UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE N.E., SUITE 1200
ATLANTA, GEORGIA 30303-1200

March 30, 2020

Mr. Don Moul
Vice President, Nuclear Division and Chief
Nuclear Officer
Florida Power & Light Company
Mail Stop: NT3/JW
15430 Endeavor Drive
Jupiter, FL 33478

SUBJECT: TURKEY POINT UNITS 3 & 4 – DESIGN BASIS ASSURANCE INSPECTION (TEAMS) INSPECTION REPORT 05000250/2020010 AND 05000251/2020010

Dear Mr. Moul:

On February 14, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Turkey Point Units 3 & 4 and discussed the results of this inspection with Brian Stamp and other members of your staff. The results of this inspection are documented in the enclosed report.

Five findings of very low safety significance (Green) are documented in this report. Five of these findings involved violations of NRC requirements; one was determined to be Severity Level IV. We are treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or the significance or severity of the violations documented in this inspection report, 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; and the NRC Resident Inspector at Turkey Point Units 3 & 4.

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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; and the NRC Resident Inspector at Turkey Point Units 3 & 4.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

James B. Baptist, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 05000250 and 05000251
License Nos. DPR-31 and DPR-41

Enclosure:
As stated

cc: Distribution via LISTSERV®

SUBJECT: TURKEY POINT UNITS 3 & 4 – DESIGN BASIS ASSURANCE INSPECTION (TEAMS) INSPECTION REPORT 05000250/2020010 AND 05000251/2020010 Dated March 30, 2020

ADAMS ACCESSION NUMBER: ML20090G865

☐ SUNSI Review		☐ Non-Sensitive ☐ Sensitive		☐ Publicly Available ☐ Non-Publicly Available		
OFFICE	RII/DRS	RII/DRS	RII/DRS	RII/DRS	RII/DRS	RII/DRS
NAME	T. Fanelli	P. Braxton	C. Franklin	J. Lizardi - Barreto	M. Schweig	James Baptist
DATE	03 30 /2020	03/ 30 /2020	3/30 /2020	03/ 30 /2020	03/ 30 /2020	03/ 30 /2020
OFFICE	Contractors	Contractors				
NAME	M. Yeminy	S. Kobylarz				
DATE	03/30 /2020	03/30 /2020				

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000250 and 05000251
License Numbers:	DPR-31 and DPR-41
Report Numbers:	05000250/2020010 and 05000251/2020010
Enterprise Identifier:	I-2020-010-0021
Licensee:	Florida Power & Light Company
Facility:	Turkey Point Units 3 & 4
Location:	Homestead Florida
Inspection Dates:	January 27, 2020 to February 14, 2020
Inspectors:	P. Braxton, Reactor Inspector T. Fanelli, Senior Reactor Inspector C. Franklin, Reactor Inspector J. Lizardi-Barreto, Construction Inspector M. Schwieg, Reactor Inspector S. Kobylarz, Contractor M. Yeminy, Contractor
Approved By:	James B. Baptist, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a design basis assurance inspection (teams) inspection at Turkey Point Units 3 & 4, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Incorrect Ampacity for Offsite Power Circuitry			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000251,05000250/2020010-01 Open/Closed	None (NPP)	71111.21M
The team identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (10 CFR) 50, Appendix B, Criterion III, "Design Control" for the licensee's failure to translate electric cable ampacity design basis limits into specifications, procedures, and instructions. Specifically, the licensee incorporated unanalyzed higher ampacity limits into plant operating procedures, which could cause the plant's second source of offsite power to fail under load, which was a performance deficiency.			

Failure to Load Test Offsite Power Source			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000251,05000250/2020010-02 Open/Closed	None (NPP)	71111.21M
The team identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (10 CFR) 50, Criterion XI, "Test Control" for the licensee's failure to periodically perform all testing required for the cross-tie cable to the opposite units startup transformer (SUT) and the second source of offsite A.C. power as a whole, under conditions as close to design as practical for the full operation sequence that brings the offsite A.C. source into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.			

Three Examples of Inadequate Design Control for Safety Related Structural Concrete			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000251,05000250/2020010-03 Open/Closed	[H.12] - Avoid Complacency	71111.21M
The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to implement adequate design control measures during repair activities on safety-related structural concrete.			

Two Examples of Failure to Evaluate Design Changes that Adversely Degraded Original Plant Design

Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000251,05000250/2020010-04 Open/Closed	None (NPP)	71111.21M
The team identified two examples of an Severity level IV Green NCV of Title 10 CFR 50.59.(d)(1), "Changes, Tests and Experiments," and of Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to include a written evaluation which provides the bases for the determination that plant changes did not require a license amendment pursuant to paragraph (c)(2) of 10 CFR 50.59 by ensuring that the quality of the original plant design was neither degraded nor adversely affected by subsequent plant changes or modifications in accordance with the site Quality Assurance (QA) Program document FPL-NQA-100, Revision 2, dated 1973.			

Harsh Environments from High-Energy Line Breaks			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000251,05000250/2019011-01 Closed	None	71111.21N
The NRC identified a Green Non-Cited Violation (NCV) of 10 CFR 50.49.(d), “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” for the licensee's failure to provide the analyses of high energy line breaks (HELBs) including cracks in piping in the vicinity of onsite power equipment necessary for safe shutdown of the nuclear plant. Specifically, the licensee failed to provide the required analyses of the environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the locations in the turbine building where the equipment must perform.			

Additional Tracking Items

None.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.21M - Design Bases Assurance Inspection (Teams)

The inspectors evaluated the following components and listed applicable attributes, permanent modifications, and operating experience:

Design Review - Risk-Significant/Low Design Margin Components (IP Section 02.02) (5 Samples)

(1) RHR and HHSI Suction and Supply Headers
•Material condition and configuration (e.g., visual inspection during a walkdown)
•Consistency between station documentation (e.g. procedures) and vendor specifications
•Corrective maintenance records, and corrective action history
•Compliance with UFSAR, TS, and TS Bases
•Calculations: (pump head, capacity, NPSH, Vortexing)
•Normal and emergency operating procedures
•Completed surveillance tests to ensure acceptance criteria have been met

(2) 125VDC Distribution Panel 3D23
• Material condition and configuration (e.g., visual inspection during a walkdown)
• Operating environment
• Consistency between station documentation (e.g. procedures) and vendor specifications
• Maintenance effectiveness
• Corrective maintenance records, and corrective action history
• Breaker short circuit capacity
• Panel loading
• Load voltage adequacy
• Overcurrent protection and coordination

(3) 4160V 3A Switchgear Cross-tie to Unit 4A Startup Transformer
• Material condition and configuration (e.g., visual inspection during a walkdown)
• Operating environment
• Consistency between station documentation (e.g. procedures) and design analyses
• Maintenance effectiveness
• Corrective maintenance records, and corrective action history
• Cross-tie procedure adequacy
• Cross-tie cable load current

• Adequacy of voltage during cross-tie
• Cross-tie breaker 3AA22 overcurrent setting and calibration testing
• Cross-tie maintenance, surveillance, and load testing

(4) Unit 3 & Unit 4 EDG Sequencers
•Surveillance testing and recent test results
•Compliance with UFSAR, TS, and TS Bases
•Material condition and configuration (i.e. visual inspection during walkdown)
•Adequacy of corrective action activities

(5) Unit 3 & Unit 4 Emergency Diesel Generator Room Ventilation
•Visual non-intrusive inspection (walk down) to assess the installation configuration, material condition, and potential vulnerability to hazards
•Normal and emergency operating procedures
•Protection against external external events (seismic and tornado)
•Maintenance effectiveness (e.g., MR, procedures)
•Vendor specification
•Set-points and instrument uncertainty
•Room heat up/ventilation
•Flow rate tests
•System Health (Failures, CRs, OP Evals)
•Modifications

Design Review - Large Early Release Frequency (LERFs) (IP Section 02.02) (1 Sample)

(1) Unit 4 CCW Heat Exchangers/Pumps/Head Tanks, and TPCW isolation valve for CCW POV-4882, POV-4883
•Heat exchanger design (number of tubes, number of passes)
•Shell flow rate and tubes flow rate
•Availability of cooling water
•HX testing/cleaning
•Pump flow rates and pressure/head capacity curve/NPSH
•Vortex formation
•Head Tank design (elevation and capacity)/pressure rating/fill source and capability/interaction with the surge tank
•Relief valve location and design
•Valve size and capacity/operating conditions
•Operator capability to actuate the valve

Modification Review - Permanent Mods (IP Section 02.03) (2 Samples)

(1) MSP-290147, Correction to Locked Rotor Accident Analysis
(2) EC 291973, Unit 4 Fuel Handling Building Concrete Repairs
EC 280927, EDP For Repair Of U3 Main Steam Platform Concrete Wall Associated With Pipe Support 3-MSH-3A

Review of Operating Experience Issues (IP Section 02.06) (1 Sample)

(1) IN-17-06, Battery and Battery Charger Short Circuit Current Contributions to a Fault on the Direct Current Distribution System

71111.21N - Design Bases Assurance Inspection (Programs)

The inspectors evaluated [list program reviewed] program implementation through the sampling of the following components:

INSPECTION RESULTS

Incorrect Ampacity for Offsite Power Circuitry			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000251,05000250/2020010-01 Open/Closed	None (NPP)	71111.21M

The team identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (10 CFR) 50, Appendix B, Criterion III, “Design Control” for the licensee’s failure to translate electric cable ampacity design basis limits into specifications, procedures, and instructions. Specifically, the licensee incorporated unanalyzed higher ampacity limits into plant operating procedures, which could cause the plant’s second source of offsite power to fail under load, which was a performance deficiency.

<u>Description</u>: Each nuclear Unit, 3 & 4, uses the opposite unit’s startup transformer (SUT) as the required second source of Alternating Current (A.C.) offsite power per technical specification 3.8.1. For Unit 3, this is identified as an emergency cross-tie between the 3A 4160 Volt (V) switchgear, breaker 3AA05, the Unit 4 SUT (4X03) bushing, and then to Unit 4 breaker 4AA22. For Unit 4, this is identified as an emergency cross-tie between the 4A 4160V switchgear, breaker 4AA05, the Unit 3 SUT (3X03) bushing, and then to Unit 3 breaker 3AA22. The cross-tie cabling is sized to original plant design at 1250 MCM (1 MCM = 1,000 circular mills). The team observed that the Unit 3 cross-tie cable from the transformer included installation in outdoor covered cable trays with exposure to full sun, exposed conduits, and underground raceways for the routing to the 3A switchgear. The updated final safety analysis report (UFSAR) Table 8.2-1 limited 1250 MCM cables to 485 amperes. The UFSAR considered installed cable configurations to determine the ampacities. However, the team found that on 11/25/09, the plant operating procedures incorporated a maximum limit of 600 amperes for the cross-tie.

The licensee confirmed the 485 ampere UFSAR Table 8.2-1 ampacity limit in a calculation performed in 1967, but could not find the basis for the 600 ampere limit that was allowed in plant procedures since 11/25/09. The licensee stated the load on the cross-tie would normally be maintained below the 485 ampere limit as a basis for evaluating the operability for the identified condition. Loading the cable to over 485 amperes would adversely affect the reliability and availability of the offsite circuit because it could result in the failure of the cable and the loss of the second emergency source of offsite power.

Corrective Actions: The licensee entered the condition into their corrective action program.

Corrective Action References: Action Request 02343114

<u>Performance Assessment</u>:

Performance Deficiency: The failure to ensure that cable ampacities were controlled in accordance with the UFSAR Table 8.2-1 was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the 600 ampere limit in plant procedures exceeded the UFSAR design basis ampacity of the cross-tie cable which adversely affected the availability and reliability of the second source of offsite power. Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined the finding was of very low safety significance (Green) because the finding was a design or qualification deficiency of a mitigating SSC and the SSC maintained its functionality. Cross-Cutting Aspect: Not Present Performance. No cross cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.
Enforcement: Violation: Title 10 CFR 50, Appendix B, Criterion III, "Design Control," states, in part, that, "Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, since 11/25/09 the site did not translate the UFSAR ampacity design basis for the cross-tie cable into procedures and instructions. Specifically, the 600 ampere limit in plant procedures adversely affected the availability and reliability of the second source of offsite power, because the limit exceeded the ampacity of the cross-tie cable and could cause the cable to fail under load. Enforcement Action: This violation is being treated as an non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Load Test Offsite Power Source			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000251,05000250/2020010-02 Open/Closed	None (NPP)	71111.21M
The team identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (10 CFR) 50, Criterion XI, "Test Control" for the licensee's failure to periodically perform all testing required for the cross-tie cable to the opposite units startup transformer (SUT) and the second source of offsite A.C. power as a whole, under conditions as close to design as practical for the full operation sequence that brings the offsite A.C. source into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.			

Description: Description: Each nuclear Unit, 3 & 4, uses the opposite unit's SUT, via a 1250 MCM cross-tie cable, as it's required second source of A.C. offsite power per technical specification (TS) 3/4.8.1, "A.C. SOURCES. The limiting conditions for operation established for TS 3.8.1.1, required, in part, "as a minimum, the following A.C. electrical power sources…:
a. Two startup transformers and their associated circuits"

The surveillance requirement (SR) for the cross-tie circuit established that the testing consist of transferring the unit's power supply from the auxiliary transformer to the startup transformer. The team determined that the SR had never been performed. This was because the requisite circuit design to accomplish the testing was never installed. The UFSAR analysis section that applies to these cross-tie circuits in Section 8.2.2.1.2.1, "General Design Criteria (GDC) as Defined In 10 CFR 50 Appendix A," specified, in part, that "GDC 18 - Inspection and Testing of Electric Power Systems, the design of the electric power distribution system at Turkey Point does permit appropriate periodic inspection and testing of important areas and features. The testing and inspection of the electric power distribution system is governed by the surveillance requirements of Section 3/4.8 of the Turkey Point Technical Specifications." This UFSAR specification established that the surveillance for the cross-tie would periodically test: (1) the operability and functional performance of the cross-tie cable and (2) the operability of the second source of A.C. offsite power as a whole and, under conditions as close to design as practical, the full operation sequence that brings the A.C. source into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system. The team identified that the cross-ties have never been load tested during the life of the plant and the cross-tie cables insulation have never been monitored for degradation during the life of the plant.

The team found the failure to periodically test the cross-tie circuit under load does not conform with the licensing basis as described in the UFSAR and TS.

Corrective Actions: The licensee entered the condition into their corrective action program and performed an operability evaluation.

Corrective Action References: Action Request 02344617

Performance Assessment:

Performance Deficiency: The failure to periodically load test the offsite power cross-tie between units in accordance with the UFSAR Chapter 8, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to load test the unit emergency offsite power cross-tie failed to ensure the reliability and capability of the offsite power circuits.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined the finding was of very low safety significance (Green) because the finding was a design or qualification deficiency of a mitigating SSC and the SSC maintained its functionality.

Cross-Cutting Aspect: Not Present Performance. No cross cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.
Enforcement: Violation: 10 CFR 50, Appendix B, Criterion XI, “Test Control,” states, in part, “A test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, operational tests during nuclear power plant operation.” Contrary to the above, since 1972 the site failed to establish a test program to assure that all testing required to demonstrate that the Unit emergency offsite power cross-tie will perform satisfactorily in service was performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, the licensee failed to assure the performance of all testing required to demonstrate (1) the operability and functional performance of the cross-tie cable and (2) the operability of the second source of A.C. offsite power as a whole and, under conditions as close to design as practical, the full operation sequence that brings the A.C. source into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system. Enforcement Action: This violation is being treated as an non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Three Examples of Inadequate Design Control for Safety Related Structural Concrete			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000251,05000250/2020010-03 Open/Closed	[H.12] - Avoid Complacency	71111.21M
The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee’s failure to implement adequate design control measures during repair activities on safety-related structural concrete.			
Description: The team observed work and testing activities and reviewed design documents associated with the structural concrete repairs and cathodic protection system installation for Turkey Point Unit 4 Fuel Handling Building. The inspectors identified three examples of licensee failure to implement adequate design control measures during repair activities on safety-related structural concrete: • Design Change Package EC 291973, "Unit 4 Fuel Handling Building Concrete Repairs," Revision 3, was issued for concrete repair work on the Fuel Handling Building. EC 291973 determined that the horizontal reinforcement steel bars within the concrete walls of the building were not structural members. However, the building code for structural concrete American Concrete Institute (ACI) 318, Building Code Requirements For Structural Concrete And Commentary, Section 14.3.3, required a minimum ratio of horizontal reinforcement (rebar) area to gross concrete area. In			

addition, EC 291973 Drawing CP-51, U4 FHB West Wall Cathodic Protection Cathodic Protection Details, Revision 0, illustrated the drilling of the horizontal reinforcement bars without safety related procedures or instructions controlling the work activity, for the installation of cable connections of the cathodic protection system. The holes drilled into the steel reinforcing bars reduced the cross-sectional area and strength of these horizontal reinforcement bars.

- Specification CN-2.11, "Specification for Concrete Testing, Placing, Curing and Finishing," Revision 7, required concrete cylinder tests per American Society for Testing and Materials (ASTM) C39, "Standard Test Method for Compressive Strength of Cylindrical Concrete Specimen," in accordance with ACI Code 318. However, Field Change Request (FCR) 007, Unit 4 Fuel Handling Building Concrete and Cathodic Protection System, Revision 0, for EC 291973 changed the testing to cube tests in order to align testing with the mortar vendor’s instructions per ASTM C109, "Standard Test Method for Compressive Strength of Hydraulic Cement Mortars." The site used a test method, intended for mortar mixtures, for the testing of concrete mixtures which was not an acceptable method for testing concrete strength and was not be in accordance with ACI Code 318 requirements.

- ACI Code 318 and 349 required the evaluation of flexural and shear loading on both vertical and horizontal direction for the selection of slab thickness and for reinforcement required to control deformation and assure adequate shear and flexural strengths. Calculation 200024-01 did not check flexural and shear loading for determining the design controlling condition, and therefore it did not determine if additional reinforcement was needed beyond the code minimum required reinforcement. Section 8.2 of Calculation 200024-01 did not check shear loading to ensure that flexural loading controls on a one-way upper wall, and Section 8.1 of the same calculation did not check flexural loading for ductility ratio on horizontal reinforcement bar. In addition, the calculations evaluated did not adequately evaluate design loading, including crane loads.

Corrective Actions: The licensee entered the issue into their corrective action program.

Corrective Action References: Action Requests 2344653 and 2344656

Performance Assessment:

Performance Deficiency: The failure to identify structural reinforcement, perform credible concrete strength testing, and perform credible structural loading evaluations in accordance with ACI 318/349, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Design Control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to ensure that structural concrete was designed and installed to safety standards commensurate with the safety function failed to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The team determined the finding was of very low safety significance (Green) because it only represented a degradation of the radiological barrier function for the spent fuel pool building. Cross-Cutting Aspect: H.12 - Avoid Complacency: Individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Individuals implement appropriate error reduction tools. Specifically, the licensee failed to consider and incorporate design requirements and acceptance limits contained in applicable design documents in order to perform reinforce concrete repairs in accordance with manufacturer's testing instructions, design calculations and drawings.
Enforcement: Violation: Title 10 CFR 50, Appendix B, Criterion III, "Design Control," required in part, that applicable regulatory requirements and the design basis, for structures, systems, and components (SSCs), are correctly translated into specifications, drawings, procedures, and instructions; and that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to the above, since April 19, 2019, the site failed to ensure that applicable regulatory requirements and the design basis, for safety related structural concrete were correctly translated into specifications, drawings, procedures, and instructions; and that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Enforcement Action: This violation is being treated as an non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Two Examples of Failure to Evaluate Design Changes that Adversely Degraded Original Plant Design			
Cornerstone	Significance/Severity	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green Severity Level IV NCV 05000251,05000250/2020010-04 Open/Closed	None (NPP)	71111.21M
The team identified two examples of an Severity level IV Green NCV of Title 10 CFR 50.59.(d)(1), "Changes, Tests and Experiments," and of Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to include a written evaluation which provides the bases for the determination that plant changes did not require a license amendment pursuant to paragraph (c)(2) of 10 CFR 50.59 by ensuring that the quality of the original plant design was neither degraded nor adversely affected by subsequent plant changes or modifications in accordance with the site Quality Assurance (QA) Program document FPL-NQA-100, Revision 2, dated 1973.			
Description: The site Quality Assurance (QA) Program document FPL-NQA-100, Revision 2, dated 1973, stated, in part, that "the quality of the original plant design is neither degraded			

nor adversely affected by subsequent plant changes or modifications." Procedure EN-AA-203-1102, "Safety Classification Determination," Revision 7 states "SSCs whose purpose is to initiate automatic safety features that are required for accident prevention and mitigation or to shut down the reactor and maintain it in a safe condition, are Safety Related." The team identified two examples where modification did not meet the above criteria.

Example 1: The team reviewed diesel room temperature calculations, PTN-3FJE-91-016, "Heat Loss Calculation for EDG 3A/3B Rooms," Revision 1, JPN-PTN-SEEP-91-007, "Temperature Rating of Electrical Equipment in Emergency Diesel Generator Rooms 3A and 3B," Revision 1, and NAI-1483-001, "Generator Room Heat Up Analysis," Revision 1. These calculations determined the room exhaust fans were required to maintain the diesel engine room below its maximum normal allowable temperature and thus the EDG safety function. Calculation NAI-1483-001 also determined that without the exhaust fan operating and an outside temperature of 50 degrees Fahrenheit (ºF), the maximum EDG room temperature will reach equilibrium at approximately 115.9ºF, which is above the maximum normal allowable room temperature. The licensee's analyses determined that the exhaust fans are required when operating the EDGs when outside temperature is above 45ºF. The Unit 3 EDGs cannot reliably perform their safety function if the EDG room exhaust fans fail to run. This was a change to the plant design. Prior to this, the design of the plant specified that "the safety related ventilation will be provided by the diesel engine radiator fans when the diesel engine is operating. It was determined that the forced ventilation due to the diesel engine cooling fans will maintain the diesel engine room below its maximum normal allowable temperature." Once this change from original plant design was identified, these fans were not treated as safety related in accordance with the site classification criteria. The design change was also not evaluated to ensure the quality of the original plant design was neither degraded nor adversely affected by this change. The team noted two design issues with the exhaust fans. First, the EDG room exhaust fans did not meet design specification 5610-M-36, "Exhaust Fans for Ventilation," Revision 3, which required backdraft dampers, which were either not installed or were removed at some point in time. This allowed the fans to freewheel in reverse from breezes blowing through the rooms. Second, the exhaust fans were not seismically qualified in accordance with quality standards commensurate with their safety function. The reliance on non-Appendix B equipment and acceptance of design flaws in the exhaust fan design result in more than a minimal increase in the likelihood of occurrence of a malfunction of the safety related electrical equipment in the diesel rooms that were previously evaluated in the final safety analysis report.

Example 2: In 1983, in Plant Change and Modification (PCM) 83-141 package the licensee modified the safety related load center and switchgear rooms (LCSWGR). The PCMs purpose was to upgrade fire barriers around the site including the LCSWGR. These modifications sealed both trains of LCSWGR preventing natural air flow. Prior to this, each train of the LCSWGR were vented to the outside, and were partially open to one another allowing air to flow between them. The PCM stated, in part, "fan 3V15 must be removed from the 4160V Switchgear room to allow the fan opening to be closed for the purpose of installing a fire door and barrier between Switchgear rooms 3A and 3B." These prior features would allow natural air circulation to flow through the rooms. In 1992 the licensee performed calculation JPN-PTN-SENJ-92-003, "Safety Assessment for Load Center and Switchgear Rooms HVAC Safety Classification." This was in response to an internal technical audit that the prior PCMs contained no basis for the statement that the HVAC facilities do not perform a safety function. The calculation stated, in part, that early post operating license modifications separated the rooms [3A & 3 B], closed the exterior wall openings and installed direct expansion air conditioning units in the rooms with condenser units located outdoors. The

calculation verified that the HVAC does not perform a safety function because the LCSWGR doors could be opened as a last resort. However, it did not identify the potential high energy line breaks (HELBs) concerns that would then affect the LCSWGR. Non-safety related and non-seismically qualified high energy fluid equipment and piping surround the LCSWGR, and currently the LCSWGR doors are not HELB barriers. The quality of the original plant design was degraded and adversely affected by these changes. The reliance on non-Appendix B equipment in the new LCSWGR configurations and the failure to recognize that the LCSWGR were exposed to possible HELBs result in more than a minimal increase in the likelihood of occurrence of a malfunction of the safety related electrical equipment in the LCSWGR that were previously evaluated in the final safety analysis report. Corrective Actions: The licensee performed a prompt operability determination to ensure the operability of the commercial components that perform safety related functions and determined that the components were operable but non-conforming. Corrective Action References: Action Requests 2343688, 2344552, and 2344655
<u>Performance Assessment</u>: Performance Deficiency: The failure to ensure that the quality of the original plant design was neither degraded nor adversely affected by changes was a performance deficiency. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the use of commercial components in safety related applications, aggravated by the design deficiencies, and inadequate seismic design, failed to ensure the required availability, reliability and capability for safety systems. Significance: The inspectors assessed the significance of the finding using Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The team determined the finding was of very low safety significance (Green) because the finding was a design or qualification deficiency of a mitigating SSC and the SSC maintained its functionality. Cross-Cutting Aspect: Not Present Performance. No cross cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.
<u>Enforcement</u>: The ROP’s significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to address this violation which impedes the NRC’s ability to regulate using traditional enforcement to adequately deter non-compliance. Severity: This violation was determined to be a severity level IV violation for the failure to include a written evaluation which provides the bases for the determination that plant changes did not require a license amendment pursuant to paragraph (c)(2) of 10 CFR 50.59 by ensuring that the quality of the original plant design is neither degraded nor adversely affected by subsequent plant changes or modifications and it was evaluated as having very low safety significance (i.e., green) by the SDP.

Violation: Title 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that “activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings.” Contrary to the above, since 1983 the site failed to accomplish activities affecting quality in accordance with instructions, procedures, or drawings. Specifically, the licensee failed to assure that the quality of the original plant design was neither degraded nor adversely affected by subsequent plant changes or modifications in accordance with the site Quality Assurance (QA) Program document FPL-NQA-100. Enforcement Action: This violation is being treated as an non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Harsh Environments from High-Energy Line Breaks			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000251,05000250/2019011-01 Closed	None	71111.21N
The NRC identified a Green Non-Cited Violation (NCV) of 10 CFR 50.49.(d), “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” for the licensee's failure to provide the analyses of high energy line breaks (HELBs) including cracks in piping in the vicinity of onsite power equipment necessary for safe shutdown of the nuclear plant. Specifically, the licensee failed to provide the required analyses of the environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the locations in the turbine building where the equipment must perform.			
Description: The inspectors reviewed unresolved item 05000250, 05000251/2019011-01 and consulted additional NRC offices and determined that a violation existed. Per the inspection procedure 71111.21N, “Design Bases Assurance Inspection (Programs),” for environmental qualification, the inspectors verified “that there are no potential high energy break locations (verify using review of licensing basis) located in areas determined to be a mild environment.” The Turkey Point UFSAR, Section 5.4.1, “Design Basis,” for “Pipe Whipping Restraints,” described the current licensing basis (CLB) for the postulation of HELBs outside containment. Section 5.4.1 stated in part, “an analysis was performed to analyze high energy lines outside the containment for pipe failures… these requirements were initially established post Operating License as a result of a request by the Atomic Emergency Commission (AEC) in 1972. This request was clarified later to provide changes and corrections to the guide entitled ‘General Information Required for Consideration of the Effect of a Piping System Break Outside Containment,’ (References 7 and 8).” References 7 and 8 are the (the Giambusso Letter and errata thereto). The Giambusso letter provided the criteria, used to determine the design basis piping break locations in ASME Section III, Class 1, 2, and 3 piping systems. These classes are specifically applied to seismically qualified safety related pipes. The breaks included the double-ended pipe rupture. The errata specified, in part, that “where pipes carrying high energy fluid are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to			

those essential structures and systems."

The turbine building had high energy line configurations adjacent to unprotected onsite power equipment that are required for safe shutdown. The high energy lines in these areas were neither safety related nor seismically qualified per the updated final analysis report (UFSAR) Appendix 5A, titled "Seismic Classification & Design Basis." Some of these configurations included:
In unit three,

- a motor control center (MCC) with diesel auxiliaries adjacent to main-steam lines, high energy pumps, and multiple feedwater lines.
- a diesel main power feeder was noted adjacent to main-steam lines.
- In both units 3 & 4,
- high energy fluids sources were observed adjacent to onsite power distribution switchgear rooms without designated HELB barriers.

In addition, time critical operator actions prop-open the doors to the switchgear and load center rooms would expose the equipment to the various sources of high energy fluids mentioned above. The inspectors determined that HELBs could credibly subject onsite power equipment to harsh environments for which they were not qualified. Further, the inspectors noted that Information notice (IN) 2000-20, titled "Potential Loss of Redundant Safety Related Equipment Because of the Lack of High-Energy Line Break Barriers," described such conditions, as above, as potentially risk significant.

The inspectors noted license amendments increased the core thermal power by 16.8%. The increased power level would increase the affects evaluated in any previously completed environmental effects analyses. This power uprate was an opportunity to ensure that the previous documented break analyses of the high energy piping mentioned above was up to date. Including the evaluation of postulated open cracks in pipes carrying high energy fluid where they are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant as mentioned in the "Giambusso Letter" errata, and to which the effects of the recent changes to the mass and energy release caused by the power increases would have affected.

The inspectors asked for the documented evidence of activities affecting quality related to the HELB analyses, such as detailed licensee inspections of the piping systems, reviews of the configurations, and calculations of HELB effects supporting their environmental conclusions. The licensee was unable to supply the documentation. It was not evident to the inspectors how the licensee implemented their power uprates without their HELB analyses on hand to verify if the 1973 HELB conclusions changed. Therefore, the effects of the impact of the power uprate may be unanalyzed in some areas of the plant.

Corrective Actions: The licensee entered this issue into their corrective action program.

Corrective Action References: Action Request 2324737

Performance Assessment:

Performance Deficiency: The failure to provide the required analyses of the environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the locations in the turbine building where the safe shutdown equipment must perform in accordance with the 10 CFR 50.49 was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, HELBs in the turbine building could credibly create harsh environments surrounding the safety related power trains and challenge critical safety functions.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined the finding was of very low safety significance (Green) because the finding did not cause an actual reactor trip AND the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition.

Cross-Cutting Aspect: Not Present Performance. No cross cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.

Enforcement:

Violation: Title 10 CFR Part 50.49.(d) required, in part, that the licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the information in paragraphs (d)(1), (2), and (3) of this section for this electric equipment important to safety in a qualification file. The applicant or licensee shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safely meet the requirements of paragraph (j) of this section.

- The performance specifications under conditions existing during and following design basis accidents
- The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.
- The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

Contrary to the above, the licensee failed to prepare a list of electric equipment important to safety in the turbine building covered by this section. In addition, the licensee failed to include the information in paragraphs (d)(1), (2), and (3) of this section for this electric equipment important to safety in a qualification file. The licensee failed to keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment is important to safely meet the requirements of paragraph (j) of this section.

- The performance specifications under conditions existing during and following design basis accidents
- The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.
- The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

Enforcement Action: This violation is being treated as an non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On February 14, 2020, the inspectors presented the design basis assurance inspection (teams) inspection results to Brian Stamp and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.21M	Calculations	18712-473-E-01	DC Voltage Drop Calculation for Safe Shutdown Components	Rev. 1
		200024-01	Unit 4 Fuel Handling Building Wall Elevation	Rev. 1
		200024-02	Units 3&4 Fuel Handling Building Wall Repair Design	Rev. 1
		5177-EF-11	Cable Ampacity in Duct Bank, Maintained Space Tray, Conduit & Free Air	Rev. 2
		87-261.6008	Emergency Diesel Generator Building, Diesel Generator Room Ventilation	Rev. 4
		CN-FPL-UPRATE-096	THD Evaluation of the Impact of Thermal Conductivity Degration on Loss of Flow and Locked Rotor Events	Rev. 0
		CN-SEE-I-11-15	Turkey Point RHR Cooldown With One CCW Heat Exchanger Out of Service	Rev. 0
		CN-SEE-III-08-32	Calculation of Turkey Point Unit 3 & 4 ECCS Injection Flows for teh Extended Power Uprates	Rev. 0
		CN-SEE-III-09-4	Turkey Point EPU RHRS Cooldown	Rev. 0
		EC-096	Cable Ampacity and Voltage Drop Calculation	Rev. 1
		FPL023-CALC-01	Turkey Point Cask Handling Facility Cooling Load Combination	Rev. 1
		JPM-TPN-SEEP-91-007	Temperature Rating of Electrical Equipment in Emergency Diesel Generator Rooms 3A and 3B	Rev. 1
		NAI-1396-008	Control Room Isolation by Intake Radiation Monitors RAD-6642/6643 for the Turkey Point EPU AST Analysis	Rev. 4
		NAI-1396-015	Turkey Point EPU Locked Rotor Radiological Analysis with Alternative Source Term	Rev. 4
		NAI-1483-001	Turkey Point Unit 3 Emergency Diesel Generator Room Heat-Up Analysis	Rev. 1
		PTN-3FJE-91-016	Heat Loss Calculation for EDG 3A and 3B Rooms	Rev. 1
		PTN-3FJE-92-024	Start-Up Transformer No. 4 Phase Overcurrent	Rev. 1

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		PTN-3FJN-91-048	EDG-3A and 3B Rom Ventilation Requirements and Temperature Rise	Rev. 1
		PTN-3LJM-07-022	Unit 3 NPSH During ECCS Recirculation	Rev. 2
		PTN-BFJM-96-004	CCW Heat Exchanger Design Basis Case and Operability Curves	Rev. 4
		PTN-BFSM-02-006	AOV Program ICW to TPCW Isolation Valve Actuator Capability	Rev. 0
		PTN-BFSM-11-020	MOV Program: NRC Generic Letter 89-10 MOV Design Basis Differential Pressure Determination - Post EPU	Rev. 0
		PTN-BFSM-11-021	NRC Generic Letter 89-10 MOV Thrust Calculation - Post EPU	Rev. 2
		PTN-BFSM-11-022	NRC Generic Letter 89-10 MOV Actuator Ecaluation - Post EPU	Rev. 5
		PTN-BFSM-14-007	Vortex Design Evaluation of Refueling Water Storage Tank	Rev. 0
		PTN-BFSM-97-04	Miscellaneous CCW Head Tank Elevation Assessment	Rev. 0
	Corrective Action Documents	2061032, 2074681, 2145289, 2183242, 2202574, 2301977, 2313653, 2272412, 2265798, 2170901, 2131691, 2022159,		
	Corrective Action Documents Resulting from Inspection	2343114, 2344617, 2343688, 2344444, 2344327,		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		2342537, 2342727, 2344112, 2344552, 2344655, 2343095, 2344112		
	Drawings	5177-265-EG-22	Circuit Breaker Fuse/Coordination Study	Rev. 9
		561 Q-M-3075	Auxiliary Feedwater System	Rev. 29
		5610-E-1	Main Single Line Unit 3, Sheet 1	Rev. 47
		5610-E-1	Main Single Line Unit 4, Sht. 2	Rev. 19
		5610-T-E-1591	Operational Diagram Electrical Distribution, Sht. 1	Rev. 82
		5612-E-1605	Battery 3A & 3B Load Profiles	Rev. 19
		5613-E-11	Electrical 125V DC & 120V Instrument AC, Sheet 1	Rev. 20
		5613-E-12	Electrical 125V DC & 120V Instrument AC	Rev. 12
		5613-E-25	Reactor Auxiliaries Boron Sefety Injection Valve LP 'A' Cold Leg MOV-3-843A, Sheet 28P	Rev. 11
		5613-E-25	Reactor Auxiliaries Residual Heat Removal Inlet Isolation Valve MOV-3-751, Sheet 42A	Rev. 9
		5613-E-25	Reactor Auxiliaries Residual Heat Removal Inlet Isolation Valve MOV-3-750, Sheet 37A	Rev. 9
		5613-E-25	Reactor Auxiliaries Loop A Hot Leg SI Stop Valve MOV-3-869, Sheet 27k	Rev. 8
		5613-E-25	Reactor Auxiliaries Residal Heat Removal Heat Exchanger Outlet Valve MOV-3-863A, Sheet 31A	Rev. 8
		5613-E-25	Reactor Auxiliaries Refueling Water Storage Isolation Valve MOV-3-864A, Sheet 27H	Rev. 7
		5613-E-27	Mechanical Auxiliaries Diesel Generator 3A Vent Fan 3V34A	Rev. 1
		5613-E-3	4KV Switchgear 3A & 3B, Sheet 1	Rev. 8
		5613-E-6	Emergency Diesel Generator 3A Load List	Rev. 21
		5613-M-16-69	Start and Control Circuit Diesel Generator 3B	Rev. 11
		5613-M-3022	Emergency Diesel Engine and oil System DG 3B Air Starting	Rev. 18

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			System	
		5613-M-3030	Component Cooling Water System	Rev. 27
		5613-M-3050	Residual Heat Removal System	Rev. 40
		5613-M-3062	Safety Injection System	Rev. 45
		5613-M-3070	Turbine Building Ventilation Load Center and Switchgear Rooms Chilled Water System Train B	Rev. 4
		5613-t-L1	Logic Diagram Sequencer, Sheet 12	Rev. 3
		5613-T-L1	EDG Engine Start	Rev. 5
		5613-T-L1, Sheet 12A	Emergency Bus Load Sequencer Loading Logic Diagram	Rev. 2
		8815-008-002	Seismic Qualification of Emergency Diesel Generator Building Rooms A & B Vent Fans	Rev. 0
		PTN-M-96-093-001	CCW System Pressurization Tank Arrangement	Rev. 0
	Engineering Changes	EC 280927	EDP For Repair Of U3 Main Steam Platform Concrete Wall Associated With Pipe Support 3-MSH-3A	Rev. 7
		EC 291973	Unit 4 Fuel Handling Building Concrete Repairs	Rev. 3
		EC 291973	Unit 4 Fuel Handling Building Concrete Repairs	Rev. 2
		FCR-001	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-002	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-003	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-004	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-005	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-006	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-007	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-008	Unit 4 Fuel Handling Building Concrete and Cathodic	Rev. 1

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			Protection System	
		FCR-009	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 1
		FCR-010	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 0
		FCR-011	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 1
		FCR-012	Unit 4 Fuel Handling Building Concrete and Cathodic Protection System	Rev. 1
		MSP-290147	Correction to Locked Rotor Accidnet analysis.	Rev. 0
	Engineering Evaluations	Functional Assessment AR 2303370	Unit 4 Fuel Handling Building Exterior Concrete Walls – Degraded	Rev. 3
	Miscellaneous	5610-030-DB-002	Component Cooling Water System Design Basis Document	04/19/2018
		5610-030-DB-002	Component Design Requirements Document Component Cooling Water System	04/19/2018
		5610-050-DB-001	Design Basis Document: Residual Heat Removal System	Rev. 14
		5610-050-DB-002	Component Design Requirements Document: Residual Heat Removal System	Rev. 15
		5610-062-DB-001	Design Basis Document: Safety Injection System	Rev. 15
		5610-062-DB-002	Component Design Requirements Document: Safety Injection System	Rev. 17
		5610-E-11	General Cable Corporation 5000V Power Service	Rev. 7
		5610-M-36	Specification for Exhaust Fans for Ventilation	Rev. 2
		5613-M-313	Instrument Setpoint List	Rev. 54
		AA1539	Limitorque Type SMB Instruction and Maintenance Manual	Rev. 3
		Concrete Test 11461	Levels 0 and 1 West Wall	Rev. 0
		Concrete Test 11467	Patch Pours Levels 4, 5, 7	Rev. 0
		Concrete Test 195-0085	East Wall North Levels 1 & 2	Rev. 0
		Concrete Test 195-0094	West Wall – ICCP	Rev. 0

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		Concrete Test 195-0095	South Wall ICCP	Rev. 0
		Concrete Test 195-0096	West Wall Repairs	Rev. 0
		Concrete Test 195-0101	West Wall – ICCP	Rev. 0
		Concrete Test 195-0104	South East Wall Repairs	Rev. 0
		Concrete Test 205-0002	West Wall – ICCP	Rev. 0
		Structural Deficiency Report U4SFB-EXT-001	Spent Fuel Building Exterior (Reinforced Concrete)	8/31/2012
		tca 13-21 SGTR w Loop.xlsm	Time Critical Actions 13-21 during a SGTR with LOOP event	02/11/2020
		TCA CCW Makeup.xlsx	Time Critical Action for CCW makeup excel	02/11/2020
		Turkey Point Plant Units 3 and 4 Subsequent License Renewal Application		Rev. 1
		V00506B	Instruction Manual for the Emergency Bus Load Sequencer Volume III	Rev. 0
		V00506D	Technical Manual for the Emergency Bus Load Sequencers	Rev. 0
		Z273	Limitorque HBC Series Installation and Maintenance	Rev. 7
	Procedures	0-ADM-561	Structures Monitoring Program	Rev. 9A
		0-ONOP-103.2	Cold/Hot Weather Conditions	Rev. 10
		0-ONOP-103.3	Severe Weather Preparations	Rev. 28A
		0-PME-003.31	Vital 120 VAC and 125 VDC Breaker Maintenance	Rev.10
		3-EOP-ECA-0.0	Loss of All AC Power	Rev.14B
		3-EOP-ECA-0.1	Loss of All AC Power Recovery Without SI Required	Rev. 5
		3-EOP-ECA-0.2	Loss of All AC Power Recovery with SI Required	Rev. 5

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		3-NOP-05	4KV Buses A, B, and D	Rev. 13
		3-NOP-070	Vital Load Center and Switchgear Rooms Chilled Water Air Conditioning System	Rev. 12
		3-ONOP-004	Loss of Offsite Power	Rev. 11A
		3-ONOP-004.1	System Restoration Following Loss of Offsite Power	Rev. 3A
		3-ONOP-004.2	Loss of 3A 4KV Bus	Rev. 4C
		3-OSP-023.1	Diesel Generator Operability Test	Rev. 12
		3-OSP-203.1	Train A Engineered Safeguards Test	Rev. 28
		4-NOP-030	Component Cooling Water System	Rev. 36
		4-NOP-075.02	AFW Backup Nitrogen System Alignment And Bottle Changeout	Rev. 8
		4-ONOP-075	Auxiliary Feedwater System Malfunction	Rev. 12
		4-OPS-062.2D	Safety Injection Pump 4A Comprehensive Pump Test	Rev. 9
		4-OSP-050.2A	Residual Heat Removal Train A Test - Standby Alignment	Rev. 7
		4-OSP-050.2B	Residual Heat Removal Train B Test - Standby Alignment	Rev. 9
		4-OSP-050.2C	Residual Heat Removal Train A Comprehensive Test - Cooldown Alignment	Rev. 17
		4-OSP-050.2D	Residual Heat Removal Tran B Comprehensive Test - Cooldown Alignment	Rev. 15
		4-OSP-050.2E	RHR Check Valve Inservice Testing	Rev. 3
		4-OSP-062.2A	Safety Injection Pump 4A Group B Pump Test	Rev. 9
		4-OSP-062.2B	Safety Injection Pump 4B Group B Pump Test	Rev. 10
		4-OSP-062.2C	Safety Injection System Inservice Valve Testing	Rev. 5
		4-OSP-062.2E	Safety Injection Pump 4B Comprehensive Pump Test	Rev. 9
		4-OSP-062.4	Safety Injection System - Full Flow Test	Rev. 5
		4-OSP-075.5	AFW Operations Surveillance Procedure	Rev. 4
		499983-01	Pull-Off Test Validation & Implementation Plan	Rev. 0
		CN-2.11	Specification for Concrete Testing, Placing, Curing and Finishing	Rev. 7
		CN-2.24	Drilled-In Expansion Anchors in Concrete St. Lucie Units 1 & 2 and Turkey Point Units 3 & 4	Rev. 13

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		CN-2.9	Specification for Concrete Materials and Mixes, Concrete Mixing and Transportation	Rev. 4
		CP-51	U4 FHB West Wall Cathodic Protection Cathodic Protection Details	Rev. 0
		EN-AA-203-1102	Safety Classification Determination	Rev. 7
		FPLCORP020-REPT-107	Aging Management Program Basis Document – Structures Monitoring	Rev. 1
		O-ADM-232	Time Critical Operator Action Program	Rev. 12
		PTN-ENG-LRAM-00-0042	Systems and Structures Monitoring Program – Licensee Renewal Basis Document	Rev. 13
		SPEC-C-042	Specification for Grout	Rev. 0
	Work Orders	40252535-14	U3 MN STM Line Support, Cracked/Spalled Concrete	05/25/17
		40014583, 40014584, 40299561-01, 40526372-01, RWO 07-12, 40217030-01, 40437521-01, 40257313-01, 40630058, 40648530, 40547572-01, 40281822-01, 40542167-01, 40632651-01, 40469953-01, 40469919-01, 40469919-03, 40632645-01, 40649640-01,		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		40217063-01, 40571153-01, 40441768-01, 40650889-01, 40217063-04		
		40252535-15	U3 MN STM Line Support, Cracked/Spalled Concrete	05/03/17