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

August 14, 2023

Jamie Coleman
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
7825 River Road, BIN 63031
Waynesboro, GA 30830

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT (VEGP), UNIT 3 – INTEGRATED INSPECTION REPORT 05200025/2023002

Dear Jamie Coleman:

On June 30, 2023, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Vogtle Electric Generating Plant (VEGP), Unit 3. On July 11, 2023, the NRC inspectors discussed the results of this inspection with Mr. Glen Chick, VEGP Units 3 & 4 Executive Vice President 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. One of these findings involved a violation of NRC requirements. We are treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violation or the significance or severity of the violation 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 Vogtle Electric Generating Plant (VEGP), Units 3 & 4.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the 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 Vogtle Electric Generating Plant (VEGP), 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,

Bradley J. Davis Signed by Davis, Bradley on 08/14/23

Bradley J. Davis, Chief
Construction Inspection Branch 2
Division of Construction Oversight

Docket No. 05200025
License No. NPF-91

Enclosure:
As stated

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SUBJECT: VOGTLE ELECTRIC GENERATING PLANT (VEGP), UNIT 3 – INTEGRATED INSPECTION REPORT 05200025/2023002 Dated August 14, 2023

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05200025
License Number:	NPF-91
Report Number:	05200025/2023002
Enterprise Identifier:	I-2023-002-0066
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Vogtle Electric Generating Plant (VEGP), Unit 3
Location:	Waynesboro, GA
Inspection Dates:	April 01, 2023, to June 30, 2023
Inspectors:	J. Eargle, Senior Resident Inspector J. England, Sr. Construction Inspector P. Gresh, Emergency Preparedness Inspector B. Griman, Resident Inspector B. Kemker, Senior Resident Inspector A. Ponko, Sr. Construction Inspector J. Walker, Sr Emergency Preparedness Inspector
Approved By:	Bradley J. Davis, Chief Construction Inspection Branch 2 Division of Construction Oversight

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Vogtle Electric Generating Plant (VEGP), Unit 3, 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 Adequately Implement Design Control Measures Resulting in Lack of Technical Justification for Support SV3-1222-SH-E804.			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05200025/2023002-02 Open/Closed	[H.6] - Design Margins	71152A
The inspectors identified a finding of very low safety significance (Green) with an associated NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to adequately implement measures to assure the design basis was correctly translated into calculation APP-1220-SHC-301 for the as-built configuration of support SV3-1222-SH-E804, which called into question the ability of the support to perform its safety related functions.			

Failure to Correctly Implement an Engineering Design Change for Updating Protective Relay Settings on Medium Voltage Switchgear Buses ES-4 and ES-6			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05200025/2023002-03 Open/Closed	[H.3] - Change Management	71153
A finding of very low safety significance (GREEN) was self-revealed when a valid automatic reactor trip signal was actuated upon the loss of power to two RCPs. The licensee failed to correctly implement an engineering design change for updating protective relay settings on medium voltage switchgear buses ES-4 and ES-6. No violation of regulatory requirements was identified.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
URI	05200025/2023002-01	Maintenance Rule Evaluations for Plant Level Events	71111.12	Open
LER	05200025/2023002-00	LER 2023-002-00 for Vogtle, Unit 3, Automatic RPS Actuation Durin9 Mode 1 Due to Incorrect Relay Settings Caused by Less Than Adequate Questioning Attitude, Validation of	71153	Closed

		Assumptions, and Interface/Guidance		
LER	05200025/2023-003-00	LER 2023-003-00 for Vogtle Electric Generating Plant (VEGP), Unit 3, Automatic Reactor Protection System Actuation During Startup Testing Due to Incorrect Turbine Control Valve Setting	71153	Closed

PLANT STATUS

At the start of this inspection period, Unit 3 was in Mode 1 (Power Operation) at about 18% power and the licensee completed corrective maintenance to allow synchronizing the main generator to the electrical grid and continuation of plant startup testing activities.

On April 1, at 4:22 a.m., the Unit 3 main generator was synchronized to the electrical grid for the first time. The licensee raised reactor power to 25% to perform power ascension testing at the 25% power testing plateau. On April 8, the licensee completed the remote shutdown workstation startup test procedure to demonstrate the ability of plant operators to conduct a remote hot shutdown and the ability to maintain the plant in Mode 3 (Hot Standby) for a simulated main control room evacuation. Operators manually tripped the reactor and the unit entered Mode 3. Following the test, plant operators transferred control of the plant from the remote shutdown workstation back to the main control room.

On April 9, Unit 3 entered Mode 2 (Startup) and the licensee performed a reactor startup. The main generator was synchronized to the electrical grid on April 10 and the licensee continued with plant startup testing activities.

On April 10, with Unit 3 at 18% power, the reactor automatically tripped due to low reactor coolant flow due to voltage decaying to the reactor coolant pumps during main generator testing activities. The trip was not complex and all safety related systems responded normally post-trip. Plant operators stabilized the plant in Mode 3 on natural circulation flow. Unit 3 remained in Mode 3 while the licensee performed corrective maintenance activities.

On April 15, Unit 3 entered Mode 2 and the licensee performed a reactor startup. The main generator was synchronized to the electrical grid on April 16 and the licensee continued with plant startup testing activities at about 40% power.

On April 22, the licensee raised reactor power to 50% to perform power ascension testing at the 50% power testing plateau. On May 1, the licensee raised reactor power to 75% to perform power ascension testing at the 75% power testing plateau.

On May 2, with Unit 3 at about 77% power, three feedwater heater strings sequentially isolated requiring plant operators to manually trip the main turbine. The rapid power reduction system actuated as designed to lower reactor power. The turbine trip caused a sudden change in steam flow to the main condenser and feedwater heaters, which caused corrosion products to become displaced, clogging of all 3 main feedwater/booster pumps suction screens and loss of the pumps. Plant operators manually tripped the reactor from 14% power prior to an automatic reactor trip on low steam generator levels. All safety related systems responded normally post-trip. Unit 3 remained in Mode 3 while the licensee performed corrective maintenance activities.

On May 16, Unit 3 entered Mode 2 and the licensee performed a reactor startup. The main generator was synchronized to the electrical grid on May 17. On May 19 the licensee raised reactor power to 75% to resume power ascension testing at the 75% power testing plateau.
On May 23, the licensee raised reactor power to 90% to perform power ascension testing at the 90% power testing plateau. On May 26, the licensee raised reactor power to 98% and continued with power ascension testing activities.

On May 29, at 4:26 a.m., the licensee raised reactor power to 100% for the first time to begin power ascension testing at the 100% power testing plateau. On June 4, the licensee reduced power to about 75% to perform planned testing involving the removal of feedwater heaters. The unit was returned to 100% power later that day. On June 6, the licensee performed a test to verify the ability of the plant's automatic control systems to sustain a reactor trip transient from 100% power. Plant operators manually tripped the reactor and the unit entered Mode 3 as planned.

On June 7, Unit 3 entered Mode 2 and the licensee performed a reactor startup. The main generator was synchronized to the electrical grid on June 8 and the licensee raised reactor power to about 33%. Power was held at about 33% to troubleshoot and correct a rod control system malfunction, which prevented power ascension.

On June 10, the licensee raised reactor power to 100% to resume power ascension testing at the 100% power testing plateau. Later that day, the licensee performed a test to evaluate the dynamic response of the plant to a main generator trip. Plant operators manually tripped the main generator output breaker and reactor power was stabilized at about 20% power. This test was expected to be a generator trip without a turbine or reactor trip to test proper operation of automatic control systems. The turbine unexpectedly tripped due to a high moisture separator reheater shell tank level. The licensee suspended startup testing and entered a maintenance outage to perform various corrective and planned maintenance activities. Plant operators manually tripped the reactor and Unit 3 entered Mode 3.

On June 11, the licensee performed a plant cooldown and Unit 3 entered Mode 5 (Cold Shutdown). On June 20, following planned and corrective maintenance activities requiring cold shutdown conditions, the licensee commenced a plant heat up and Unit 3 entered Mode 4 (Safe Shutdown). On June 21, Unit 3 entered Mode 3 to perform control rod testing. On June 24, the licensee performed a plant cooldown and Unit 3 returned to Mode 4 to complete the remaining maintenance outage activities. At the end of this inspection period, the unit was in Mode 4 and the licensee was completing maintenance.

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 performed activities described in IMC 2515, Appendix D, "Plant Status," observed risk significant activities, and completed on-site portions of IPs. 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

71111.01 - Adverse Weather Protection

Seasonal Extreme Weather Sample (IP Section 03.01) (1 Sample)

The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of seasonal summer temperatures.

(1) Passive containment cooling and the standby diesel fuel oil systems during the week of May 30.

71111.04 - Equipment Alignment

Partial Walkdown Sample [AP1000] (IP Section 03.01) (1 Sample)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Train 'A' passive containment cooling system on May 16.

71111.05 - Fire Protection

Fire Area Walkdown and Inspection Sample [AP1000] (IP Section 03.01) (1 Sample)

The inspectors evaluated the implementation of the fire protection program by conducting a walkdown and performing a review to verify program compliance, equipment functionality, material condition, and operational readiness of the following fire areas:

(1) Division A, B, C, & D instrumentation and control/penetration rooms on May 11.

71111.07A - Heat Exchanger/Sink Performance

Annual Review [AP1000] (IP Section 03.01) (1 Sample)

The inspectors reviewed the readiness and availability of the following heat exchanger and/or heat sink:

(1) Ultimate heat sink (passive containment cooling system) on April 18.

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the control room during performance of 3-GOP-306, "Plant Startup Mode 2 to 25% Power" on April 15.

71111.12 - Maintenance Effectiveness

Maintenance Effectiveness [AP1000] (IP Section 03.01) (1 Sample)

(1) The inspectors reviewed CAR 411146, “Unit 3 Reactor and Turbine Tripped Multiple Times Resulting in Challenges to Startup,” which collectively evaluated multiple main turbine and reactor trips during startup testing in March and April 2023.

Quality Control [AP1000] (IP Section 03.02) (1 Sample)

The inspectors evaluated the effectiveness of maintenance and quality control activities to ensure the following SSC remains capable of performing its intended function:

(1) Commercial grade dedication of SUA145 undervoltage relay used in Class 1E DC switchboards during the weeks of May 28 and June 11.

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management Sample [AP1000] (IP Section 03.01) (1 Sample)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

(1) During maintenance outage activities the week of May 12 while the time to boil in the reactor coolant system was less than 72 hours.

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment [AP1000] (IP Section 03.01) (4 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

(1) 10970365-Unit 3-ODS, "SG-2 MFW Line Temperature Abnormal Rise" during the week of May 21
(2) 10958075-Unit 3-ODS, "Part 21 Issued by Trillium Valves Identified Defects in Butterfly Valves with Limitorque SMB Motor Actuators Supplied to Westinghouse Electric Company from 2010 to 2016" during the weeks of May 7, June 4, and June 12
(3) 1093212-Unit 3-ODS, "Potential - IEEE 384 Violation 11202" during the week of June 25
(4) 10977659-Unit 3-ODS, "Main Control Room Temperature Limit Exceeded" during the week of June 25

71111.20 - Refueling and Other Outage Activities

Refueling/Other Outage Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated maintenance outage activities from June 10, through July 5.

71111.24 - Testing and Maintenance of Equipment Important to Risk

The inspectors evaluated the following testing and maintenance activities to verify system operability and/or functionality:

Post-Maintenance Testing (PMT) [AP1000] (IP Section 03.01) (2 Samples)

(1) 3-FWS-V012A startup feedwater pump ARC valve repair on June 14
(2) Division A source range nuclear instrument calibration/repair on June 26

Surveillance Testing [AP1000] (IP Section 03.01) (3 Samples)

(1) 3-GEN-ITPS-629, "Thermal Power Measurement and Statepoint Data Collection Startup Test Procedure", during the weeks of April 23, May 14, May 21, and May 28
(2) 3-GEN-ITPS-640, "Remote Workstation Startup Test Procedure," during the week of April 2
(3) 3-RCS-ITPS-605, "RCS Flow Measurement at Power Startup Test Procedure," during the weeks of April 23, May 14, and June 4

71114.02 - Alert and Notification System Testing

Inspection Review (IP Section 02.01-02.04) (1 Sample)

(1) The inspectors evaluated the maintenance and testing of the alert and notification system during the week of April 10, 2023.

71114.03 - Emergency Response Organization Staffing and Augmentation System

Inspection Review (IP Section 02.01-02.02) (1 Sample)

(1) The inspectors evaluated the readiness of the Emergency Response Organization during the week of April 10, 2023.

71114.04 - Emergency Action Level and Emergency Plan Changes

Inspection Review (IP Section 02.01-02.03) (1 Sample)

(1) The inspectors evaluated submitted Emergency Action Level (EALs), Emergency Plan, and Emergency Plan Implementing Procedure changes during the week of April 10, 2023. This evaluation does not constitute NRC approval.

71114.05 - Maintenance of Emergency Preparedness

Inspection Review (IP Section 02.01 - 02.11) (1 Sample)

(1) The inspectors evaluated the maintenance of the emergency preparedness program during the week of April 10, 2023.

OTHER ACTIVITIES – BASELINE

71152A - Annual Follow-up Problem Identification and Resolution

Annual Follow-up of Selected Issues (Section 03.03) (2 Samples)

The inspectors reviewed the licensee’s implementation of its corrective action program related to the following issues:

(1) Failure to Adequately Implement Design Control Measures Resulting in Lack of Technical Justification for Support SV3-1222-SH-E804
(2) Flowserve P44 Valve Failures

71152S - Semiannual Trend Problem Identification and Resolution

Semiannual Trend Review (Section 03.02) (1 Sample)

The inspectors reviewed repetitive or closely related issues documented in the licensee's CAP during the first and second quarters of 2023 to look for trends not previously identified by the licensee.

(1) Assessment and Observations

The inspectors determined the licensee's trending program was generally effective at identifying, monitoring, and correcting adverse performance trends before they could become more significant safety problems. The inspectors' evaluation did not reveal any new trends that would indicate a more significant safety issue. The inspectors determined, in most cases, issues were appropriately evaluated by the licensee's staff for potential trends at a low threshold and resolved within the scope of the CAP.

The inspectors identified a trend associated with an increase in CRs for procedure corrections required. The inspectors noted there were multiple CRs to update operating procedures CVS-SOP-001 and 3-RCS-SOP-001 to address thermal shock of RCS piping. The inspectors also noted there were multiple CRs to correct incorrect components and line ups for operating procedure 3-IDSD-SOP-001. The inspectors also noted 15 severity level 2 CRs related to battery performance.

71153 - Follow Up of Events and Notices of Enforcement Discretion

Event Follow-up [AP1000] (IP section 03.01) (2 Samples)

(1) Reactor Trip Response

On April 10, at 0048, with Unit 3 in Mode 1 at 18% power, the reactor automatically tripped due to low reactor coolant flow due to voltage decaying to the reactor coolant pumps (RCPs) during main generator testing activities. All safety related systems responded normally post-trip. Operators stabilized the plant on natural circulation flow with decay heat being removed by discharging steam via the steam generator power operated relief valves to atmosphere.

At the time of the trip, the licensee was performing switchyard circuit breaker testing. The expected plant response when opening breakers between the main turbine generator and the switchyard was a main turbine generator runback with the plant's electrical distribution system on island mode (i.e., with the main turbine disconnected from the switchyard and supplying house loads through the unit auxiliary transformers). After opening switchyard breakers 161750 and 161850, the main turbine control system was not able to maintain turbine speed causing house load voltage to lower until residual bus transfer from the unit auxiliary transformers to the reserve auxiliary transformers was initiated.

Preliminary investigation indicated that after breaker 161850 was opened, turbine control valves closed and then reopened and stabilized; however, turbine speed continued to decrease. The generator could no longer maintain proper voltage and frequency to the RCPs and as RCP speed fell below 90% a reactor trip followed by a turbine trip occurred and the main generator circuit breaker opened.

The inspectors observed operator actions post-trip, interviewed plant personnel, performed plant tours, and reviewed operator logs to evaluate operator actions during the event.

(2) Reactor Trip Response

On May 2, with Unit 3 in Mode 1 at approximately 77% reactor power, three feedwater heater strings sequentially isolated requiring a manual turbine trip. Operators manually tripped the main turbine as directed by procedure based on the loss of two or more feedwater heater strings. The rapid power reduction system actuated as designed to lower reactor power. The turbine trip caused a sudden change in steam flow to the main condenser and feedwater heaters (i.e., extraction steam). This caused corrosion products to become displaced, resulting in the clogging of all 3 main feedwater/booster pumps suction screens. With high differential pressures across the suction screens, the operators had to trip the reactor as directed by procedure and secure the main feedwater/booster pumps.

Operators manually tripped the reactor from 14% power prior to an automatic reactor trip on low steam generator levels. All safety related systems responded normally post-trip. Operators stabilized the plant with decay heat being removed by discharging steam via the turbine bypass valves to the main condenser.

The inspectors observed operator actions post-trip, interviewed plant personnel, performed plant tours, and reviewed operator logs to evaluate operator actions during the event.

Event Report [AP1000] (IP section 03.02) (2 Samples)

The inspectors evaluated the following licensee event reports (LERs):

(1) LER 05200025/2023-002-00, "Automatic Reactor Protection System Actuation During Mode 1 Due to Incorrect Relay Settings Caused by Less Than Adequate Questioning Attitude, Validation of Assumptions, and Interface/Guidance." (ADAMS Accession No. ML23135A768)
(2) LER 05200025/2023-003-00, "Automatic Reactor Protection System Actuation During Startup Testing Due to Incorrect Turbine Control Valve Setting." (ADAMS Accession No. ML23159A223) The inspectors determined that the cause of the condition described in the LER was not reasonably within the licensee's ability to have foreseen and corrected and therefore was not reasonably preventable. No performance deficiency nor violation of NRC requirements was identified.

INSPECTION RESULTS

Unresolved Item (Open)	Maintenance Rule Evaluations for Plant Level Events URI 05200025/2023002-01	71111.12
Description: Four main turbine trips and one reactor trip occurred during plant startup testing in March 2023. A fifth main turbine trip and a second reactor trip occurred in April 2023. The first turbine trip on March 15 was also the precursor to a reactor trip. The direct cause for the		

turbine trip was determined to be incorrect wiring of the auto voltage regulator current transformer protective relays (reversed polarity) by the vendor. On March 22, while attempting to close the main generator output breaker to synchronize the main generator to the electrical grid, the turbine tripped on reverse power. The direct cause of this second turbine trip was determined to be incorrect wiring of the generator circuit breaker current transformers (also reversed polarity). On March 28, while attempting to close the main generator output breaker to synchronize the main generator to the electrical grid, the turbine once again tripped on reverse power. The main turbine load control logic failed to actuate/take control of the turbine. The direct cause of this third turbine trip was determined to be wiring to connect the generator circuit breaker to the plant control system had not been installed. On March 30, the turbine tripped due to a turbine control and protection system logic problem that prevented the turbine control valves to remain open when the turbine entered load control upon closure of the generator circuit breaker. The fifth turbine trip occurred on April 10 and was the result of a reactor trip. The direct cause for the reactor/turbine trip was determined to be incorrect turbine control logic. The turbine control valve controller setting was too low, causing insufficient steam flow to the turbine resulting in decreasing turbine speed. Of the five turbine trips discussed above, only the third one was reasonably within the licensee's ability to have prevented; however, the inspectors determined the performance issue was of minor significance since there were no adverse consequences from the turbine trip and the purpose of the testing at the time was to identify proper operation of the turbine control system. The inspectors requested to review the licensee's maintenance rule evaluations associated with the above turbine and reactor trips. In response to the inspectors' questions, the licensee found that only the March 15 reactor trip and March 22 turbine trip had been evaluated. No evaluations had been performed for system or component failures associated with any of the other four turbine trips or for the April 10 reactor trip. This issue is considered to be an unresolved item pending the inspectors' review of the licensee's completed maintenance rule evaluations and condition report evaluations to determine whether a performance deficiency or violation of regulatory requirements exists. Planned Closure Actions: NRC subject matter experts will review the licensee's evaluation of the issue and will document the results in a subsequent Vogtle Unit 3 integrated inspection report. Licensee Actions: The licensee entered this issue into its CAP to evaluate the causes and implement corrective actions. Corrective Action References: CRs 10975594, 10978631 and 10978278

Failure to Adequately Implement Design Control Measures Resulting in Lack of Technical Justification for Support SV3-1222-SH-E804.			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05200025/2023002-02 Open/Closed	[H.6] - Design Margins	71152A

The inspectors identified a finding of very low safety significance (Green) with an associated NCV of 10 CFR 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to adequately implement measures to assure the design basis was correctly translated into calculation APP-1220-SHC-301 for the as-built configuration of support SV3-1222-SH-E804, which called into question the ability of the support to perform its safety related functions.

Description:

During the week of January 30, 2023, the inspectors observed 4-inch diameter safety related rigid conduits SV4-1222-ER-BXC03 and SV4-1222-ER-BXC04 supported by seismic category I support SV4-1222-SH-E804 in Unit 4 room 12207. These conduits carry cables associated with safety related electrical equipment. Specifically, AP1000 Tag No. IDSB-SB-2A, which is a 125 volt 60 cell battery in Division B of the Class 1E DC and UPS system. IDSB-SB-2A is part of the second battery bank in Division B, which is designated as the 72-hour battery bank. The second battery bank is used for those loads requiring power for 72 hours following an event of loss of all ac power sources concurrent with a design basis accident (DBA). The Class 1E DC and UPS system provides reliable power for the safety related equipment required for the plant instrumentation, control, monitoring, and other vital functions needed for shutdown of the plant. In addition, the Class 1E DC and UPS system provides power to the normal and emergency lighting in the main control room and at the remote shutdown workstation. IDSB-DB-2A supplies inverter IDSB-DU-2, which supplies panel IDSB-EA-3. This panel supplies power to the post-accident monitoring instrumentation system (PAMS), which provides the capability to monitor plant variables and systems operating status during and following an accident. PAMS also includes those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials.

Support SV4-1222-SH-E804 is specified on drawing SV4-1222-ER-619 and consists of two Unistrut conduit clamps attached to a single Unistrut channel welded to a steel filler plate, which is, in turn, welded to another steel plate attached to a primary structural wall. The configuration at the support consists of a rigid conduit section supported approximately at its midpoint with flexible conduits attached to each cantilevered end of the rigid conduit section. Given the as-built teeter-totter configuration of the support, which could result in rotational forces at the support from seismic loads, the inspectors questioned the use of a single support for the rigid conduit section at this location. Subsequently, the licensee confirmed that the same configuration exists in VEGP Unit 3 at support SV3-1222-SH-E804.

The structural design of support SV3-1222-SH-E804 is documented in calculation APP-1220-SHC-301. However, the licensee determined, during development of responses to the inspectors’ questions, that the existing analysis did not adequately address the as-built configuration of the conduits and the associated support. As a result, the inspectors concluded that the calculation did not adequately demonstrate that support SV3-1222-SH-E804 would be able to withstand the design basis loads without loss of structural adequacy or any safety related functions.

The licensee initiated Engineering and Design Coordination Reports (E&DCRs) APP-1220-GEF-501, APP-1220-GEF-502, and APP-1220-GEF-503 to evaluate the as-built configuration and address the inspectors’ questions. Each subsequent E&DCR superseded the previous E&DCR and was initiated to respond to questions raised by the inspectors on the preceding E&DCR. In all the E&DCRs, the licensee completed both a hand calculation and an analysis using the GT STRUDL structural analysis and design software program and compared the results between the two methods.

The inspectors reviewed APP-GW-S1-006, “Design Guide for Raceway Systems,” Revision 4 and noted that in Section 4 it is stated that the basic stress allowables for conduit supports utilizing light gage cold rolled channel type sections are based on the manufacturer’s published catalog values and the basic stress allowables for conduit supports utilizing structural shapes are in accordance with ANSI/AISC N-690. The inspectors further determined that the basic stress allowables for Unistrut components are summarized in Annex C, “Guidance for structural acceptance criteria for elastic design method,” of calculation APP-SH25-S3C-002, “AP1000 Seismic Category I Standard Conduit Supports.” The inspectors also noted that supports are evaluated in APP-SH25-S3C-002 to verify compliance with IEEE 628-2001 (R2006), “IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations.” Similarly, E&DCR APP-1220-GEF-503 states that SCI raceway systems for Class 1E cables shall comply with IEEE 628.

The inspectors reviewed manufacturer’s catalog information and E&DCRs APP-1220-GEF-501, APP-1220-GEF-502, and APP-1220-GEF-503. The inspectors determined that adequate technical justification was not provided in the E&DCRs for some of the assumptions used to evaluate the as-built configuration of the conduits. Specifically, the dimensions of the conduit clamp, the seismic forces, and structural acceptance criteria appeared to be nonconservative and inconsistent with the manufacturer’s catalog information, design guide APP-GW-S1-006, calculations APP-1220-SHC-301 and APP-SH25-S3C-002, and IEEE 628.

E&DCRs APP-1220-GEF-501, APP-1220-GEF-502, and APP-1220-GEF-503 lacked adequate technical justification for some of the assumptions used to evaluate the as-built configuration of the conduits. The following discussion, however, focuses on E&DCR APP-1220-GEF-503 since it documents the most current iteration of the analysis and design of support SV3-1222-SH-E804.

In E&DCR APP-1220-GEF-503, the licensee assumed that the in-plane rotational forces at the support would be resisted by a force couple developed by shear in the conduit clamp bolts. The licensee also assumed the distance between the bolts to be equal to the end to end length of the conduit clamp. Based on the manufacturer’s catalog information, however, the bolts are located 11/16 of an inch from each end of the conduit clamp, which would decrease the distance between the bolts by approximately 20%. Accounting for the reduction in distance between the bolts would lead to a corresponding increase in the in-plane rotational forces.

In E&DCR APP-1220-GEF-503, the licensee calculated the seismic forces using the equivalent static load method of analysis. This method is typically used for simple systems with a factor of 1.5 conservatively applied to the peak acceleration to account for multi-mode effects. The licensee assumed this method was appropriate for calculating the seismic forces without providing justification demonstrating that it is applicable or conservative for this specific case. Given the teeter-totter configuration of the support, however, this method of accounting for seismic forces potentially underestimates the rotational forces due to seismic loads, which could control the design of the support. Along these lines, the inspectors noted that IEEE-628, Subclause 4.10.3, “Structural Analysis,” allows both dynamic and equivalent static load analysis for calculating the effects of dynamic loads on raceway systems, but also states that the selection of the analysis method shall take into account the complexity of the system and the adequacy of the analytical technique to properly predict the response of the system while under dynamic excitation and other dynamic loads.

In many instances, the structural acceptance criteria assumed in E&DCR APP-1220-GEF-503 deviated from those provided in design guide APP-GW-S1-006, calculations APP-1220-SHC-301 and APP-SH25-S3C-002, and recommended in IEEE Standard 628. Some examples are as follows:

1. In the interaction equations of E&DCR APP-1220-GEF-503, a limit of 1.6 is assumed for load combinations, which include earthquake loads. However, the allowable stresses for the P1000 Unistrut channel provided in APP-1220-SHC-301, Appendix A.2 and assumed in the E&DCR incorporate a factor of 1.6 to account for seismic loading. As a result, the appropriate limit for the interaction equations associated with the Unistrut channel should be 1.0.
2. The allowable stresses provided for the P2558 conduit clamps in APP-SH25-S3C-002, Annex C are based on 50% of the average ultimate load capacity. Accounting for the 1.6 limit, the allowable stresses assumed in E&DCR APP-1220-GEF-503, however, are approximately 28% higher.
3. The allowable stresses for the Unistrut bolting hardware provided in APP-SH25-S3C-002, Annex C, are 53% of the ultimate capacity of the fastener and include a 1.6 increase factor to account for seismic loads. However, the allowable stresses assumed in E&DCR APP-1220-GEF-503 are based on the full ultimate capacity of the fastener and are compared to a limit of 1.6 in the interaction equations. In effect, the allowable stresses assumed for the bolts in E&DCR APP-1220-GEF-503 exceed the ultimate capacities of the fasteners. As a result, the allowable stresses assumed for the bolts in E&DCR APP-1220-GEF-503 are nonconservative and do not comply with APP-GW-S1-006 since they exceed the manufacturer's published values. Additionally, the allowable stresses assumed for the bolts in E&DCR APP-1220-GEF-503 are inconsistent with IEEE-628, which recommends in Annex C, "Guidance for structural acceptance criteria for elastic design method," that the maximum allowable stresses should not exceed 0.9 time the yield strength of the material and the allowable load should be two-thirds of the ultimate load or a load corresponding to one-half of the displacement at the ultimate load, whichever is smaller.

The licensee did not provide adequate technical justification in E&DCR APP-1220-GEF-503 for these examples where the structural acceptance criteria used to evaluate the as-built configuration of support SV3-1222-SH-E804 deviated from those provided in design guide APP-GW-S1-006, calculations APP-1220-SHC-301 and APP-SH25-S3C-002, and recommended in IEEE 628. The inspectors also noted that the cumulative impacts from inconsistencies in the spacing of the bolts, the potential underestimation of the rotational forces on the support due to seismic loads, and inadequately justified deviations in the structural acceptance criteria could result in stress ratios exceeding the allowable limits. As a result, the inspectors concluded that the ability of Support SV3-1222-SH-E804 to withstand the design basis loads without loss of structural integrity or any safety-related functions was indeterminate. The licensee generated CR 10975944 and concluded that the existing analysis indicated the support will be acceptable as is, and the next update of the analysis will resolve the inspectors' issues.

Corrective Action References: The licensee entered this violation into its CAP as CRs 10948139, 10958697, and 10966910 to evaluate the cause and to identify appropriate corrective actions.

<u>Performance Assessment</u>:

Performance Deficiency: The inspectors determined that the licensee's failure to have an analysis demonstrating that the as-built configuration of support SV3-1222-SH-E804 would meet design requirements was a performance deficiency and violation of 10 CFR 50, Appendix B, Criterion III, warranting a significance evaluation.

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.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix G, "Shutdown Safety SDP."

Cross-Cutting Aspect: H.6 - Design Margins: The organization operates and maintains equipment within design margins. Margins are carefully guarded and changed only through a systematic and rigorous process. Special attention is placed on maintaining fission product barriers, defense-in-depth, and safety related equipment.

Enforcement:

Violation: 10 CFR 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled.

APP-GW-S1-006, "AP1000 Design Guide for Raceway Systems," Section 4.0, states, in part, that the basic stress allowables for conduit supports utilizing light gage cold rolled channel type sections are based on the manufacturer's published catalog values. These allowable stresses are summarized in Annex C, "Guidance for structural acceptance criteria for elastic design method," of calculation APP-SH25-S3C-002, "AP1000 Seismic Category I Standard Conduit Supports." The design of conduit support SV3-1222-SH-E804 is documented in calculation APP-1220-SHC-301, "Structural Analysis of Cable Conduit Supports in Auxiliary Building, Areas 1 and 2, El. 82'-6"."

Contrary to the above, on or before April 20, 2023, the licensee failed to adequately implement measures to assure that the design basis was correctly translated into calculation APP-1220-SHC-301 for the as-built configuration of support SV3-1222-SH-804. Specifically, the analysis and design of support SV3-1222-SH-E804 documented in calculation APP-1220-SHC-301 did not adequately account for the as-built configuration of the conduits attached to support SV3-1222-SH-E804. Moreover, E&DCRs APP-1220-GEF-501, APP-1220-GEF-502, and APP-1220-GEF-503 completed to evaluate the as-built configuration used nonconservative dimensions for the conduit clamps and did not provide adequate technical justification for the revised assumptions used to determine the seismic loads and deviations from the structural acceptance criteria provided in design guide APP-GW-S1-006, calculations APP-1220-SHC-301 and APP-SH25-S3C-002, and recommended in IEEE Standard 628.

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

Failure to Correctly Implement an Engineering Design Change for Updating Protective Relay Settings on Medium Voltage Switchgear Buses ES-4 and ES-6			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05200025/2023002-03 Open/Closed	[H.3] - Change Management	71153
A finding of very low safety significance (GREEN) was self-revealed when a valid automatic reactor trip signal was actuated upon the loss of power to two RCPs. The licensee failed to correctly implement an engineering design change for updating protective relay settings on medium voltage switchgear buses ES-4 and ES-6. No violation of regulatory requirements was identified.			

Description: On March 15, 2023, the Vogtle Unit 3 reactor automatically tripped from about 18% power, during startup testing, due to loss of power to two of the four RCPs. The licensee was attempting to test the automatic voltage regulator by closing the field circuit breaker that provides excitation to the main generator. A main generator excitation protective relay tripped causing the main turbine to trip and switchgear buses ES-1, 2, 3, 4, 5, and 6 to attempt to fast transfer from the unit auxiliary transformers to the reserve auxiliary transformers. Buses ES-2, 4, and 6 failed to fast bus transfer as designed. Due to the loss of power to ES-4 and ES-6, RCPs 1B and 2B lost power. With reactor power above 10% (P-10), the loss of power to the two RCPs caused the RCP low speed trip setpoint of < 91% on two of four RCPs to be met, which resulted in a reactor trip with no safeguards actuation. All safety-related systems responded normally post trip. Plant operators stabilized the plant with decay heat being removed by discharging steam via the steam generator power operated relief valves to atmosphere. After verifying no electrical faults were present, operators restored power to ES-4 and ES-6 through the reserve auxiliary transformer.

The licensee's cause evaluation attributed the direct cause of the reactor trip to incorrect settings on protective relays for medium voltage switchgear buses ES-4 and ES-6. The underlying cause was a human performance error coupled with a work management process flaw. On October 11, 2022, Westinghouse issued design change E&DCR APP-ECS-GEF-537 to change protective relay settings for buses ES-1, 2, 3, 4, 5, and 6. On December 5, 2022, work management created work order 1412789 to make the design setting changes. Changes to the relay settings for buses ES-1, 2, 3, 4, 5, and 6 were being implemented under one work order. Individual work orders were not generated for each relay. Instead, one work order was created with the intent to address all six relays. Work management specifically identified bus ES-1 in the work order scope. The relays for buses ES-2 through ES-6 were included as additional equipment in the work order, but they were not included in the work scope. The work management standard for generating/processing work orders allowed multiple components to be addressed in a single work order, but it provided no guidance for conveying this approach for the purpose of planning the work. The normal practice was to have one work order per component. As a result, the implementation of the design change for the relay settings was only implemented on the relay for bus ES-1.

The licensee completed a 4-hour notification call (Event Notification 56414) on March 15 to report the valid reactor protection system actuation while critical as required by 10 CFR 50.72(b)(2)(iv)(B). The licensee submitted LER 05200025/2023-002-00 to report this event in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in automatic actuation of the RPS.

Corrective Actions: The licensee entered this issue into its CAP as CRs 10956663 to evaluate the cause and to identify appropriate corrective actions. The relay settings were subsequently corrected for buses ES-2, 3, 4, 5 and 6.
Performance Assessment: Performance Deficiency: The inspectors determined the licensee's failure to correctly implement an engineering design change for updating protective relay settings on medium voltage switchgear buses ES-4 and ES-6 was a licensee performance deficiency warranting a significance evaluation. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Consistent with the guidance in IMC 0612, "Issue Screening," Appendix B, "Issue Screening Directions," dated August 8, 2022, the inspectors determined the performance deficiency was a finding of more than minor significance because it was associated with the design control attribute of the initiating events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to correctly implement the design change for updating protective relay settings on buses ES-4 and ES-6 resulted in the failure of the buses to fast bus transfer following a main turbine trip, which caused a loss of power to two RCPs and a reactor trip. The inspectors also reviewed the examples of minor issues in IMC 0612, Appendix E, "Examples of Minor Issues," dated January 1, 2021, and found no similar examples. Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." In accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," dated December 20, 2019, Table 3, "SDP [Significance Determination Process] Appendix Router," the inspectors determined this finding would require review using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated November 30, 2020, since it involved a transient initiator with the unit operating at power. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, Exhibit 1, "Initiating Events Screening Questions," and determined this finding would require a detailed risk evaluation because the finding caused a 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 (e.g., loss of condenser, loss of feedwater). The Region II Senior Risk Analyst (SRA) conducted an assessment of the risk significance of the finding using SAPHIRE 8, Version 8.2.6 and the Vogtle 3&4 SPAR Model, Version 8.81, dated August 14, 2022. The SRA conservatively set the exposure time to one month (the actual was from the date of initial criticality on March 6 until the event on March 15, or nine days). The reactor protection system actuation was valid due to the loss of two RCPs at greater than 10% power. The SRA modelled the condition as buses ES-4 and ES-6 being unavailable due to the performance deficiency. The SRA set ECS-BUS-FOP-ES4 and ECS-BUS-FOP-ES6 to true. The dominant accident sequence was a loss of component cooling water, with a failure of a main feedwater pump, failure of primary and secondary relief valves, and a failure of in-containment refueling water storage tank injection and automatic

depressurization system stage four. The change in core damage frequency was less than 1E-7. Based on the results of the detailed risk evaluation, the inspectors determined the finding was of very low safety significance. Cross-Cutting Aspect: H.3 - Change Management: Leaders use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority. The inspectors determined the finding had a cross-cutting aspect of Change Management in the Human Performance area because the proximate cause was attributed to the failure to correctly use a systematic process for implementing the design change. (H.3)
Enforcement: Inspectors did not identify a violation of regulatory requirements associated with this finding.

EXIT MEETINGS AND DEBRIEFS

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

- On July 11, 2023, the inspectors presented the integrated inspection results to Mr. Glen Chick, VEGP Units 3 & 4 Executive Vice President and other members of the licensee staff.
- On April 13, 2023, the inspectors presented the Emergency Preparedness Program Inspection results to Mr. P. Martino and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.01	Corrective Action Documents Resulting from Inspection	CR 1097664	NRC Inspection Comments	
	Procedures	3-DOS-SOP-001	Standby Diesel Fuel Oil System	1.0
		3-EHS-SOP-001	Special Process Heat Tracing System	1.0
		3-PCS-SOP-001	Passive Containment Cooling System	2.0
		B-GEN-OPS-009	Cold Weather Checklist	3.0
71111.04	Drawings	APP-PCS-M6-001	Piping and Instrumentation Diagram Passive Containment Cooling System	12
		APP-PCS-M6-002	Piping and Instrumentation Diagram Passive Containment Cooling System	14
		APP-PCS-M6-003	Piping and Instrumentation Diagram Passive Containment Cooling System	10
		APP-PCS-M6-004	Piping and Instrumentation Diagram Passive Containment Cooling System	10
	Procedures	3-PCS-SOP-001	Passive Containment Cooling System	2.0
71111.05	Corrective Action Documents Resulting from Inspection	CR 10971150	Incorrect fire extinguisher shown on B-PFP-ENG-001-F3113	
	Fire Plans	B-PFP-ENG-001-F3113	Pre-Fire Plan - Auxiliary Building Non-RCA El. 100'0"	2.0
71111.07A	Calculations	APP-PCS-M3C-015	PCS Minimum Cooling Water Flow Rates and Tank Sizing	5
	Procedures	B-ADM-CSP-001	Periodic Analysis Scheduling Program	3.0
		B-PCS-CHM-001	Chemistry of the Passive Containment Cooling Storage Tank (PCCWST)	1.0
71111.11Q	Procedures	3-GOP-306	Plant Startup Mode 2 to 25% Power	M=0.12
71111.12	Corrective Action Documents	CAR 411146	Unit 3 Reactor and Turbine Tripped Multiple Times Resulting in Challenges to Startup	
		CR 10960257	Main Turbine Trip on GCB [Generator Circuit Breaker] BU	

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			86 Lockout	
		CR 10961184	Main Turbine Trip on a GCB BU 86 Lockout	
		CR 10961224	Turbine Reference Load Anomalies During Turbine Trip	
	Corrective Action Documents Resulting from Inspection	CR 10975446	MRule (a)(1) Evaluation Required for PLE [Plant Level Event] on 05/02/2023	
		CR 10975524	MRule (a)(1) for PLE on 05/02/2023	
		CR 10975594	NMP-ES-027 SV34 Procedural Violation Due to Lack of Location Codes	
		CR 10978278	Need Additional Maintenance Rule Evaluation for Event Documented in CR 10956663	
		CR 10978631	Request for MREVAL for 04/10/2023 Turbine and Reactor Trip	
	Engineering Evaluations	EVAL-VEGP34-ECS-05850	(a)(1) Review for Unit 3 Reactor Trip on March 15, 2023	05/25/2023
		TE 1125486	Perform Maintenance Rule Evaluation for CR 10956663 for Unit 3 Reactor Trip	
		TE 1125964	Perform Maintenance Rule Evaluation for CR 10958946 - ZAS	
	Miscellaneous		Maintenance Rule Implementation Guidance for 103(g)	02/14/2022
		Regulatory Guide 1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	4
	Procedures	APP-DS01-V0M-001	AP1000 DS01 Class 1E DC Switchboards - Technical Manual	0
		NMP-ES-027	Maintenance Rule Program	Version 10.4
		NUMARC 93-01	Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	4F
	Shipping Records	Purchase Order # SNG10287998	DC Undervoltage Relay; 120 - 300V	0
		Testing Purchase Order # SNG10288342	Test of UV Relay Model SUA145	11/14/2022
71111.13	Corrective Action Documents Resulting from	CR 10979257	NRC walkdown finding	
		CR 10979556	Shutdown Safety Report Not Generated as Required	06/13/2023
		CR 10979629	Potential Pathways around Protected Equipment Postings	06/13/2023

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Inspection			
	Procedures	3-RNS-SOP-001	Normal Residual Heat Removal System	7.0
		B-ADM-OPS-018	Protected Division and Protected Equipment Program	2.0
		B-ADM-OPS-018	Protected Division and Protected Equipment Program	2.0
		NMP-DP-001	Operational Risk Awareness	24.0
		NMP-GM-031	On-Line Configuration Risk Management Program	9.3
		NMP-GM-031-001	Online Maintenance Rule (a)(4) Risk Calculations	8.0
71111.15	Corrective Action Documents	CR 10970365	SG-2 MFW Line Temperature Abnormal Rise	
	Miscellaneous	VEGP 3&4 Technical Specifications and Bases	Technical Specification 3.7.3, Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Control Valves (MFCVs)	
		VEGP 3&4 UFSAR, Section 10.4.7	Condensate and Feedwater System	11.2
	Operability Evaluations	1093212-Unit 3-ODS	Potential - IEEE 384 Violation 11202	6/29/23
		10958075-Unit 3-ODS	Part 21 Issued by Trillium Valve USA	3/29/23
		10970365-Unit 3-ODS	SG-2 MFW Line Temperature Abnormal Rise	05/19/2023
		10977659-Unit 3-ODS	Main Control Room Temperature Limit Exceeded	06/24/2023
		SVP-SV0-230107	Westinghouse Response to TE 1130298 for Main Control Room Temperature Excursion	06/13/2023
	Procedures	NMP-AD-012	Operability Determinations	16.1
71111.24	Corrective Action Documents	CR 10974118	3-GEN-ITPS-629 Review Criteria No Met at 90% Plateau	
		CR 10975390	Secondary Tuning Required - Consideration for Wider Power Control Bands	
	Corrective Action Documents Resulting from	CR 10976287	Recommend Enhancing 3-PMS-OTS-16-007 to Include a Manual Calculation Method Similar to What Is Performed in 3-GEN-ITPS-629 Att 4 in Case the Calorimetric NAP Is	

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Inspection		Unavailable	
		CR 10976308	WEC to Reason for Deviation Between NAPS and the Manual Calculation Method	
		CR 10979172	Calculation Error Found in Spreadsheet Used for 3-GEN-ITPS-629 Att. 4 TDR#8	
		CR 10979845	3-GEN-ITPS-629 Attachment 6 Not Updated	
	Procedures	3-GEN-ITPS-629	Thermal Power Measurement and Statpoint Data Collection Startup Test Procedure	Version 5.0
		3-GEN-ITPS-640	Remote Workstation Startup Test Procedure	Version 3.0
		3-PMS-OTS-16-028	Division A Source Range Nuclear Instrumentation Calibration	7.0
		3-RCS-ITPS-605	RCS Flow Measurement at Power Startup Test Procedure	Version 4.0
	Work Orders	SNC1499197	Replace 3-PMS-JW-005A Source Range preamplifier Assembly	1.0
71152A	Calculations	APP-1220-SHC-301	Structural Analysis of Cable Conduit Supports in Auxiliary Building, Areas 1 and 2, El. 82'-6	Revision 2
		APP-SH25-S3C-002	AP1000 Seismic Category I Standard Conduit Supports	Revision 3
	Corrective Action Documents	CAR 417576	Stem Broke/Separated from Plug	
		CR 1095154	3-HDS-V012B potential stem/disk separation	
		CR 10958192	MFP 'C' miniflow potential stem/actuator separation	
		CR 10963631	Stem broke/separated from Plug	
		CR 10968674	3-HDS-V012B has a broken stem	
	Corrective Action Documents Resulting from Inspection	CR 10948139	NRC Questions Regarding Support for 2 Conduits in U4 Room 12207	
		CR 10958697	NRC Identified Error in Conduit Support Calculation	
		CR 10966910	Additional NRC Concerns with Analysis of Support 1222-SH-E804	
		CR 10975944	Condition in CR 10966910 Needs Ops Review	
	Engineering Changes	APP-1220-GEF-501	Single Support on Rigid 4" Conduit Lines APP-1222-ER-BXC03 and APP-1222-ER-BXC04	Revision 0
		APP-1220-GEF-502	Single Support on Rigid 4" Conduit Lines APP-1222-ER-BXC03 and APP-1222-ER-BXC04	Revision 0
		APP-1220-GEF-	Single Support on Rigid 4" Conduit Lines APP-1222-ER-	Revision 0

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		503	BXC03 and APP-1222-ER-BXC04	
	Miscellaneous	APP-G1-V8-001	AP1000 Electrical Installation Specification	Revision 12
		APP-GW-C1-001	AP1000 Civil/Structural Design Criteria	Revision 5
		APP-GW-S1-006	AP1000 Design Guide for Raceway Systems	Revision 4
		IEEE Std 628-2001	IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations	R2006
71152S	Corrective Action Documents	CR 10936554	Urgent alarm for M2 bank during withdrawal	1/3/2023
		CR 10937351	Multiple Urgent Alarms while Cycling SD1 (3-PLS-ITPS-601)	1/5/2023
		CR 10937712	Urgent Alarms on Digital Rod Control System (DRCS)	1/7/2023
		CR 10937781	Urgent Alarms on Digital Rod Control System	1/7/2023
		CR 10938992	OPDMS Rod Insertion limits Screen Not Indicating Properly	1/12/2023
		CR 10939092	Request for WEC review of unit 3 DRCS troubleshooting	1/12/2023
		CR 10941552	3-PLS-JD-RDM001 Requires Engineering Troubleshooting Support	1/22/2023
		CR 10945495	Rod Deviation Alarm Unexpected During Rod Withdrawal for 3-PLS-ITPS-605	2/3/2023
		CR 10945554	Most recent trend for Rod Control Urgent Alarms	2/4/2023
		CR 10945714	Temporary Procedure Change for 3-PLS-ITPS-605	2/4/2023
71153	Corrective Action Documents	CAR 404874	U3 Rx Trip and ECS Post Rx Trip Response	
		CAR 411146	Unit 3 Reactor and Turbine Tripped Multiple Times Resulting in Challenges to Startup	05/15/2023
		CR 10956663	While Attempting to Close the Main Generator Field Circuit Breaker in Support of B-ZVS-MEM-001 (Main Generator AVR Startup Test), Received a Rx Trip	
		CR 10956706	Following Rx Trip on 3-15-23, 86 Lockout Was Received on UATs 2A, 2B and 2C, 3-ECS-ES-1,3,5 Fast Transferred to RAT 4B, 3-ECS-ES-2 Residual Bus Transferred to RAT 4A, Standby Diesel B Started, 3-ECS-ES-4 & 6 De-energized	
		CR 10956975	Request Work Order to Implement E&DCR APP-ECSGEF-537 on ECS relays	
		CR 10963375	Reactor Trip	
		CR 10963376	Unit 3 Reactor Trip	
	Engineering	SNC1455668	ES Bus 1-6 Protective Relay Fast Bus Transfer Logic	0

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Changes		Change (OAR for EDCR SV3-ECS-GEF-506)	
	Engineering Evaluations	TE 1125789	WEC to Provide ASR [Advanced Software Release] for Flow Control Valve	
	Miscellaneous		Unit 3 Control Room Logs and Ovation Trend	03/15-16/2023
		3-23-001	Reactor Trip Report	03/17/2023
		LER 05200025/2023-002-00	Automatic Reactor Protection System Actuation During Mode 1 Due to Incorrect Relay Settings Caused by Less Than Adequate Questioning Attitude, Validation of Assumptions, and Interface/Guidance	05/15/2023
		LER 05200025/2023-003-00	Automatic Reactor Protection System Actuation During Startup Testing Due to Incorrect Turbine Control Valve Setting	06/08/2023
	Work Orders	SNC1455296	Implement E&DCR SV3-ECS-GEF-506 and DCP SNC1455668 on ECS Relays	04/18/2023