Pump 1B1, Rev. 19
8770-G-086, Sht. 1, Unit 1 Flow Diagram Miscellaneous Systems, Rev. 53
8770-G-125, Sht. DO-AC-1, Unit 1 Large Bore Piping Isometric, Diesel Oil, Rev. 6
8770-G-169, Sht. 1, Unit 1 Diesel Oil Piping, Rev. 13
EC 271287-C-002, Sht. 1, Unit 2 Diesel Oil Mark No. DOH-2, Rev. 4
EC 271287-C-005, Sht. 1, Unit 2 Diesel Oil Mark No. DOH-1, Rev. 1
EC 271287-C-008, Sht. 1, Unit 2 Diesel Oil Storage Tank Berm Mesh Doors (D02, D03), Rev. 4
EC 271287-C-009, Sht. 1, Unit 2 Diesel Oil Storage Tank Berm Tornado Doors (D04, D05), Rev. 5
EC 271287-C-014, Sht. 1, Unit 2 Diesel Oil Storage Tank Berm Details/Panels Mesh Doors (D02, D03), Rev. 4
EC 274718-C-100, Sht. 1, 2, 3, St. Lucie Unit 1 Emergency Cooling Water System Barrier Wall, Rev. 0
EC275025-E-004, U2 Schematic Diagram Charging Pump 2B, Rev. 1
EC275025-E-005, U2 Schematic Diagram Charging Pump 2C, Rev. 1
EC275025-E-008, U2 Schematic Diagram Charging Pump 2A, Rev. 1
ENG-10036-1016, U1 Control Wiring Reactor Aux Building Supply Fan HVS-4A, Rev. 3

ENG-10036-1068, Unit 1 Control Wiring Heater Drain Pump 1B, Rev. 1
ENG-10036-1084, Unit 1 Control Wiring 480V MCC 1A-5 Non-Essential Loads Breaker, Rev. 0
ENG-10036-1086, Unit 1 Control Wiring 480V MCC 1A-5 Non-Essential Loads Breaker, Rev. 0
ENG-10036-1092, Unit 1 Control Wiring Diagram 480V SWGR 1A1 MCC Feeders, Rev. 0
ENG-10036-1168, Unit 1 Control Wiring Diagram SIAS ‘A’ and ‘B’ Isolation Relays, Rev. 4

Miscellaneous Documents

0010434, Plant Fire Protection Guidelines, Rev. 45
10 CFR Applicability Determination Form, 1/2-ADM-09.23- Time Critical Operator Action Program, Rev. 0
2998-14080, I/M-NY422711 Atmospheric Dump Valves 1-MV-08-18A, 18B, 19A, 19B, Rev. 9
2998-14874, Power Operated Relief Valves (Solenoid Operated), Rev. 13
2998-15651, Excore Neutron Flux Monitoring System, Rev. 12
ANF-89-151(P)(A), ANF-Relap Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, dated April 1992
AR 01696992 Engineering Disposition St. Lucie Evaluation for the Use of Wood Flour in the Condensers, dated 10/16/11
AR-PCR 016970441-LOI-27.01 Wood Flour Addition in Circulating Water Pump
ASME NQA-1, 1994 Edition, Quality Assurance Program Requirements for Nuclear Facility Applications
Chevron Texaco Letter on Compatibility/Product Performance and Formulas for Texaco-Branded Products in the United States, dated 11/25/2001
CR 2010-3121, 2A EDG 2A1 Fan Blades Engineering Interim Disposition #1, dated 2/11/10
ENG-SPSL-05-0204, Alternate NIS Excore Detector Arrangement, dated 7/21/05
FLO 8770-292C, Electrical Coaxial Cable Specs, dated 7/9/74
FLO 8770-292J, Electrical Coaxial Cable Specs, dated 2/12/75
GIR 10-009, 2A EDG Cooling Fan Blade Surface Corrosion Extent of Condition Examination, dated 2/10/10
GIR 12-003, Visual Examination and UT Inspection of 1B1 EDG Cooling Fan Blades, dated 1/12/12
FPL-05-16, Responses to Detector Replacement Detector Physics Evaluation Requirements, dated 01/14/2015
Instruction Manual No. 259, NIS Vendor Manual
JPN-PSL-SEMP-91-030, NRC GL 89-10 Program Description, Rev. 6
L-2002-078, Update COLR Methodologies in Technical Specification 6.9.1.11, dated 5/22/02
L-2002-242, FPL RAI Response for Core Operating Limits Report Methodologies, dated 12/5/02
L-2004-036, Proposed License Amendment- Operating Limits Report Methodologies-Supplement, dated 211/04
L-2014-201, Submittal of Quality Assurance Topical Report (QATR FPL-1) Revision 13, dated 07/1/13
List of all non-MA cables in MA raceway, dated 3/6/15
LTR-PO-05-3, Responses to St. Lucie Unit 2 Excore Detector Replacement Detector Physics Evaluation Requirements, dated 01/14/2005
MSP 06159, Linear Power Range Excore Detector #8 Replacement, Rev. 0
NAMS Data Sheets for EDG 2B FO Pump Motor, dated 3/2/15
NextEra Energy Quality Assurance Topical Report (FPL-1), Rev. 14
PC/M 04009, Excore Detector Replacement, Rev. 1
PC/M 06162, St. Lucie Unite 1 Cycle 21 Reload, Rev. 0
PCM 157-294, GE NEC Setting Guidance, Rev. 0
Procurement Engineering Evaluation 358053, dated 12/8/95

PSL-00181226, Valve, Solenoid, With Upper Seat- ASME SB637 UNS N07718 INC, dated 7/12/12
PSL-ENG-SEFJ-15-001, Alternate NIS Excore Detector Arrangement (CC-1 Instead of LR-C), Rev. 0
PSL-ENG-SEMS-12-002, Containment Spray Flow Rate Modification Design Inputs, Rev. 1
PSL-ENG-SEMS-97-018, Periodic Verification of Design Basis Capability of Safety Related Motor Operated Valves for NRC GL-96-05, Rev. 2
PSLP-90-2373, Procurement Technical Evaluation for Neutron Detectors Nos. 1 through 10, dated 12/18/91
SIA Report 1001256.401, EDG Radiator Fan Structural Analysis and Fitness for Service, Rev. 1
SL-BF-80-264, St. Lucie Plant Unit #1 Backfit Program Air Conditioning for RAB 43' Level BFI 86-02/PCM 132-80, dated 10/10/80
Specification Number 13172-PE-726, Project Specification for Solenoid Operated Relief Valves for Florida Power and Light Company St. Lucie Unit 2, Rev. 1
TIMI030, QC Inspection Summary Data, Dedication Results, Certificates of Analysis and Destructive Testing, dated 4/30/12
Westinghouse Instruction Manual, Neutron Detectors (Fission & Uncompensated Ion Chambers), Manufactured for Florida Power and Light Company, Contract 19367

<u>Work Orders</u>
33012493-01, Makeup to Cond Stor Tk N2 Seal, dated 08/24/2011
34009891-02, SFTY LIN PWR Range Det #8 for Uncompensated Ion Chamber, dated 05/22/07
34009891-06, SFTY LIN PWR Range Det #8 for Uncompensated Ion Chamber, dated 05/31/07
38013739-12, RCP 2B1; Perform Seal-Flow Test, dated 03/01/11
38022877-02, Motor Operate Valve for CNTMT Sump Feed to LPSI/HPSI B & C PPs, dated 3/3/12
40006384-01, DG ENG 1B1 Clean, UT Inspect and Coat Fan, dated 2/7/12
40006384-03, DG ENG 1B1 PMT Fan, dated 2/5/12
40049782-02, Weld in New Pipe/DRWG: EC250013-M-220, dated 01/27/12
40049782-08, EPU WP4225 Remove/Install LB Piping – CNTMT SPR PP 1A, dated 01/31/12
40049783-10, EPU WP4322 DOST 1A Install New Overflow Line, dated 12/31/11
40049783-11, EPU PMT EC250014: Cal Level Switches/ISLT DOST LIS-17-9A, dated 01/07/12
40065765-08, EPU WP4268 Remove/Install Piping – CS Pump 1B, dated 01/31/12
40065982-09, EPU WP4326 DOST 1B Install New Overflow Line, dated 02/09/2012
40065982-10, EPU PMT EC250014: Cal Level Switches/ISLT DOST LIS-17-9B, dated 02/07/12
40069668-02, 480V MCC 2B7-C01: Breaker Tripped during 2B EDG Run, dated 2/15/11
40100876-26, Replace Relays in the 2A EDG Cntl Panel due to End of Life, dated 2/13/14
40119369-01, 1A2 INL WTR Box; Add Wood Flour to 1A2 Circ Intake Well, dated 11/10/11
40124499-13, PMT, Leak Check TCW Heat Exchanger 2A and Affected Components, dated 11/15/12
40124499-27, PMT, FIT-13-50A, dated 10/28/12
40125310-08, PMT, Leak Check TCW Heat Exchanger 2B and Affected Components, dated 11/13/12
40125310-21, PMT, FIT-13-50B, dated 10/28/12
40125310-22, Perform 2-PTP-57 Performance Evaluations for A and B Trains, Unit 2 TCW 2A/2B Heat Exchangers Replacement, dated 1/09/13
40137484-01, Corrosion and Hole in 1B1 DG Engine Fan Blades
40298234-01, U2 HVE-8A Main Purge Fan, dated 3/13/14
40298234-03, U2 HVE-8A Main Purge Fan Post Maintenance Test, dated 3/13/14
40366832-03, U1: RPS Cab C, NI SFTY CHNL, Install Temp Mod EC 283213, dated 2/4/15

Corrective Action Program Documents generated as a result of the inspection
AR 2025140, Minor UFSAR Discrepancies in Chapter 9.5A for PSL 1 & 2
AR 2026398, EC 235953 UFSAR Update Discrepancy
AR 2026460, Page 15.6-48 Missing From NAMS Version of U2 UFSAR
AR 2029394, Components Contains Oil
AR 2029401, 1B DOST Valve Leaking
AR 2029405, 2B DOST Room Floor Drain is Clogged
AR 2029406, Unit 2 DOST: Open Conduits
AR 2029407, Unit 1 DOST Overflow Line Flange Leaking
AR 2029652, NRC 50.59/Mod Inspection Potential Finding Re: EC 282213
AR 2030173, 1-LOI-23.01 Inadequate 10CFR50.59 Screen
AR 2030177, SG Blowdown Limits for Accident Analyses
AR 2030290, Completed Copy 1-LOI-23.01 Not Found for 11/22/11 Test
AR 2030820, EN-AA-203-1201-F01 Does Not Match NEI 96-07
AR 2030822, Elevated Temperature for MOV Operation
AR 2031217, Actions Required for Functionality Assessment- SGBD U1
AR 2031218, Actions Required for Functionality Assessment- SGBD U2