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

December 20, 2022

Bob Coffey
Executive Vice President, Nuclear Division and Chief Nuclear Officer
Florida Power & Light Company
700 Universe Blvd
Mail Stop: EX/JB
Juno Beach, FL 33408

SUBJECT: TURKEY POINT UNITS 3 & 4 – DESIGN BASIS ASSURANCE INSPECTION (PROGRAMS) INSPECTION REPORT 05000250/2022011 AND 05000251/2022011

Dear Bob Coffey:

On November 18, 2022, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Turkey Point Units 3 & 4 and discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

Two findings of very low safety significance (Green) are documented in this report. Two of these findings involved violations of NRC requirements. 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.

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,

Signed by Baptist, James
on 12/20/22

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

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SUBJECT: TURKEY POINT UNITS 3 & 4 – DESIGN BASIS ASSURANCE INSPECTION (PROGRAMS) INSPECTION REPORT 05000250/2022011 AND 05000251/2022011 – dated December 20, 2022

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OFFICE	RII/DRS	RII/DRS	RII/DRS	RII/DRS	RII/DRS
NAME	C. Franklin	G. Ottenberg	R. Fanner	T. Fanelli	J. Baptist
DATE	12/05/2022	12/05/2022	12/19/2022	12/19/2022	12/20/2022

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000250 and 05000251
License Numbers:	DPR-31 and DPR-41
Report Numbers:	05000250/2022011 and 05000251/2022011
Enterprise Identifier:	I-2022-011-0027
Licensee:	Florida Power & Light Company
Facility:	Turkey Point Units 3 & 4
Location:	Homestead, FL.
Inspection Dates:	October 31, 2022 to November 18, 2022
Inspectors:	T. Fanelli, Senior Reactor Inspector C. Franklin, Reactor Inspector G. Ottenberg, Senior Reactor Inspector
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 (programs) 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

Failure to Address Valve Capability with Preloaded Bolts			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000250,05000251/2022011-01 Open/Closed	None (NPP)	71111.21N.02
The NRC identified a finding and associated Green non-cited violation (NCV) of 10 CFR 50.55a(b)(3)(ii), for the licensee's failure to confirm that valve bolted joints could perform in accordance with established acceptance criteria to provide reasonable assurance that safety-related motor operated valves (MOVs) could perform their design-basis safety function.			

Failure to Properly Categorize Valves in Accordance with ASME OM Code			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000250,05000251/2022011-02 Open/Closed	None (NPP)	71111.21N.02
The inspectors identified a Green finding and associated non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(f), for the licensee's failure to properly categorize valves MOV-3(4)-856A(B), High Head Safety Injection (HHSI) Pump Recirculation to Refueling Water Storage Tank (RWST), as Category A valves in accordance with ASME Code for Operation and Maintenance of Nuclear Power Plants (OM) Code, subsection ISTC.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
URI	05000250,05000251/2022011-03	Containment Purge Valve Seat Leakage Integrity After Seismic Events	71111.21N.02	Open

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

<u>71111.21N.02 - Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements</u>

<u>POV Review (IP Section 03) (8 Samples)</u>

The inspectors:

a. Determined whether the sampled POVs are being tested and maintained in accordance with NRC regulations along with the licensee's commitments and/or licensing bases. Specific Guidance
b. Determined whether the sampled POVs are capable of performing their design-basis functions.
c. Determined whether testing of the sampled POVs is adequate to demonstrate the capability of the POVs to perform their safety functions under design-basis conditions.
d. Evaluate maintenance activities including a walkdown of the sampled POVs (if accessible).

(1) MOV-3-535, Reactor Coolant System - Pressurizer Power Operated Relief Valve (PORV) Stop Valve {3" Velan Gate Valve w/ AC Limitorque SMB-00-10 actuator}
(2) MOV-4-856A, Safety Injection (SI) - High Head Safety Injection (SI) Pump Recirculation to Refueling Water Storage Tank {2" Copes Vulcan Globe Valve w/ AC Limitorque SMB-000-5 actuator}
(3) MOV-4-863B, Residual Heat Removal - RHR Alternate Discharge Isolation {6" Aloyco Gate Valve w/ AC Limitorque SMB-00-15 actuator}
(4) MOV-3-843A, Safety Injection - SI Cold Leg Injection Valve {4" Anchor/Darling Gate Valve w/ AC Limitorque SMB-0-15 actuator}
(5) MOV-3-860A, Residual Heat Removal - Containment South Recirculation Sump Isolation Valve {14" Anchor/Darling Gate Valve w/ AC Limitorque SMB-0-40 actuator}
(6) PCV-3-455C, Reactor Coolant System - Pressurizer PORV {3" Copes Vulcan Globe Valve w/ Copes Vulcan 000-160 Diaphragm-Reverse acting actuator}
(7) POV-3-2602, Containment Purge - Containment Purge Exhaust {54" Henry Pratt Butterfly Valve w/ Chicago Fluid Power A3M Pivoting Cylinder actuator}
(8) POV-4-4883, Inlet Cooling Water - ICW/turbine plant cooling water (TPCW) Isolation Valve to Hx 3B {30" Henry Pratt Butterfly Valve w/ Bettis G3014-SR4 actuator}

INSPECTION RESULTS

Failure to Address Valve Capability with Preloaded Bolts			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000250,05000251/2022011-01 Open/Closed	None (NPP)	71111.21N.02
The NRC identified a finding and associated Green non-cited violation (NCV) of 10 CFR 50.55a(b)(3)(ii), for the licensee's failure to confirm that valve bolted joints could perform in accordance with established acceptance criteria to provide reasonable assurance that safety-related motor operated valves (MOVs) could perform their design-basis safety function.			

Description: The NRC staff in the initial Generic Letter (GL) 89-10 (10 CFR 50.54f) requested nuclear power plant licensees to take additional measures to provide assurance that safety-related MOVs will function when subjected to design-basis conditions. The staff listed specific recommended actions for safety-related MOVs under the "Recommended Actions" section of this generic letter. In GL 89-10, Supplement 1, Question 16, the staff stated, in part, that to determine the conditions under which the MOV must perform its safety function, the licensee should consider all relevant factors that may affect the capability of the MOV to perform its function.

During the implementation of inspection procedure (IP) 71111.21N.02, the inspectors sampled the RHR alternate discharge isolation MOV-4-863B for review of its capability to perform its design-basis safety function. The MOV was purchased in 1970. The procedure (0-GME-102.10, "Motor Operated Valve Operator, Inspection and Overhaul SMB-00," Rev 5,) specified 155 foot-pounds (ft-lbs) of preload torque for the actuator and yoke bolts. The inspectors reviewed FPL calculation PTN-BFJM-92-022 Revision 1, titled, "Crane MOV Thrust Calculation," dated April 1994 to determine the thrust conditions under which the MOV must perform its intended function and its effects on the bolted joints. The calculation, in part, compared the expected thrust generated by the MOV actuator to the bolt yield strength (the point at which permanent material deformation will occur), to determine whether the bolts structural limit would be exceeded during expected operating conditions. The specific bolt material was identified as A193-B8 stainless steel. This steel was designated by the American Society of Mechanical Engineers (ASME) as approximately 18% chromium & 8% nickel (18Cr-8Ni) with specific ratios of other elements that contribute to the material characteristics. At 100 °F, the A193-B8 material had a minimum yield strength of 30 kilo-pounds per square inch (ksi) and a minimum tensile strength of 75 ksi, and the minimum yield strength at its design temperature of 400 °F was only 21.4 ksi. The calculation identified that the bolts were the second weakest component during valve operation. The calculation did not include the contribution of the preload torque on these effects when evaluating bolt strength or bolted joint strength.

The inspectors noted that the preload torque specified in procedure 0-GME-102.10 exceeded the minimum yield strength by more than 200% at all temperatures and approached the 75 ksi tensile strength. In addition, during historical overhaul of the valve some bolts were found to be deformed and required replacement as described in condition report (CR) 94-0530. The licensee planned to reuse these bolts to reassemble the valves after maintenance; however, an acceptable torque value for bolt re-use was noted to be less than 65% of the minimum yield strength, which the licensee did not meet. The inspectors noted that the bolts have not been challenged where the bolt material is weaker at 400 °F with valve thrust loads applied. The inspectors determined that the effect of the preload impacts the conclusion in the Crane MOV thrust calculation. The inspectors determined that the overstress, potential

deformations, and fatigue of these bolts over many years adversely affects the reasonable assurance that the MOV can perform its design-basis safety function. In addition, the inspectors found, during walkdowns, that these bolts do not have ANSI/ASME or ASTM, A193-B8 markings as specified by the drawings or calculations, for traceability. Rather, the bolts are marked with a general 300 series (18Cr-8Ni) steel, untraceable to any specific formulation. A literature review determined that the bolts may be considered to have a 30 ksi minimum yield strength and a 70 ksi minimum tensile strength. The licensee intends to confirm the bolt materials as result of the inspection. The inspectors considered this lack of identification markings to further place the capability of these valves to perform their safety function in question. Corrective Actions: The licensee entered this issue into the corrective action program and performed an operability evaluation Corrective Action References: **AR 2441013, and 2441048**
<u>Performance Assessment</u>: Performance Deficiency: The failure to confirm that valve bolted joints could perform in accordance with established acceptance criteria to provide reasonable assurance that safety-related MOVs could perform their design-basis safety function 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 address bolted joint performance in accordance with established acceptance criteria affected the availability and reliability of MOVs that respond to initiating events. Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The inspectors assessed the significance of the finding using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” Using Exhibit 2, Mitigating Systems Screening Questions, Item A.5 was most appropriate because the degraded condition represented a loss of a PRA system and/or function as defined in the Plant Risk Information Book (PRIB). A regional senior risk analyst (SRA) performed a risk analysis. The detailed risk assessment used the guidance in Appendix A and using SAPHIRE 8 version 8.2.6 and the Turkey Point SPAR model Revision 8.80 dated 5/26/2022 to model the condition. The SRA assumed an exposure time of 1 year (max allowed by SDP) and modelled the condition by adjusting the common cause failure (CCF) terms for the failure to run of the motor driven RHR pumps. Since MOV 863A&B are 8“ valves, Internal flooding would have to be considered as well but it is isolable. The SRA conservatively chose a one order of magnitude adjustment to the nominal CCF terms to bound the condition. The dominant accident sequence was a medium break loss of coolant accident (MLOCA) with a failure of Low pressure and High Pressure recirculation. The increase in core damage probability was less the 1E-8. Therefore, the finding is characterized as very low safety significance (GREEN) 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 50.55a(b)(3)(ii), OM condition: Motor-Operated Valve (MOV) testing, states that licensees must comply with the provisions for testing MOVs in ASME OM Code, ISTC 4.2, 1995 Edition with the 1996 and 1997 Addenda, or ISTC–3500, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(iv) of this section, and must establish a program to ensure that MOVs continue to be capable of performing their design basis safety functions. Contrary to the above, since April 20, 1994, the licensee failed to ensure that MOVs continue to be capable of performing their design-basis safety functions. Specifically, the licensee did not address the resulting valve bolted joint effectiveness due to preloading when evaluating the relevant factors that might affect the capability of the sampled MOV to perform its design-basis safety function. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Properly Categorize Valves in Accordance with ASME OM Code			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000250,05000251/2022011-02 Open/Closed	None (NPP)	71111.21N.02
The inspectors identified a Green finding and associated non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(f), for the licensee's failure to properly categorize valves MOV-3(4)-856A(B), High Head Safety Injection (HHSI) Pump Recirculation to Refueling Water Storage Tank (RWST), as Category A valves in accordance with ASME Code for Operation and Maintenance of Nuclear Power Plants (OM) Code, subsection ISTC.			
Description: The MOV-3(4)-856A(B) valves are arranged in series on the HHSI pump minimum flow line return path to the RWST and are boundary valves between the HHSI system and the RWST, which is vented to atmosphere. The valves have a passive function to remain open during the injection mode of operation of the emergency core cooling system (ECCS) following a loss-of-coolant accident (LOCA) and have an active function to close when the ECCS is realigned to take a suction from the containment sump. The valves must be closed and leak tight to prevent the radioactive fluid from the sump from being released to atmosphere via the RWST vent. Further, NRC Information Notice (IN) 91-56, Potential Radioactive Leakage to Tank Vented to Atmosphere, also identified the need to consider the potential for these valves to be an atmospheric release pathway following a LOCA. The Turkey Point Updated Final Safety Analysis Report (UFSAR), section 14.3.5, Environmental Consequences of a Loss-of-Coolant Accident, describes the assumptions used in the LOCA dose calculation of record which utilizes alternative source term (AST) methodologies. Specifically, it states: "The LOCA dose consequence analysis is consistent with the guidance provided in Appendix A of Reference 1, "Assumptions for Evaluating the Radiological Consequences of a LWR [light water reactor] Loss-of-Coolant Accident," as discussed below:" And further states:			

"ECCS Leakage to the RWST. The ECCS backleakage to the RWST is assumed to be 0.1 gph [gallons per hour] based upon doubling of the expected total seat leakage through both sets of motor operated valves which isolate the recirculation flow from the RWST."

The 'expected total seat leakage' of 0.1 gph referred to in the UFSAR is a summation of the leakage from the RWST suction isolation valves MOV-3(4)-864A(B) and the ECCS pump minimum recirculation flow RWST isolation valves MOV-3(4)-856A(B), however, ensuring the valves' seat leakage remains within this assumed amount of leakage depends on the valve setup achieving seat stress values sufficient to make the valve leak tight. The applied seat stress depends on the closing control switch trip setpoint and any inertial and relaxation effects that occur after the control switch trip.

The inspectors observed that the current LOCA dose calculation of record inputs and assumptions were based on AST methodologies, which were incorporated into the Turkey Point licensing bases through a license amendment issued June 23, 2011. The inspectors also reviewed the licensee's follow-up to NRC IN 91-56, which revealed that the licensee's dose calculation in place prior to the implementation of the AST amendment did not explicitly account for the leakage pathway to the RWST. Following the specification of a maximum amount of seat leakage in the current AST dose calculation, the MOV-3(4)-856A(B) valves were not reassigned to be Category A within the station's inservice testing (IST) program.

The current UFSAR identified a specific maximum amount of seat leakage in the closed position for fulfillment of the valves' required dose mitigation functions, however, the IST program did not consider this maximum specified seat leakage when categorizing the valves in accordance with ASME OM Code, ISTC-1300. Specifically, the 2004 Edition of the ASME OM Code, Subarticle ISTC-1300, which is applicable to the licensee's current IST interval, states, "Valves within the scope of this Subsection shall be placed in one or more of the following categories… Category A: valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s), as specified in ISTA-1100." Further, paragraph ISTC-3610, Scope of Seat Leakage Rate Test, required that "Category A valves shall be leakage tested, except that valves which function in the course of plant operation in a manner that demonstrates functionally adequate seat leak-tightness need not be additionally leakage tested." The MOV-3(4)-856A(B) valves are maintained open during normal plant operation and are only exercised closed during cold-shutdown testing and are not demonstrated to have functionally adequate seat-tightness during normal plant operation. Consequently, the leakage testing requirements of ISTC-3610 were also not met.

Corrective Actions: Following the discovery that the MOV-3(4)-856A(B) valves had not been leakage tested, the licensee reviewed the results of recent diagnostic tests and determined that there was a reasonable expectation that the valves would be leak tight based on their ability to achieve the seat stress necessary for the valves' corresponding leakage classification at the current control switch trip settings. One of the valves, MOV-4-856A, required the reduction of some conservatism in the design assumptions to conclude that adequate seat stress would be applied.

Corrective Action References: AR 2442093 – MOV-4-856A OM Code IST Classification

Performance Assessment:

Performance Deficiency: The failure to properly categorize valves MOV-(3)4-856A(B), HHSI Pump Recirculation to RWST as Category A valves in accordance with ASME OM Code, subsection ISTC, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the SSC and Barrier Performance 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 the leak tight integrity of MOV-3(4)-856A(B) MOVs, adversely affected the barrier preventing radio nuclides from entering the atmosphere through the RWST vent.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The finding was determined to be Green because it did not represent an actual open pathway in the physical integrity of reactor containment (valves, airlocks, etc.) and did not involve an actual reduction in function of hydrogen igniters. Specifically, the licensee was able to show that, at the current switch settings, there was a reasonable expectation that the valves would be leak tight if required to perform their function following an accident requiring ECCS containment sump recirculation.

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 50.55a(f)(4), Inservice testing standards requirement for operating plants, required in part that, "Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves that are within the scope of the ASME OM Code must meet the inservice test requirements (except design and access provisions) set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) and (3) of this section and that are incorporated by reference in paragraph (a)(1)(iv) of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components." The ASME OM Code and addenda that was applicable to the fifth (current) 120 month interval was the 2004 Edition through 2006 Addenda (incorporated by reference in paragraph (a)(1)(iv)), required in subarticle ISTC-1300, that, "Valves within the scope of this Subsection shall be placed in one or more of the following categories… Category A: valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s), as specified in ISTA-1100." Further, sub-paragraph ISTC-3610, Scope of Seat Leakage Rate Test, required that "Category A valves shall be leakage tested, except that valves which function in the course of plant operation in a manner that demonstrates functionally adequate seat leak-tightness need not be additionally leakage tested."

Contrary to the above, since 2011, when seat leakage for the MOV-3(4)-856A(B) valves was limited to maximum amount in the closed position, the licensee failed to place the MOVs into Category A in accordance with ISTC-1300 and failed to leak test the MOVs in accordance with ISTC-3610.

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

Unresolved Item (Open)	Containment Purge Valve Seat Leakage Integrity After Seismic Events URI 05000250,05000251/2022011-03	71111.21 N.02

Description: The UFSAR Section 5A-1.2, Design Classification of Structures, Systems and Equipment, states in part, that "Class 1 structures, systems and equipment are those whose failure could cause uncontrolled release of radioactivity in excess of the established guidelines as prescribed in 10 CFR 50.67, those essential for immediate and long-term operation following a loss-of-coolant accident to either cool the core or reduce the containment pressure, those required to function after a loss of power occurrence or steam line break to permit a controlled NSSS cool-down, or those required for a safe shutdown. Associated with Class 1 structures, systems and equipment are their supports, enclosures, piping, wiring, controls, power sources and switchgear. They are designed to withstand the appropriate earthquake loads applied simultaneously with other applicable loads without loss of function.

The UFSAR Section 9.8.1, "Auxiliary Building Ventilation and Containment Purge Systems," subsection 9.8.1.2, "System Design and Operation," stated in part, that the function of the purge valve was to quick-close for bubble tight shut-off.

The licensee could not provide evaluations that demonstrated that the containment purge butterfly valves seat to disc interface could remain functional after a seismic event. The displacements experienced by large butterfly valves could misalign the seat and disc. A small displacement could represent a large leakage rate from containment exceeding the valves allowable leakage rate.

The only evaluation presented was a Henry Pratt stress calculation (Weak Link Analysis) performed in response to NRC Generic Letters (GL) 79-46 and 79-54, both concerned "Containment Purging and Venting During Normal Operation." The GLs addressed the required "tests or analyses to show that containment purge or vent valves would shut without degrading containment integrity during the dynamic loads of a design basis loss of coolant accident (DBA-LOCA)." Further, the GLs specified that "purge and vent valve structural elements (valve/actuator assembly) must be evaluated to have sufficient stress margins to withstand loads imposed while valve closes during a design basis accident. Torsional shear, shear, bending, tension, and compression loads/stresses should be considered. Seismic loading should be addressed." The team noted that this did not specifically address sealing integrity of a closed valve after a seismic event. The Turkey Point General Design Criteria (GDC), criterion 2, Performance Standards (Category A), required in part, that "those systems and components of reactor facilities which are essential to the prevention of accidents which could affect, the public health and safety or to mitigation of their consequences should be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be Imposed by natural phenomena such as earthquakes..." This GDC design requirement predated the GLs 79-46 & 54 requirements and required containment isolation valves to maintain capability to protect the public from radiation.

The Pratt evaluation stated that these NRC guidelines for demonstration of operability of purge and vent valves, required the evaluation will demonstrate: A, the valve closure time during a LOCA will be less than or equal to the no-flow time demonstrated during shop tests; B, this analysis consists of a static analysis of the valve components indicating if the stress levels under combined seismic and LOCA conditions; and that, Sealing integrity can be evaluated against chemical, radiation, and temperature effects. The team noted that seismic considerations were not used for sealing integrity.

Planned Closure Actions: The inspectors need to review additional information from the licensee concerning correspondence with the NRC regarding leak tight qualification of the purge valve.

Licensee Actions: The licensee entered this issue into the corrective action program

Corrective Action References: AR 2442084

EXIT MEETINGS AND DEBRIEFS

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

- On November 18, 2022, the inspectors presented the design basis assurance inspection (programs) inspection results to Bob Coffey and other members of the licensee staff.
- On December 8, 2022, the inspectors presented the Re-exit inspection results to Michael Durbin and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.21N.02	Calculations	Cust PO: 5177-099	Pratt Henry Stress Report for 54 in RIA with Pratt Containment Isolation/Purge Valve Analysis w/Pratt Two Plate Cylinder Operator	Rev 1
71111.21N.02	Calculations	PTN-3FSE-07-001	Unit 3 Safety Related AC Electrical Distribution PSB-1, Short Circuit, Voltage Drop & Bus Loading Analysis- ETAP Program	4
71111.21N.02	Calculations	PTN-4FSE-07-002	Unit 4 Safety Related AC Electrical Distribution PSB-1, Short Circuit, Voltage Drop & Bus Loading Analysis- ETAP Program	3
71111.21N.02	Calculations	PTN-BFJM-92-020	Velan-MOV Thrust Calculation	1
71111.21N.02	Calculations	PTN-BFJM-92-021	ANCHOR/DARLING - MOV THRUST CALCULATION	3
71111.21N.02	Calculations	PTN-BFJM-92-022	Crane MOV Thrust Calculations	Rev 1
71111.21N.02	Calculations	PTN-BFJM-92-023	Copes-Vulcan MOV Thrust Calculation	1
71111.21N.02	Calculations	PTN-BFJM-95-015	EPRI MOV Performance Prediction Program- Anchor Darling DDG Valves	4
71111.21N.02	Calculations	PTN-BFJR-99-011	PTN AOV Risk Ranking	6
71111.21N.02	Calculations	PTN-BFSM-02-001	AOV Program-Power Operated Relief Valve (PORV) Valve/Actuator Capability	2
71111.21N.02	Calculations	PTN-BFSM-02-006	AOV Program - ICW to TPCW Isolation Valve/Actuator Capability	0
71111.21N.02	Calculations	PTN-BFSM-11-020	MOV Program: NRC Generic Letter 89-10 MOV Design Basis Differential Pressure Determination- Post-EPU	1
71111.21N.02	Calculations	PTN-BFSM-11-021	NRC Generic Letter 89-10 MOV Thrust Calculation- Post-EPU	4
71111.21N.02	Calculations	PTN-BFSM-11-022	NRC Generic Letter 89-10 MOV Actuator Evaluation- Post-EPU	6
71111.21N.02	Calculations	PTN-BMHE-11-	Motor Operated Voltage Drop Calculation	3

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		005		
71111.21N.02	Calculations	PTN-BMHE-11-007	Calculation to Determine the Starting Torque, Current, nd Voltage of AC Valve Actuator Motors at Elevated Temperatures	2
71111.21N.02	Calculations	PTN-ENG-SESS-01-048	Engineering Evaluation Scoping and Categorization for Air Operated Valves	0
71111.21N.02	Corrective Action Documents		ARs 02394456, 02004712, 02410840, 02239372, 02371000,	
71111.21N.02	Corrective Action Documents	AR 02410497	PEN 36 (CTMT PURGE- POV-3-2602/2603) Failed as Left LLRT	10/03/2022
71111.21N.02	Drawings	5610-M-1200-54	4" No. S350 WDD-ASA Series 1500 Welding Ends Outside Screw & Yoke Gate Valve	7
71111.21N.02	Drawings	5610-M-1200-65	14" Series 300 Darling MOV-3-860A, *-862A, *-862B	3
71111.21N.02	Drawings	5610-M-430-171, Sheet 5	Safeguards System	28
71111.21N.02	Drawings	5610-M-470-49	2" Class 1500# (SI Recirc)	4
71111.21N.02	Drawings	5610-M-83-1	Component Drawing Cylinder and Spring-Operated RIA Butterfly Valve-54"	Rev 10
71111.21N.02	Drawings	5613-E-25 SH59C	Control Circuitry and Contact Development for SV-3-2602	Rev 1
71111.21N.02	Drawings	5613-E-25, Sheet 27D	Reactor Auxiliaries Containment Sump Isolation Valve MOV-3-860A	7
71111.21N.02	Drawings	5613-E-25, Sheet 28P	Reactor Auxiliaries Boron Safety Injection Valve LP 'A' Cold Leg MOV-3-843A	11
71111.21N.02	Drawings	5613-M-3041, Sheet 2-4	Reactor Coolant System	45
71111.21N.02	Drawings	5613-M-3050, Sheet 1	Residual Heat Removal System	41
71111.21N.02	Drawings	5613-M-3053 SH1	P&ID Containment Purge and Penetration Cooling System	Rev 26
71111.21N.02	Drawings	5613-M-3062, Sheet 1	Safety Injection System	46
71111.21N.02	Drawings	5613-M-3062, Sheet 2	Safety Injection System	25

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.21N.02	Drawings	5613-M-3064, Sheet 1	Safety Injection Accumulator System Inside Containment	28
71111.21N.02	Drawings	5614-E-25 SH31B	Reactor Auxiliaries Residual Heat Removal Heat Exchanger Outlet Valve MOV-4-863B Control Circuitry	Rev 8
71111.21N.02	Drawings	5614-E-25 SH31B1	Reactor Auxiliaries Residual Heat Removal Heat Exchanger Outlet Valve MOV-4-863B Contact Development	Rev 2
71111.21N.02	Drawings	5614-E-25, Sheet 122A	Reactor Auxiliaries SI Mini Recirc. Valve MOV-4-856A	5
71111.21N.02	Drawings	5614-E-25, Sheet 122A1	Reactor Auxiliaries SI Mini Recirc. Valve MOV-4-856A	2
71111.21N.02	Drawings	5614-M-3019, Sheet 1	Intake Cooling Water System	42
71111.21N.02	Drawings	5614-M-3050 SH1	P&ID Residual Heat Removal System	Rev 41
71111.21N.02	Drawings	5614-M-3062, Sheet 1	Safety Injection System	41
71111.21N.02	Engineering Changes	EC 284522	TMD to defeat Annunciator AN-A-4/1 for PCV-3-455C	0
71111.21N.02	Engineering Changes	EC 288188	Evaluate use of Low Friction (Teflon) Packing Set in MOV-3-843A&B, MOV-3-869, and MOV-3-872 (SPEC-M-097)	2
71111.21N.02	Engineering Changes	EC 294531	Evaluate use of OEM Recommended Upgraded Alternate Wedge Pin Material for Anchor/Darling Double Disc Gate MOVs. Update Drawings & VTM as needed.	0
71111.21N.02	Engineering Changes	PC/M 09-055	Revise the Control Logic for MOV-4-381, 856A, 856B, 1400, 1401, 1402, and 6386 to bypass the close torque switch until the valve disc reaches the seat	0
71111.21N.02	Engineering Changes	PC/M 86-181	SI Pump Mini Recirculation Valve Actuator Replacement	0
71111.21N.02	Engineering Changes	PC/M-01-014	MOV-3-843A/B and MOV-3-869 Modification SI System Enhancement	0
71111.21N.02	Engineering Changes	PC/M-06-041	MOV-3-843A/B close circuit- switch change	0
71111.21N.02	Engineering Evaluations	7-3071-3, 7-3071-4	Containment Isolation/Purge Valve Analysis Florida Power & Light Turkey Point Units 3 & 4 54” RIA Valve W/Pratt 2-	9/16/1981

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			Plate Cylinder Operator	
71111.21N.02	Engineering Evaluations	CN-FSE-06-64	RHR Miniflow Loop Heatup Time	1
71111.21N.02	Engineering Evaluations	JPN-PTN-SEMP-94-029	NRC Generic Letter 89-10 MOV Summary Report	2
71111.21N.02	Engineering Evaluations	JPN-PTN-SEMP-95-002	GL 89·10 MOV PROGRAM LOAD SENSITIVE BEHAVIOR ENGINEERING EVALUATION	1
71111.21N.02	Engineering Evaluations	PTN-BFSM-02-010	AOV Program - Containment Purge Exhaust Valve/Actuator Capability	Rev 0
71111.21N.02	Engineering Evaluations	PTN-ENG-SEMS-13-003	Evaluation of JOG MOV Periodic Verification Classification and Impact on the PTN GL 89-10/96-05 MOV Program	0
71111.21N.02	Miscellaneous	5610-050-DB-001	RESIDUAL HEAT REMOVAL SYSTEM	10/25/2016
71111.21N.02	Miscellaneous	5610-050-DB-002	RESIDUAL HEAT REMOVAL SYSTEM	04/17/2018
71111.21N.02	Miscellaneous	5610-062-DB-001	SAFETY INJECTION SYSTEM	15
71111.21N.02	Miscellaneous	5610-062-DB-002	SAFETY INJECTION SYSTEM	17
71111.21N.02	Miscellaneous	5610-M-83	Specification for Containment Purge Butterfly Valve	Rev 1
71111.21N.02	Miscellaneous	Equipment Specification 676258	Motor Operated Valves	2
71111.21N.02	Miscellaneous	FOP-91-035	Turkey Point Units 3 & 4, Potential Leakpath to Atmosphere via RWST during "Piggyback" Mode CTRAC-91-0603-34, File No. REA-92-018	08/18/1994
71111.21N.02	Miscellaneous	L-75-247	Turkey Point Plant Units 3 & 4, Docket Nos. 50-250 & 50-251, ECCS Analysis	05/21/1975
71111.21N.02	Miscellaneous	L-79-293	Turkey Point Units 3 & 4, Docket Nos. 50- 250 & 50-251, NUREG-0578	10/22/1979
71111.21N.02	Miscellaneous	SPEC-M-097	Valve Packing Selection, Configuration, and Use at FPL/FPLE Nuclear Plants	0
71111.21N.02	Miscellaneous	STD-M-003	Engineering Guidelines for Sizing and Evaluation of Limitorque Motor Operators	7
71111.21N.02	Miscellaneous	V000244	Vendor manual Henry Pratt, Installation and Service Instructions Butterfly Valves - 48 and 54 Inch	Rev 9
71111.21N.02	Miscellaneous	V000368	Vendor Manual Aloyco Manually Operated Gate And Globe Valves and Self Actuated Swing Check Valves	Rev 7

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.21N.02	Miscellaneous	VTM V000250	Copes Vulcan: Installation, Operation & Maintenance Instructions -Diaphragm Operator Valves	15
71111.21N.02	Miscellaneous	VTM Z736	VALVE, DOUBLE DISC GATE.	4
71111.21N.02	Miscellaneous	VTM: V000202	Velan Valve Corp: Instruction Manual - Motor Operated and Manual Valves	4
71111.21N.02	Miscellaneous	VTM: V000218	Limitorque: Limitorque Instruction and Maintenance Manual	10
71111.21N.02	Miscellaneous	VTM: V000247	Henry Pratt Co: Operation and Maintenance Manual for Rubber Seated Butterfly Valves.	15
71111.21N.02	Procedures	0-ADM-502	In-Service Testing (IST) Program	2
71111.21N.02	Procedures	0-ADM-541	Air Operated Valve Program	11
71111.21N.02	Procedures	3-EOP-ES-1.3	Transfer to Cold Leg Recirculation	7
71111.21N.02	Procedures	3-OSP-041.4	Overpressure Mitigating System Nitrogen Backup Leak and Functional Test	11
71111.21N.02	Procedures	3-OSP-050.2A	Residual Heat Removal Train A Test- Standby Alignment	10A, completed 09/16/2022
71111.21N.02	Procedures	3-OSP-062.2C	Safety Injection System Inservice Valve Testing	8, completed 06/17/2022
71111.21N.02	Procedures	4-OP-050	Residual Heat Removal System	Rev 26
71111.21N.02	Procedures	4-OSP-206.2	Quarterly Inservice Valve Testing	35
71111.21N.02	Procedures	ER-AA-100	Air Operated Valve Program	4
71111.21N.02	Procedures	ER-AA-100-1000	Air Operated Valve Program Implementation	1
71111.21N.02	Procedures	MA-AA-100-1013	Air Operated Valve Diagnostic Testing and Inspection	6
71111.21N.02	Procedures	STD-M-034	AOV Design, Maintenance, and Testing, St. Lucie & Turkey Point Nuclear Plants	2
71111.21N.02	Work Orders		40432482 01, 40432482 02, 40147198 69, 40446332 01, 40002201 01, 40065687 01, 40711443 01, 40782930 01, 40147198 65, 40035082 01, 40754231 01, 40668730 01, 40495703 01, 40727797 01, 40724456 01, 40466142 01, 40654417 02, 40654417 01, 40654417 03, 40654417 01, 40654417 06, 40666337 05, 40799117 01, 40739919 02, 40810402 01, 40757642 01, 40809740 01, 40542488 01	
71111.21N.02	Work Orders	40748369 01	POV-4-4883: Low Margin for IST Closure Time	02/22/2021
71111.21N.02	Work Orders	WO 30019100 01	MOV-4-863B DP Test in Motor Operated Valve Operator or	10/28/2003

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			RHR Pump Recirc to RWST	
71111.21N.02	Work Orders	WO 37018290 01	Actuator for Containment Purge Air Return Piston Operated Valve POV-3-2602 Overhaul AOV Actuator	10/6/2007
71111.21N.02	Work Orders	WO 40012863 01	POV-3-2602: Degraded Liner ID'D (Replace Seat)	7/23/2012
71111.21N.02	Work Orders	WO 40189381 01	EQ-MOV-4-863B Operator Overhaul/Static	10/8/2014
71111.21N.02	Work Orders	WO 40326904 01	T.S. POV-3-2602 Position Indication Channel Calibration	11/15/2015
71111.21N.02	Work Orders	WO 40638537 01	POV-4-4883 Replace Local Contact Switch (EOC) AR	05/06/2020
71111.21N.02	Work Orders	WO 40725094 02	RCS PORV Actuator Overhaul/Maintenance	11/10/2021
71111.21N.02	Work Orders	WO 40725094 03	Overpressure Mitigation System Nitrogen Backup Leak and Function Test	11/09/2021
71111.21N.02	Work Orders	WO 40754211 01	Inservice Valve Testing Hot Standby to Cold Shutdown MOV-4-863B IST	3/16/2022