UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

August 13, 2025

Brad Kapellas, Site Vice President
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION – INTEGRATED INSPECTION REPORT 05000416/2025002

Dear Brad Kapellas:

On June 30, 2025, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Grand Gulf Nuclear Station. On July 14, 2025, the NRC inspectors 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. 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 IV; the Director, Office of Enforcement; and the NRC Resident Inspector at Grand Gulf Nuclear Station.

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 IV; and the NRC Resident Inspector at Grand Gulf Nuclear Station.

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 Dodson, Douglas
on 08/13/25

Douglas E. Dodson II, Chief
Reactor Projects Branch C
Division of Operating Reactor Safety

Docket No. 05000416
License No. NPF-29

Enclosure:
As stated

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05000416/2025002 – DATED AUGUST 13, 2025

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05000416/2025002
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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000416
License Number:	NPF-29
Report Number:	05000416/2025002
Enterprise Identifier:	I-2025-002-0005
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	Port Gibson, MS
Inspection Dates:	April 1, 2025, to June 30, 2025
Inspectors:	N. Brown, Resident Inspector P. Elkmann, Senior Emergency Preparedness Inspector L. Moore, Emergency Preparedness Inspector B. Pannabecker, Resident Inspector A. Smallwood, Senior Resident Inspector
Approved By:	Douglas E. Dodson II, Chief Reactor Projects Branch C Division of Operating Reactor Safety

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Grand Gulf Nuclear Station, 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 Evaluate and Establish Preventive Maintenance for the Main Condenser Gland Seal Steam Controller			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000416/2025002-01 Open/Closed	[P.5] - Operating Experience	71153
A self-revealed finding of very low safety significance (Green) was identified for the licensee's failure to properly evaluate and appropriately classify the gland seal steam controller power supply 1N21K602B, which is critical equipment, and develop a maintenance strategy in accordance with station procedures. As a result, the condenser gland seal steam controller, a power supply with electrolytic capacitors, failed, which resulted in a loss of main condenser vacuum leading to a manual reactor scram on November 10, 2024.			

Failure to Correct Leakage Into the Safety-Related Auxiliary Building in a Timely Manner			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000416/2025002-02 Open/Closed	[P.2] - Evaluation	71153
The inspectors reviewed a self-revealed finding of very low safety significance (Green) and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to correct leakage into the safety-related auxiliary building. As a result, leakage during rainfall led to a partial loss of division II electrical power to some safety-related equipment, and licensed operators inserted a manual reactor scram on February 15, 2025.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000416/2025-001-00	Manual Scram due to Degraded Condenser Vacuum	71153	Closed
LER	05000416/2024-005-00	Loss of Seal Steam Controller Power Resulted in Manual Reactor Shutdown	71153	Closed

PLANT STATUS

At the beginning of the inspection period, Grand Gulf Nuclear Station, Unit 1, was operating at 100 percent rated thermal power (RTP). On the morning of May 12, 2025, the unit reduced power to 79 percent RTP to support electrical grid maintenance and returned to 100 percent RTP later that day. On June 15, 2025, the unit lowered power to 95 percent RTP for approximately 2 hours for planned control rod maintenance and then returned to 100 percent RTP. On June 17, 2025, in response to lowering condenser vacuum, the licensed operators inserted a manual scram, and the unit remained shutdown until reactor startup commenced on June 19, 2025. The unit reached 100 percent RTP on June 20, 2025, briefly lowered power to 81 percent RTP for a rod pattern adjustment on June 22, 2025, and then returned to full RTP where it remained at or near for the remainder of the inspection period.

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

71111.04 - Equipment Alignment

Partial Walkdown Sample (IP Section 03.01) (4 Samples)

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

(1) division II emergency safety feature bus 16BB2 load center on April 2, 2025
(2) division I control rod drive system on April 17, 2025
(3) low pressure core spray on April 24, 2025
(4) division III standby service water, high pressure core spray room cooler piping on June 17, 2025

71111.05 - Fire Protection

Fire Area Walkdown and Inspection Sample (IP Section 03.01) (6 Samples)

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 II control room air conditioning room, fire area OC303 on April 18, 2025

(2) upper cable spreading room, fire area OC702, 189 ft elevation in the control building on April 21, 2025
(3) electrical switchgear rooms, fire zones OC214 and OC215, 111 ft elevation of the control building on April 29, 2025
(4) safety-related cable room, fire area 1A539 on May 8, 2025
(5) division III diesel generator room, fire area 1D306 on June 13, 2025
(6) containment vent equipment room, fire area 1A405 on June 18, 2025

Fire Brigade Drill Performance Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the on-site fire brigade training and performance during an unannounced fire drill associated with an auxiliary building cable tray fire in fire zone 1A417 on June 13, 2025.

71111.06 - Flood Protection Measures

Flooding Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated internal flooding mitigation protections in the auxiliary building 93 ft elevation near room 1A111 for the containment/auxiliary building floor seal on April 3, 2025.

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 reactor startup on June 18, 2025.

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated a licensed operator simulator scenario on April 15, 2025.

71111.12 - Maintenance Effectiveness

Maintenance Effectiveness (IP Section 03.01) (1 Sample)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components (SSCs) remain capable of performing their intended function:

(1) standby service water to plant service water crosstie valve, P41-F189, stem nut removal and replacement (work order (WO) 00555111) on May 23, 2025

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management Sample (IP Section 03.01) (2 Samples)

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) containment vent isolation valve M41-F026 testing and main switchyard breaker work probabilistic risk on April 16, 2025
(2) engineered safety feature transformer 12 maintenance work and risk mitigation system lineup on April 29, 2025

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment (IP Section 03.01) (5 Samples)

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

(1) CR-GGN-2025-02027, division I standby liquid control pump oil leak on May 1, 2025
(2) CR-GGN-2025-01814, main feedwater isolation valve, B21-F065A, steam leakage on May 22, 2025
(3) CR-GGN-2025-02406, division 1 switchgear battery room ventilation fan on May 23, 2025
(4) CR-GGN-2025-02283, division 1 standby diesel generator load shedding and sequencing on June 5, 2025
(5) CR-GGN-2025-02047, valves with magnesium rotors in the auxiliary building steam tunnel on June 9, 2025

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (1 Sample)

The inspectors evaluated the following temporary or permanent modifications:

(1) EC-54269133, feedwater isolation valve, B21-F065A, stuffing box enclosure/clamp steam leak mitigation on May 19, 2025

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) (IP Section 03.01) (5 Samples)

(1) WO 54076950, engineered safety feature transformer 11, post-maintenance testing on March 25, 2025
(2) WO 52880248, residual heat removal valve, E12-F074B, post-maintenance testing on April 14, 2025

(3) WO 54263330, division II control room air conditioner compressor replacement, post-maintenance testing on April 21, 2025
(4) WO 54131838, low pressure core spray valve, E21-F012, stem nut replacement stroke testing on April 23, 2025
(5) WO 52907446, high pressure core spray valve, E22-F001, post-maintenance testing on June 24, 2025

Surveillance Testing (IP Section 03.01) (3 Samples)

(1) WO 54234953, low pressure core spray quarterly functional testing on April 22, 2025
(2) WO 54238535, division III diesel generator monthly testing on May 6, 2025
(3) WO 54012824, division III condensate storage tank level functional testing on June 30, 2025

Inservice Testing (IST) (IP Section 03.01) (1 Sample)

(1) WO 54234953, low pressure core spray pump flow and vibration performance testing on May 14, 2025

Reactor Coolant System Leakage Detection Testing (IP Section 03.01) (1 Sample)

(1) CR-GGN-2025-01948, increase in unidentified reactor coolant system leakage on June 2, 2025

71114.01 - Exercise Evaluation

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

The inspectors reviewed the licensee's design and conduct of their March 18, 2025, Biennial Exercise, and observed licensee performance to determine whether the scenario provided opportunities for the emergency response organization to demonstrate key emergency response capabilities, and whether the licensee's emergency response organization performed in a manner that would have protected the health and safety of the public had the events occurred.

(1) The inspectors evaluated the conduct of the biennial emergency plan exercise. The exercise scenario simulated an Operating Basis Earthquake on March 18, 2025, that caused the lockout of vital electrical buses, a break in the high pressure containment spray discharge line, loss of reactor level leading to emergency depressurization, extensive fuel damage, and a radiological release requiring protective actions for the public. The inspectors observed performance in the simulator control room, the Technical Support Center, the Operations Support Center and in-plant, and in the Emergency Operations Center. The inspectors discussed exercise performance with FEMA Region IV at a post-exercise meeting on March 18, 2025, and FEMA Region VI at a post-exercise meeting on March 25, 2025.

71114.06 - Drill Evaluation

Additional Drill and/or Training Evolution (2 Samples)

The inspectors evaluated:

(1) licensed operator drill and exercise performance during a simulator scenario on May 22, 2025
(2) yellow team emergency preparedness drill on May 28, 2025

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified licensee performance indicator submittals listed below:

IE03: Unplanned Power Changes per 7000 Critical Hours Sample (IP Section 02.02) (1 Sample)

(1) April 1, 2024, through March 31, 2025

IE04: Unplanned Scrams with Complications (USwC) Sample (IP Section 02.03) (1 Sample)

(1) April 1, 2024, through March 31, 2025

MS05: Safety System Functional Failures (SSFFs) Sample (IP Section 02.04) (1 Sample)

(1) April 1, 2024, through March 31, 2025

71152S - Semiannual Trend Problem Identification and Resolution

Semiannual Trend Review (Section 03.02) (1 Sample)

(1) The inspectors reviewed the licensee's corrective action program to identify potential trends in corrective action issues related to degraded doors that might be indicative of a more significant safety issue.

71153 - Follow Up of Events and Notices of Enforcement Discretion

Event Report (IP Section 03.02) (2 Samples)

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

(1) LER 05000416/2024-005-00, "Loss of Seal Steam Controller Power Resulted in Manual Reactor Shutdown" (ML25009A011). The circumstances surrounding this LER and a Green finding are documented in the Inspection Results section of this report. This LER is closed.
(2) LER 05000416/2025-001-00, "Manual Scram Due to Degraded Condenser Vacuum" (ML25106A148). The circumstances surrounding this LER and a Green, non-cited violation are documented in the Inspection Results section of this report. This LER is closed.

INSPECTION RESULTS

Observation: Condition Reports Documenting Issues Related to Plant Doors	71152S

The inspectors reviewed documents entered into the corrective action program for the following:

1. complete, accurate, and timely documentation of the issue identified in the corrective action program
2. evaluation and timely disposition of operability and reportability issues
3. consideration of extent of condition and cause, generic implications, common cause, and previous occurrences
4. classification and prioritization of the problem's resolution commensurate with the safety significance
5. identification of corrective actions that are appropriately focused on correcting the problem
6. completion of corrective actions in a timely manner commensurate with the safety significance of the issue

For the inspection period, the inspectors observed a potential trend associated with plant doors. Specifically, the inspectors noted multiple condition reports (CRs) related to plant doors. Doors in a nuclear power plant provide several safety functions with some doors credited for more than one safety function: physical security and access control; fire prevention; flooding prevention; containment boundaries and radiation area and high radiation area boundaries; and high energy line break protection. The inspectors noted that from January 1 to June 30, 2025, approximately 140 CRs were generated documenting various issues with plant doors. Five of these CRs documented doors that were not properly secured. Security management initiated a rollup CR to trend and correct this issue. The inspectors identified 14 CRs related to door alarms being inoperable or locked in continuous alarm. The inspectors also noted that a total of 37 of these CRs related to degraded door components such as locking mechanisms, latching mechanisms, door hinges, and door frames. Many safety-related doors had multiple issues or repeat issues: the main control room access door, as documented in CR-GGN-2025-00130, CR-GGN-2025-00171, CR-GGN-2025-00258, and CR-GGN-2025-00364; and the door to division 1 standby service water as documented in CR-GGN-2025-01117, CR-GGN-2025-01777, CR-GGN-2025-01853, CR-GGN-2025-01952, CR-GGN-2025-02082, CR-GGN-2025-02096, and CR-GGN-2025-02173. As a result of the inspectors' observations, the licensee entered this observation into their corrective action program as CR-GGN-2025-03128. The inspectors determined that the individual conditions and any performance deficiencies identified were minor because they were not precursors to a significant event, would not have the potential to lead to a more significant safety concern if left uncorrected, and did not adversely affect one of the cornerstone objectives.

Failure to Adequately Evaluate and Establish Preventive Maintenance for the Main Condenser Gland Seal Steam Controller			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000416/2025002-01 Open/Closed	[P.5] - Operating Experience	71153

A self-revealed finding of very low safety significance (Green) was identified for the licensee's failure to properly evaluate and appropriately classify the gland seal steam controller power supply 1N21K602B, which is critical equipment, and develop a maintenance strategy in accordance with station procedures. As a result, the condenser gland seal steam controller, a power supply with electrolytic capacitors, failed, which resulted in a loss of main condenser vacuum leading to a manual reactor scram on November 10, 2024.

Description: On November 10, 2024, at 3:37 a.m., operators inserted a manual scram due to a lowering main condenser vacuum. Using procedure 05-1-02-V-8, "Loss of Condenser Vacuum," Revision 29, operators had already reduced reactor power to 84 percent using recirculation pumps prior to inserting the scram. All reactor protection system functions performed as expected, and all control rods were verified to be fully inserted. Main condenser vacuum continued to degrade resulting in a loss of main feed pumps, closure of the main turbine bypass valves, and closure of the main steam isolation valves (MSIVs). The MSIVs closed at 5:45 a.m. when main condenser vacuum reached 9" of Hg and continued falling. This resulted in a loss of the main turbine condenser for reactor vessel level control, pressure control, and use of the main condenser as a heat sink for decay heat removal. When the main feed pumps became unavailable, the licensee transferred control of reactor vessel level to the reactor core isolation cooling (RCIC) system. Upon closure of the MSIVs, the licensee transitioned to pressure control of the vessel using a safety relief valve. By 10:02 a.m., station operators restored vacuum to the main condenser using mechanical vacuum pumps, recovered feedwater, and reopened the MSIVs, restoring the normal means of reactor vessel level control, pressure control, and decay heat removal. Once the plant was stable, operators proceeded with an orderly reactor cooldown and entered a forced outage period to make repairs to plant systems.

The licensee determined the loss of condenser vacuum was caused by a failed power supply, 1N21K602B, which supplies power to the main condenser gland seal steam controller. Without power to the gland seal steam controller, a pathway existed for air leakage into the main condenser, resulting in the loss of main condenser vacuum.

The licensee's root cause evaluation for this issue, documented in condition report (CR) CR-GGN-2024-06083, states the root cause was, "Failure of aged components due to expected component life being exceeded." The evaluation also notes, "The loss of seal steam electrical power, seal steam flow and loss of condenser vacuum was caused by the age-related internal component failure of the 1N21K602B Seal Steam Power supply." The inspectors noted the root cause evaluation did not explicitly identify the actual age of the power supply. Through additional communication with the licensee and original vendor, the inspectors determined the power supply was 19 years and 2 months old when it failed.

The inspectors noted that power supply 1N21K602B, a critical component, is composed of multiple electrical components including electrolytic capacitors, which begin to degrade on the date of manufacture, not the date of installation. Power supplies with electrolytic capacitors begin to degrade over time primarily due to the evaporation or leakage of the internal

electrolytes, which impacts the ability of the capacitor to maintain its electrical properties. In Electric Power Research Institute's (EPRI) guidance TR12175-1999 for capacitors and TR1003096-2001 for power supplies containing capacitors, EPRI specified that under the best conditions, a 5-to-10-year replacement schedule can be expected. Later, the NRC issued Information Notice 2012–11, "Age-Related Capacitor Degradation." The NRC concluded that, "Operating experience has shown that capacitors have finite lifetimes. Placing these capacitors in a periodic preventive maintenance program that accounts for both time in storage and time in service can address the adverse effects of aging capacitors in equipment circuitry and prevent equipment failures."

The inspectors also noted that the licensee had generated multiple CRs documenting internal operating experience related to degraded electrolytic capacitors in power supplies. As early as 2005, the station became aware of issues related to electrolytic capacitors, as documented in CR-GGN-2005-05373, which includes a report from the Boiling Water Reactors Owner Group (BWROG) that includes 32 recommendations on reducing scrams at boiling water reactors. Recommendation number 28 specifically recommends establishing a program to address electrolytic capacitor failures. Additionally, the licensee generated CR-GGN-2007-00873, CR-GGN-2008-02486, and CR-GGN-2008-05950, which all document issues related to the storage life of power supplies and electronic control boards with electrolytic capacitors. Additionally, CR-GGN-2016-01119, CR-GGN-2018-02854, and CR-GGN-2021-00574 document power supplies that failed in service due to age-related degradation of electrolytic capacitors. These CRs also noted inconsistencies with industry life span recommendations related to power supplies with electrolytic capacitors. Additionally, the licensee conducted an EN-DC-175, "Single Point Vulnerability" analysis of power supply 1N21K602B on December 3, 2014, which documented a risk for a reactor feed pump trip that would require a unit power reduction. The root cause evaluation in CR-GGN-2024-06083 points out that a single point vulnerability was not identified for a loss of condenser vacuum, which the licensee documented as a possible cause. Finally, on September 25, 2021, power supply 1N21K602A (same make and model as the subject power supply, 1N21K602B), failed, resulting in multiple control room alarms and loss of feedwater and condensate system controllers. Following the failure, the licensee completed an equipment failure evaluation. The evaluation did not note that the 1N21K602A power supply was over 17 years old when it failed and that the power supply's electrolytic capacitors were susceptible to age-related degradation. Hence, the inspectors determined that the 2021 failure and the series of other operating experience items were opportunities for the licensee to identify the causes and extent of condition that ultimately affected the 1N21K602B power supply that failed in November 2024.

The inspectors noted procedure EN-DC-153, "Preventive Maintenance Component Classification," Revision 11, Step 5.2 [4], states, "Components that are classified as High Critical, Low Critical, FLEX Portable Equipment, or FLEX Support equipment shall be included in the [preventive maintenance (PM)] program and have a Maintenance Strategy developed in accordance with EN-DC-335," and EN-DC-335, "PM Basis Template," Revision 5, Section 4 [9] states in part, that system engineers, component engineers and subject matter experts are responsible for assisting in review of fleet PM templates for inclusion of industry and fleet operating experience. Additionally, EN-DC-355, Step 5.2.3 [1] states in part, that PM templates can be derived from internal and external operating experience, vendor requirements, lessons learned from equipment failures, and recommendations from EPRI or industry group participation.

Corrective Actions: Licensee corrective actions included replacing the failed main turbine gland seal steam controller power supply, conducting a root cause evaluation, conducting a

common cause evaluation for other vulnerable station power supplies with electrolytic capacitors, and determining if there are other single point vulnerabilities that may cause a plant trip. The licensee reported this event as Licensee Event Report 05000416/2024-005-00 on January 9, 2025. As part of their corrective actions, the licensee has updated the station procedure, 07-S-13-61, “Power Supply / Inverter Conditioning / Capacitor Reforming,” to include the date of manufacture using supply chain documentation for scheduling planned replacement of power supplies with electrolytic capacitors. This step requires the technicians installing a replacement power supply to determine and record the actual date of manufacture of the power supply in the work order documentation.

Corrective Action References: This issue was entered into the licensee’s corrective action program as condition report CR-GGN-2024-06083.

Performance Assessment:

Performance Deficiency: The licensee’s failure to properly evaluate and appropriately classify gland seal steam controller power supply 1N21K602B, which is critical equipment, and have a maintenance strategy developed in accordance with EN-DC-153, “Preventive Maintenance Component Classification,” and EN-DC-335, “PM Basis Template,” was a performance deficiency. Specifically, the failure to identify the power supply as a critical component with a single point vulnerability and consider relevant industry operating experience, vendor recommendations, internal operating experience, and component failure degradation mechanisms when determining the appropriate PM strategy and task frequency for the component, led to the failure to perform adequate preventive maintenance, mitigate, or remove the vulnerability, whose failure resulted in a manual scram and loss of condenser.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance 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. Specifically, the performance deficiency caused the operators to insert a reactor scram.

The inspectors assessed the significance of the finding using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” Using Exhibit 1, “Initiating Events Screening Questions,” and the “Transient Initiators” question, the inspectors determined that 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 (loss of condenser and loss of feedwater). Therefore, the inspectors determined that a detailed risk evaluation was required for this finding.

A senior reactor analyst performed a detailed risk evaluation. The analyst determined that a reactor scram event was a necessary outcome of the condition attributable to the performance deficiency, which involved lowering of condenser vacuum at a particular rate. The analyst also determined that this condition would lead to a Group 1 isolation event (MSIV closure) at the applicable low condenser vacuum setpoint, such as what occurred on this occasion, unless action were to have been taken by operators to mitigate the loss of vacuum condition prior to reaching this setpoint. The analyst reviewed licensee procedure 05-1-02-V-8, “Loss of Condenser Vacuum,” and determined that procedural guidance existed whereby operators could have made use of available plant equipment (specifically, mechanical vacuum pumps) following the scram event to prevent this loss of mitigating equipment (i.e., reactor feedwater pump trip and Group 1 isolation) from occurring. The analyst also determined that this action

also represented a potential means of enabling a recovery of power conversion system (PCS) functions, including the reactor feedwater function and turbine bypass valve availability, if initially lost due to the loss of condenser vacuum condition.

The Grand Gulf SPAR Model version TLU1 (based on version 8.82) dated April 7, 2025, along with SAPHIRE version 8.2.11 were used for this evaluation. The analyst determined that model version 8.82 did not provide credit for the maximized control rod drive injection (CRD) function to provide a sufficient means of inventory control for decay heat removal in the event of a failure of other high-pressure injection sources and a failure to manually depressurize the reactor. The analyst reviewed thermal hydraulic evaluations obtained from the licensee as well as licensee procedural guidance for operator actions to maximize CRD flow in response to events that involve failures of other high-pressure injection sources. The analyst determined that credit for the CRD function should be applied in cases where another high-pressure injection source (specifically, the RCIC system) was initially successful for greater than 5 hours following a full-power transient event. This modeling adjustment was implemented with the above referenced TLU1 model. The analyst also assumed that the use of Diverse and Flexible Coping (FLEX) Strategies for station blackout (SBO) events should be credited. To credit the use of FLEX, the analyst adjusted the basic event FLX-XHE-XE-ELAP (Operators Fail to Declare ELAP When Beneficial) probability to 1.0E-2 for both the nominal and conditional risk cases.

The analyst quantified the SPAR model to determine that the baseline conditional core damage probability (CCDP) associated with a general transient event (TRANS) was 1.65E-7. The analyst modeled the risk impact of the condition attributable to the performance deficiency by adjusting the basic event IE-TRANS (General Plant Transient) to a value of 1.0. The analyst assumed that the basic event MSS-MSV-OC-STEAM (Steam Valves [MSIVs] Fail to Remain Open) would serve as an applicable surrogate for the probability of operators failing to prevent a loss of PCS functions from occurring given a scram event with the applicable rate of loss of condenser vacuum that existed. Based on an assessment of the availability of plant procedures and associated equipment for operators to prevent a loss of PCS functions from occurring during this event, and considering that PCS functions were in fact lost during the response to this event on this occasion, the analyst qualitatively assumed that an estimated conditional probability for a similar loss of PCS functions to occur during an event such as this would be approximately 6E-1 to 7E-1. Finally, the analyst considered that the basic event PCS-XHE-XL-TRANS (Power Conversion System Recovery Fails During Transient) would represent the conditional probability that operators would fail to implement PCS recovery actions given the initial loss of condenser vacuum attributable to this condition. Given the availability of procedures and plant equipment to enable PCS recovery actions, the analyst estimated a conditional probability for the failure of a longer-term recovery of PCS functions of 1E-2.

Using these assumptions (i.e., 7E-1 and 1E-2 for the applicable conditional probabilities discussed above), the analyst re-quantified the TRANS event tree to determine an estimated CCDP of 8.92E-7. Subtracting the corresponding baseline CCDP referenced above, the analyst determined that the applicable increase in CCDP attributable to this finding was approximately 7.27E-7, which corresponds to a finding of very low safety significance (Green). As a sensitivity analysis and a conservative bounding case, the analyst considered a conditional risk case with the conditional probability of an initial loss of PCS functions set to 1.0 and a longer-term PCS recovery failure probability of 1E-1, which resulted in an increase in CCDP of 9.95E-7, which also corresponds to a finding of very low safety significance (Green). The analyst also qualitatively considered that further crediting of the CRD mitigating

function for cases involving initial success of the high pressure core spray (HPCS) function for more than 5 hours, similar to the credit applied for initial success of the RCIC function reflected in the TLU model developed for this evaluation, could be similarly applied to result in still lower actual risk associated with this finding.

The dominant sequence applicable for this analysis involved a loss of the condenser availability along with failures of the HPCS, RCIC, and CRD functions and a failure to manually depressurize the reactor. The analyst also considered the impact of the finding on large early release frequency (LERF) and determined that risk attributable to LERF was not a dominant metric in the significance determination for this finding. Based on the above considerations, the analyst concluded that the significance of this finding is very low safety significance (Green).

Cross-Cutting Aspect: P.5 - Operating Experience: The organization systematically and effectively collects, evaluates, and implements relevant internal and external operating experience in a timely manner. Specifically, the licensee did not effectively implement and institutionalize, through changes to station processes, procedures, equipment, and training programs, operating experience associated with age-related failure of power supplies with electrolytic capacitors, which resulted in a plant scram and loss of condenser.

Enforcement: The inspectors did not identify a violation of regulatory requirements associated with this finding.

Failure to Correct Leakage Into the Safety-Related Auxiliary Building in a Timely Manner			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000416/2025002-02 Open/Closed	[P.2] - Evaluation	71153

The inspectors reviewed a self-revealed finding of very low safety significance (Green) and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to correct leakage into the safety-related auxiliary building. As a result, leakage during rainfall led to a partial loss of division II electrical power to some safety-related equipment, and licensed operators inserted a manual reactor scram on February 15, 2025.

Description: On February 15, 2025, at 10:37 p.m., an overcurrent condition occurred on breaker 152-161, which is a breaker powered from the division II safety-related bus supplying load control centers (LCCs) 16BB2 and 16BB4. This resulted in a loss of power to motor control centers (MCCs) 16B21, 16B41, and 16B42. Multiple control room alarms occurred, and at 10:40 p.m., licensed operators inserted a manual scram in accordance with station procedures due to lowering main condenser vacuum. The partial loss of the division II safety-related bus led to a loss of power to the gland seal steam controller for the main turbine condenser. Other safety-related loads that lost power included drywell instrumentation; containment instrumentation; motor operated valves (MOV) for division II containment isolation; division II standby service water MOVs for the division II drywell purge system; the division II diesel jacket water pump; suppression pool instrumentation isolations; and the division II drywell cooling fans. Upon investigation, the station found that excessive water had intruded into the 16BB2 LCC onto the 4160 volt to 480 volt LCC transformer, creating the over current condition. The licensee determined water leakage into the auxiliary building during severe weather and heavy rainfall and enclosure building roof membrane leakage were found to be the cause of the water intrusion into the division II safety-related bus, LCC 16BB2. The

licensee notified the NRC of the manual reactor scram and verified the plant was in a safe shutdown condition. NRC inspectors walked down the affected areas early February 16, 2025, and confirmed the presence of water in the auxiliary building on elevations 185 ft, 166 ft, and 139 ft, and in switchgear room 1A308 where the LCCs 16BB2 and 16BB4 are located.

The inspectors noted that the auxiliary building of Grand Gulf Nuclear Station (GGNS) is a seismic Class 1 structure as described in the Updated Final Safety Analysis Report (UFSAR) for GGNS, Section 3.8.4.1.1.1. The UFSAR states, in part, that the auxiliary building completely encircles containment and houses safety-related systems such as residual heat removal (RHR), reactor core isolation cooling (RCIC), high pressure core spray (HPCS), low pressure core spray (LPCS), and portions of the control rod drive (CRD) system. The auxiliary building also contains electrical and instrumentation piping rooms, ventilation systems, and electrical equipment including LCCs and motor control centers (MCC).

The inspectors noted that Section 3.8.4.2 of the UFSAR states, in part, that seismic Category I structures are designed in accordance with the regulations, codes, standards, and specifications listed in subsection 3.8.3.2 including Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants.” Regulatory Guide 1.102, which states, in part, that analysis supporting invulnerability of safety-related structures, systems, and components (SSCs) from the effects of local probable maximum precipitation (PMP) should be performed using the point rainfall value of the PMP for the area. As described in the UFSAR, Section 2.4.2.3.3.2.1.2, rainfall intensity, the PMP for GGNS is calculated to be 8.2 inches in 30 minutes. Additionally, Section 2.4.3.2 of the UFSAR states, in part, that the plant roof drainage system is designed based on the 100-year, 1-hour rainfall and was checked to ensure that during PMP, no damage can occur to safety-related plant structures.

The inspectors noted a history of condition reports identifying leaking into the auxiliary building. Specifically, on March 11, 2016, NRC inspectors noted water traveling down the outer containment wall on the 166 ft elevation of the auxiliary building during routine walkdowns. This observation was documented in condition report (CR)-GGN-2016-02253. CR-GGN-2016-02309 documents a follow up inspection of the enclosure building on the 185 ft elevation, which is directly above the water intrusion identified by NRC inspectors. This CR documented a “large amount of water” located below a roof drain indicating leakage in the enclosure building roof. The CR also noted several roof beams and structural supports for the enclosure building showed signs of corrosion from water intrusion. The licensee identified issues with the roof membrane material of the enclosure building and documented CR-GGN-2016-07818. On February 18, 2021, licensee personnel documented rainwater leakage into the auxiliary building on the 139 ft, 166 ft, and 185 ft elevations from rainwater. The licensee placed absorbent pads in many areas of the plant and documented the rainwater in leakage in CR-GGN-2021-01386. CR-GGN-2022-09501 documents issues with the paver stones used to hold the roof membrane material of the enclosure building in place. Each of the CRs regarding the enclosure building and auxiliary building roof were screened as non-adverse.

The inspectors noted during their walkdowns that the rainwater intrusion into the auxiliary building on February 15, 2025, was consistent with the paths and locations noted in both CR-GGN-2016-02253 and CR-GGN-2021-01386. Additionally, the inspectors noted station procedure EN-LI-102, “Corrective Action Program,” Revision 54, states that adverse conditions are required to be corrected in the corrective action program and are subject to the rigor necessary to evaluate and thoroughly resolve important and significant issues. The

inspectors determined that water intrusion into the auxiliary building via rain, whether it lands on safety-related equipment or not, is a condition adverse to quality and is required to be tracked and corrected in accordance with the licensee's corrective action program.

Corrective Actions: The licensee initiated CR-GGN-2025-00785 and conducted an apparent cause analysis. Additionally, the licensee reported this event as LER 05000416/2025-001-00 on April 16, 2025. The licensee had roof contractors respond to the site, adjust the roof membrane to align with the roof drain scuppers and sealed tears identified during roof inspections. The licensee has also increased the periodicity of their roof inspections.

Corrective Action References: CR-GGN-2025-00785

Performance Assessment:

Performance Deficiency: The licensee failed to adequately and timely address or correct leakage into the safety-related auxiliary building in accordance with station procedure EN-LI-102, "Corrective Action Program," Revision 54, which requires the site to identify conditions adverse to quality and track them in the corrective action program to correct them, and Title 10 CFR Part 50, Appendix B, Criterion XVI, which states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Failure to take appropriate and timely action was within the licensee's ability to foresee and correct and is 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 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. Specifically, the performance deficiency led to partial loss of the division II safety-related bus resulting in a loss of power to the main turbine gland seal steam controller, which resulted in licensed operators inserting a manual scram on February 15, 2025.

Significance: The inspectors assessed the significance of the finding using IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using Exhibit 1, "Initiating Events Screening Questions," and the "Support System Initiators" questions, the inspectors determined that this event resulted in an actual partial loss of a support system (division II 16BB2 and 16BB4 LCCs and their respective loads—safety-related alternating current power). Therefore, a detailed risk evaluation was required.

A senior reactor analyst performed a detailed risk evaluation. The analyst determined that a reactor scram event was a necessary outcome of the condition attributable to the performance deficiency, which involved lowering of condenser vacuum. The Grand Gulf SPAR Model version TLU1 (based on version 8.82) dated April 7, 2025, along with SAPHIRE version 8.2.12 were used for this evaluation. The analyst determined that model version 8.82 did not provide credit for the maximized CRD function to provide a sufficient means of inventory control for decay heat removal in the event of a failure of other high-pressure injection sources and a failure to manually depressurize the reactor. The analyst reviewed thermal hydraulic evaluations obtained from the licensee as well as licensee procedural guidance for operator actions to maximize CRD flow in response to events that involve failures of other high pressure injection sources. The analyst determined that credit for the CRD function should be applied in cases where another high-pressure injection source (specifically, the RCIC system) was initially successful for greater than 5 hours following a full-power transient event. This

modeling adjustment was implemented with the above referenced TLU1 model. The analyst also assumed that the use of Diverse and Flexible Coping (FLEX) Strategies for station blackout (SBO) events should be credited. To credit the use of FLEX, the analyst adjusted the basic event FLX-XHE-XE-ELAP (Operators Fail to Declare ELAP When Beneficial) probability to 1.0E-2 for both the nominal and conditional risk cases.

The analyst quantified the SPAR model to determine that the baseline core damage frequency (CDF) associated with a general transient event (TRANS) was 1.65E-7. The analyst modeled the risk impact of the condition attributable to the performance deficiency by adjusting the basic event IE-TRANS (General Plant Transient) to a value of 1.0. The analyst reviewed the components powered from the load centers and MCCs affected by the condition resulting from the performance deficiency. The following basic events were set to TRUE to represent the impact of the loss of power to the affected components:

a. SSW-MOV-CC-F054 (SSW Supply Valve F054 to CCW HTXs Fails to Open)
b. SSW-MOV-CC-F067 (CCW HTX Discharge Isolation MOV F067 to SSW Fails to Open)
c. SPM-MOV-CC-F001B (Drain Valve F001B Fails to Open)
d. SPM-MOV-CC-F002B (Drain Valve F002B Fails to Open)

The analyst re-quantified the TRANS event tree with these modeling adjustments to determine an estimated conditional CDF of 3.05E-7. Subtracting the corresponding baseline CDF referenced above, the analyst determined that the applicable delta-CDF attributable to this finding was approximately 1.40E-7, which corresponds to a finding of very low safety significance (Green). The dominant sequence applicable for this analysis involved failures of either the main condenser or the main feedwater function along with failure of the HPCS, RCIC, and CRD functions and a failure to manually depressurize the reactor. The analyst also considered the impact of the finding on large early release frequency (LERF) and determined that risk attributable to LERF was not a dominant metric in the significance determination for this finding. Based on the above considerations, the analyst concluded that the significance of this finding is very low safety significance (Green).

Cross-Cutting Aspect: P.2 - Evaluation: The organization thoroughly evaluates issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Specifically, the licensee failed to properly classify, prioritize, and evaluate roof and building leakage issues according to their significance, which resulted in the licensee's failure to take adequate and timely corrective actions.

Enforcement:

Violation: Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected.

Contrary to the above, from March 11, 2016, to June 30, 2025, measures were not established to assure that conditions adverse to quality were promptly identified and corrected. Specifically, the licensee failed to correct leakage into the safety-related auxiliary building, a condition adverse to quality, which resulted in loss of safety-related equipment and a plant scram.

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

EXIT MEETINGS AND DEBRIEFS

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

- On May 1, 2025, the inspectors presented the exit meeting for the exercise evaluation inspection (IP 71114.01) inspection results to Jason Richardson, General Manager of Plant Operations, and other members of the licensee staff.
- On July 14, 2025, the inspectors presented the integrated inspection results to Brad Kapellas, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.04	Corrective Action Documents	CR-GGN-YYYY-NNNN	2025-01605, 2024-02052, 2024-03308	
71111.04	Procedures	04-1-01-C11-1	Control Rod Drive Hydraulic System	162
71111.04	Procedures	04-1-01-E21-1	Low Pressure Core Spray System	46
71111.04	Procedures	04-1-01-P41-1	Standby Service Water System	156
71111.05	Corrective Action Documents	CR-GGN-YYYY-NNNN	2025-01668, 2025-02176, 2025-01872, 2025-01958, 2025-2150, 2025-02178, 2025-02843	
71111.05	Fire Plans	A-30	CONTAINMENT VENT EQUIP ROOM - 1A405	3
71111.05	Fire Plans	A-46	CABLE SPACE ROOM 1A539	2
71111.05	Fire Plans	C-08	ESF HVAC EQUIPMENT - OC303	5
71111.05	Fire Plans	C-16	UPPER CABLE SPREADING ROOM (U-1) OC702	3
71111.05	Fire Plans	DG-04	HPCS (DIV III) DIESEL GENERATOR ROOM 1D306	8
71111.05	Procedures	EN-OP-125	Fire Brigade Drills	000
71111.06	Procedures	05-1-02-VI-1	Off-Normal Event Procedure Flooding	120
71111.11Q	Procedures	03-1-01-1	Cold Shutdown to Generator Carrying Minimum Load	205
71111.11Q	Procedures	05-S-01-EP-2M1-3	Emergency Procedure RPV Control Modes 1-3	0
71111.12	Work Orders		00555111	
71111.13	Corrective Action Documents	CR-GGN-YYYY-NNNN	2025-01120, 2025-01798, 2025-01811	
71111.15	Corrective Action Documents	CR-GGN-YYYY-NNNN	2025-01814, 2025-01755, 2025-02027, 2025-02283, 2025-02047, 2025-02406	
71111.15	Procedures	04-1-01-C41-1	Standby Liquid Control System	128
71111.15	Work Orders		54266820	
71111.18	Engineering Changes	EC-54269133	B21-F065A Stuff Box Enclosure / Clamp Steam Leak Mitigation	
71111.24	Corrective Action Documents	CR-GGN-YYYY-NNNN	2025-02047, 2025-02283, 2025-01221, 2025-01948	
71111.24	Work Orders		52880248, 54263330, 54076950, 54238535, 54234953, 54131838, 52583441, 00187044, 52907446, 52907445, 54012824,	
71114.01	Corrective Action	CR-GGN-YYYY-	2025-01364, 2025-01382, 2025-01588, 2025-01589,	

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Documents	NNNN	2025-01590, 2025-01591, 2025-01592, 2025-01593, 2025-01594, 01595	
71114.01	Miscellaneous		Dress Rehearsal Exercise Evaluation Report	02/22/2023
71114.01	Miscellaneous	GIN 2023-000059	EOF Mini Drill Critique	
71114.01	Miscellaneous	GIN 2023-000115	Blue Team Exercise Critique	07/26/2023
71114.01	Miscellaneous	GIN 2023-00096	First Half Semi-Annual Health Physics Drill Report	
71114.01	Miscellaneous	GIN 2023-00140	Red Team Exercise Critique	09/13/2023
71114.01	Miscellaneous	GIN 2023-00140	Red Team Exercise Critique	09/13/2023
71114.01	Miscellaneous	GIN 2023-00155	Green Team Exercise Critique	10/25/2023
71114.01	Miscellaneous	GIN 2024-00010	2023 GGNS Onsite Accountability Drill	2023
71114.01	Miscellaneous	GIN 2024-00014	Second Half 2023 Semi-Annual Health Physics Drill	2023
71114.01	Miscellaneous	GIN 2024-00128	Yellow-Red Team Turnover Drill	08/28/2024
71114.01	Miscellaneous	GIN 2025-00005	Onsite Medical Drill with CC Medical Center and ASAP Ambulance	12/03/2024
71114.01	Miscellaneous	GIN 2025-00009	2024 Radiological Monitoring Drill	12/20/2024
71114.01	Miscellaneous	GIN 2025-00010	First Half Semi-Annual Health Physics Drills	06/06/2024
71114.01	Miscellaneous	GIN 2025-00010	Second Half 2024 Semi-Annual Health Physics Drill	12/18/2024
71114.01	Miscellaneous	GIN 2025-0004	12Dec2024 2024 Onsite Accountability Drill	12/12/2024
71114.01	Procedures		Grand Gulf Nuclear Station, Emergency Action Level Technical Bases,	3
71114.01	Procedures	05-S-02-VI-3	Off Normal Event Procedure, Earthquake	117
71114.01	Procedures	10-S-01-1	Activation of the Emergency Plan, 10/27/2020	133
71114.01	Procedures	10-S-01-12	Radiological Assessment and Protective Action Recommendations, 11/28/2022	50
71114.01	Procedures	10-S-01-14	Emergency Radiation Monitoring, 05/19/2019	26
71114.01	Procedures	EN-EP-306	Drills and Exercises, 02/10/2022	11
71114.01	Procedures	EN-EP-313	Offsite Dose Assessment using the Unified RASCAL Interface, 04/22/2020	4
71114.01	Procedures	EN-EP-603	Emergency Notifications, 12/15/2021	1
71114.01	Procedures	EN-EP-609	Emergency Operations Facility Operations, 05/01/2023	7
71114.01	Procedures	EN-EP-610	Technical Support Center Operations, 05/01/2023	8
71114.01	Procedures	EN-EP-611	Operations Support Center Operations, 05/01/2023	8
71114.01	Procedures	EPP 01-02	Grand Gulf Nuclear Station, Emergency Action Levels Wall	A

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			Chart	
71114.06	Corrective Action Documents	CR-GGN-YYYY-NNNN	2025-02315, 2025-02365, 2025-01803, 2025-02371, 2025-02443, 2025-02535	
71152S	Corrective Action Documents	CR-GGN-YYYY-NNNN	2025-00130, 2025-00133, 2025-00171, 2025-00258, 2025-00299, 2025-00308, 2025-00320, 2025-00336, 2025-00337, 2025-00352, 2025-00364, 2025-00371, 2025-00380, 2025-00427, 2025-00552, 2025-00638, 2025-00648, 2025-00742, 2025-00815, 2025-00889, 2025-00890, 2025-00898, 2025-00955, 2025-00967, 2025-00971, 2025-00972, 2025-01059, 2025-01095, 2025-01276, 2025-01353, 2025-01378, 2025-01401, 2025-01476, 2025-01478, 2025-01520, 2025-01572, 2025-01613, 2025-01653, 2025-01655, 2025-01656, 2025-01726, 2025-01727, 2025-01742, 2025-01743, 2025-01762, 2025-01773, 2025-01777, 2025-01815, 2025-01853, 2025-01863, 2025-01914, 2025-01926, 2025-01979, 2025-02015, 2025-02017, 2025-02060, 2025-02067, 2025-02082, 2025-02096, 2025-02150, 2025-02187, 2025-02218, 2025-02357, 2025-02359, 2025-02395, 2025-02402, 2025-02423, 2025-02465, 2025-02473, 2025-02481, 2025-02538, 2025-02557, 2025-02612, 2025-02650, 2025-02694, 2025-02789, 2025-02808, 2025-02818, 2025-02872	
71152S	Procedures	EN-OE-100	Operating Experience Program	37
71153	Corrective Action Documents	CR-GGN-YYYY-NNNN	2005-05373, 2007-00873, 2008-02468, 2008-05950, 2015-06106, 2016-01119, 2018-02854, 2018-03716, 2020-11792, 2021-00574, 2021-05060, 2022-00960, 2022-08870, 2024-06083, 2016-02253, 2016-02309, 2016-03522, 2016-07818, 2017-00814, 2017-06597, 2021-01386, 2022-09501, 2025-00785	
71153	Procedures	05-1-02-I-1	Off-Normal Event Procedure Reactor SCRAM	139
71153	Procedures	05-1-02-V-8	Off-Normal Event Procedure Loss of Condenser Vacuum	29
71153	Procedures	EN-DC-150	Condition Monitoring of Maintenance Rule Structures	18
71153	Procedures	EN-DC-153	Preventive Maintenance Component Classification	11

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71153	Procedures	EN-DC-335	PM Basis Template	5
71153	Procedures	EN-FAP-MA-010	Conduct of Preventive Roof Inspections	2
71153	Procedures	EN-LI-102	Corrective Action Program	54
71153	Procedures	EN-OP-120	Operator Fundamentals	3
71153	Work Orders		00381286, 00440438, 52704346	