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

April 20, 2026

EAF-RIV-2026-0015

Charles Kharrl, President,
CEO and CNO
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION, UNITS 1 AND 2 – NOTICE OF VIOLATION; NRC INSPECTION REPORT 05000498/2026090 AND 05000499/2026090

Dear Charles Kharrl,

This letter refers to the inspection completed on March 31, 2026, at South Texas Project Electric Generating Station (STP). The purpose of the inspection was to review the third-party causal evaluation for probable causes of the July 24, 2024, loss of offsite power event and subsequent switchyard performance to identify whether performance deficiencies or violations existed, as documented under unresolved item (URI) 05000498, 05000499/2024050-07, "Switchyard Fire Cause and Subsequent Switchyard Component Performance," in U.S. Nuclear Regulatory Commission (NRC) Inspection Report 05000498/2024050 and 05000499/2024050, dated November 21, 2024, Agencywide Documents Access and Management System (ADAMS) Accession No. ML24320A137. The enclosed inspection report (Enclosure 2) presents the results of this inspection. On March 31, 2026, the inspectors conducted a telephonic exit meeting with Robert Norris, Site Vice President, and other members of your staff.

Based on the results of this inspection, one apparent violation was identified and was considered for escalated enforcement action in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's website at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. The apparent violation involved the failure to obtain a license amendment pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, as required by 10 CFR 50.59 (c)(2), prior to implementing a proposed change that resulted in a more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated).

During the exit meeting, Robert Norris of your staff informed Patricia Vossmar, Director (Acting), Division of Operating Reactor Safety, that STP accepted the violation as described in the exit meeting, and the categorization of the violation as Severity Level III. In addition, STP declined the opportunity to discuss this issue at a predecisional enforcement conference or to provide a written response. After considering the information developed during the inspection, the NRC concluded that a violation exists and is appropriately categorized as Severity Level III.

The violation is cited in Enclosure 1, Notice of Violation (Notice), and the circumstances surrounding the violation are described in the enclosed inspection report (Enclosure 2).

The NRC considered whether credit was warranted for *Identification* and *Corrective Action*, in accordance with the civil penalty assessment process in Section 2.3.4 of the NRC Enforcement Policy. Because you exerted noteworthy effort in identifying the underlying cause and extent of conditions, the NRC concluded that credit is warranted for *Identification*. The NRC also concluded that *Corrective Action* credit is warranted for your prompt and comprehensive corrective actions. Your corrective actions to address the violation are documented in the enclosed inspection report.

Therefore, to encourage prompt identification and comprehensive correction of violations, I have been authorized not to propose a civil penalty in this case. However, significant violations in the future could result in a civil penalty. In addition, issuance of this Severity Level III violation constitutes escalated enforcement action that may subject you to increased inspection effort, specifically implementation of non-routine inspection follow-up on the Notice of Violation 05000498,05000499/2026090-01, “Failure to Obtain a License Amendment Prior to Implementing a Change to the Offsite Power System,” and any related issues using Inspection Procedure 92702, “Follow-Up on Traditional Enforcement Actions Including Traditional Enforcement Violations, Deviations, Confirmatory Action Letters, and Orders.”

The NRC has concluded that information regarding: (1) the reason for the violation; (2) the corrective actions that have been taken and the results achieved; and (3) the actions remaining to achieve full compliance are already adequately addressed on the docket in the enclosed inspection report. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

In accordance with 10 CFR 2.390 of the NRC’s “Agency Rules of Practice and Procedure,” a copy of this letter, the enclosures, and your response, if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or in the NRC’s ADAMS, accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary information so that it can be made available to the public without redaction. The NRC also includes significant enforcement actions on its website at http://www.nrc.gov/reading-rm/doc-collections/enforcement/actions.

If you have any questions concerning this matter, please contact Douglas E. Dodson II, of my staff at 817-200-1148.

Sincerely,

Signed by Monninger, John
on 04/20/26

John D. Monninger
Regional Administrator

Docket Nos. 05000498 and 05000499
License Nos. NPF-76 and NPF-80

Enclosures:
1. Notice of Violation
2. NRC Inspection Report 05000498/2026090 and 05000499/2026090
   w/ Attachment: Significance Determination Process Evaluation

SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION, UNITS 1 AND 2 – NOTICE OF VIOLATION; NRC INSPECTION REPORT 05000498/2026090 AND 05000499/2026090 – DATED APRIL 20, 2026

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ADAMS Accession Nos. Package: ML26098A142, Inspection Report: ML26098A144, Econcurrence Case: 20260408-80031

NOTICE OF VIOLATION

STP Nuclear Operating Company
South Texas Project, Units 1 and 2

Docket Nos. 05000498 and 05000499
License Nos. NPF-76 and NPF-80
EAF-RIV-2026-0015

During an NRC inspection completed on March 31, 2026, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 50.59, “Changes, Tests, and Experiments,” Section (c)(2)(ii) requires, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change if the change would result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated).
>
> 10 CFR 50.59(d)(1) requires, in part, that a licensee shall maintain records of changes in the facility made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change does not require a license amendment pursuant to paragraph (c)(2) of this section.
>
> Contrary to the above, on March 7, 2014, the licensee failed to obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the Updated Final Safety Analysis Report (UFSAR). Specifically, the licensee allowed a change to the offsite power system that included relocation of the shunt reactors and failed to conduct a written evaluation or provide a basis for the determination that the change did not require a license amendment in accordance with 10 CFR 50.59 (c)(2). The relocation of shunt reactor 2 increased the likelihood that a smoke plume of ionized gases from a fire would adversely affect adjacent overhead line conductors of the offsite power system, which is an SSC described in the UFSAR, and therefore increased the likelihood of a loss of offsite power (LOOP). As a result, on July 24, 2024, a shunt reactor 2 fire occurred, and the overhead C-phase conductor associated with the #1 standby transformer and the A-phase conductor of the Unit 1 main transformer were engulfed by the smoke plume. This condition resulted in phase-to-ground faults, which tripped the 345 kV north bus and caused the Unit 1 LOOP and reactor trip on July 24, 2024.

This is a Severity Level III violation (NRC Enforcement Policy, Section 6.1.c.6).

The NRC has concluded that information regarding: (1) the reason for the violation; (2) the corrective actions that have been taken and the results achieved; and (3) the actions remaining to achieve full compliance are already adequately addressed on the docket in NRC inspection report 05000498/2026090 and 05000499/2026090.

However, if the description therein does not accurately reflect your position or your corrective actions, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 within 30 days of the date of the letter transmitting this Notice of Violation. In that case or if you choose to respond, clearly mark your response as a “Reply to a Notice of Violation; EAF-RIV-2026-0015,” and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, U.S. Nuclear

Regulatory Commission, Region IV, 1600 East Lamar Blvd., Arlington, Texas 76011-4511, and a copy to the NRC Resident Inspector at the South Texas Project Electric Generating Station and email it to R4Enforcement@nrc.gov.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or in the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html. Therefore, to the extent possible, the response should not include any personal privacy or proprietary information so that it can be made available to the public without redaction.

If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request that such material is withheld from public disclosure, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information).

Dated this 20th day of April 2026

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000498 and 05000499
License Numbers:	NPF-76 and NPF-80
Report Numbers:	05000498/2026090 and 05000499/2026090
Enterprise Identifier:	I-2026-090-0000
Licensee:	STP Nuclear Operating Company
Facility:	South Texas Project Electric Generating Station, Units 1 and 2
Location:	Wadsworth, TX
Inspection Dates:	August 6, 2025, to March 31, 2026
Inspectors:	L. Flores, Resident Inspector S. Graves, Senior Reactor Inspector R. Kumana, Senior Project Engineer J. Vera, Senior Resident Inspector
Approved By:	Douglas E. Dodson II, Chief Reactor Projects Branch C Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an NRC inspection at South Texas Project Electric Generating Station, Units 1 and 2, 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 Obtain a License Amendment Prior to Implementing a Change to the Offsite Power System			
Cornerstone	Severity	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level III NOV 05000498,05000499/2026090-01 Open EAF-RIV-2026-0015	Not Applicable	71153
The NRC noted a self-revealed violation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, for the licensee's failure to obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the Updated Safety Analysis Report (UFSAR). Specifically, the licensee allowed a change to the offsite power system that included relocation of the shunt reactors and failed to conduct a written evaluation or provide a basis for the determination that the change did not require a license amendment in accordance with 10 CFR 50.59 (c)(2).			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000498,05000499/2024-004-01	Loss of Offsite Power Resulting in Unit 1 Automatic Reactor Trip and Actuation of Emergency Diesel Generators and Auxiliary Feedwater Pumps	71153	Closed
LER	05000498,05000499/2024-004-00	Loss of Offsite Power Resulting in Unit 1 Automatic Reactor Trip and Actuation of Emergency Diesel Generators and Auxiliary Feedwater Pumps	71153	Closed
URI	05000498,05000499/2024050-07	Switchyard Fire Cause and Subsequent Switchyard Component Performance	71153	Closed

INSPECTION SCOPES

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

OTHER ACTIVITIES – BASELINE

71153 - Follow-Up of Events and Notices of Enforcement Discretion

Event Report (IP Section 03.02) (1 Sample)

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

(1) LER 05000498,05000499/2024-004-00, "Loss of Offsite Power Resulting in Unit 1 Automatic Reactor Trip and Actuation of Emergency Diesel Generators and Auxiliary Feedwater Pumps" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24263A145) and LER 05000498,05000499/2024-004-01, "Loss of Offsite Power Resulting in Unit 1 Automatic Reactor Trip and Actuation of Emergency Diesel Generators and Auxiliary Feedwater Pumps," (ML25254A061). The inspection conclusions associated with these LERs are documented in this report under Inspection Results Section 71153. These LERs are Closed. The related unresolved item (URI) 05000498, 05000499/2024050-07, "Switchyard Fire Cause and Subsequent Switchyard Component Performance)," is also closed and documented in this report under Inspection Results Section 71153.

INSPECTION RESULTS

Failure to Obtain a License Amendment Prior to Implementing a Change to the Offsite Power System			
Cornerstone	Severity	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level III NOV 05000498,05000499/2026090-01 Open EAF-RIV-2026-0015	Not Applicable	71153
The NRC noted a self-revealed violation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, for the licensee's failure to obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. Specifically, the licensee allowed a change to the offsite power system that included relocation of the shunt reactors and failed to conduct a written evaluation			

or provide a basis for the determination that the change did not require a license amendment in accordance with 10 CFR 50.59 (c)(2).
Description: The original design of the South Texas Project (STP) switchyard contained two shunt reactors, one connected to the north bus (shunt reactor 1 (RT1)), and one connected to the south bus (shunt reactor 2 (RT2)), with each connected to their respective bus by a 1,600-ampere circuit switcher. Shunt reactors are reactive power compensators that are used in high-voltage power systems to control voltage levels. These devices generate inductive reactance, absorbing reactive power and reducing voltage increases caused by the charging current of long transmission lines. Shunt reactor 2 was a 362 kV, 150 Megavolt-Amperes Reactive (MVAR) 3-phase core form mineral-oil-filled shunt reactor manufactured by the General Electric Company in 1985. Shunt reactor 2, which contained approximately 15,700 gallons of mineral oil, did not have automatic fire detection or fire suppression installed. In the original switchyard configuration, RT2 was located between the independent offsite power circuits for Unit 1. CenterPoint Entergy (CNP), the switchyard operator, developed a proposed design change in 2013 to relocate the shunt reactors and replace the existing circuit switchers with circuit breakers because of obsolescence concerns. The relocation placed RT2 in closer proximity to the towers supporting the high-voltage conductors for both independent transmission circuits for Unit 1 (i.e., Unit 1 main generator output and No. 1 standby transformer). Between June and October 2014, CenterPoint Energy completed the design change and placed the modified shunt reactors in service. On July 24, 2024, at 7:02 a.m., RT2, now located closer to the high-voltage conductor towers in the switchyard, failed due to a phase-to-ground fault and caught fire. As a result, STP Unit 1 experienced a LOOP and reactor trip, and STP Unit 2 experienced a partial loss of offsite power and an unplanned power reduction to approximately 90 percent power at the request of the grid operator. The smoke plume from the large fire reached the nearby overhead C-phase conductor associated with the No. 1 standby transformer and ionized gases in the plume allowed a phase-to-ground fault, which tripped the 345 kV north bus. Approximately 16 seconds later, the north bus was automatically energized when the associated feeder breakers reclosed. The smoke plume subsequently allowed a phase-to-ground fault affecting the Unit 1 main generator transformer A-phase conductor. This led to the LOOP on Unit 1 and the automatic reactor trip. In September of 2024, the NRC conducted a special inspection at STP to identify the circumstances surrounding the July 24 event. This inspection was documented in NRC inspection report 05000498/2024050 and 05000499/2024050 (ML24320A137). This report opened unresolved item (URI) 05000498, 05000499/2024050-07, “Switchyard Fire Cause and Subsequent Switchyard Component Performance.” The URI was held open pending NRC review of the licensee’s evaluation of the causes of the event. Due to the significance of the event, the licensee performed a root cause evaluation to determine the causes. The licensee’s cause evaluation identified a failure to meet the requirements of 10 CFR 50.59 as part of the overall root cause, which stated: CenterPoint Energy had not established and implemented adequate management controls governing design changes to equipment in the Switchyard to ensure that changes were consistent with design requirements and aligned with requirements and recommendations of applicable Industry Standards; and subsequently, STPNOC did

not rigorously implement management controls during the review and approval of design changes developed by CNP.

In particular, in March 2014, licensee personnel completed a 10 CFR 50.59 screening in accordance with licensee Procedure 0PGP05-ZA-0002, “10CFR50.59 Evaluations.” At the time, the licensee concluded that a 10 CFR 50.59 evaluation was not required for the design change. However, the screening did not include the design aspect of moving the location of the shunt reactors, only the replacement of circuit switchers with circuit breakers. The licensee’s cause evaluation further concluded that the location change increased the potential for a single failure of RT2 to potentially impact both offsite power circuits, increasing the likelihood of LOOP. Therefore, a 10 CFR 50.59 evaluation would have been required. Finally, the cause evaluation stated that if the licensee had performed a full 10 CFR 50.59 evaluation, the station likely would have concluded that the design change could not be installed as proposed without prior NRC approval.

The inspectors reviewed the licensee’s cause evaluation and performed an independent review against the requirements of 10 CFR 50.59. The inspectors determined that the offsite power system and shunt reactors were a part of the facility described by the UFSAR. UFSAR Section 8.1.4.1 describes the offsite power design bases, and UFSAR Section 8.2 describes the shunt reactors and their function as part of the 345kV offsite power distribution system. The location of the shunt reactors is shown in UFSAR figure 8.2-1, “Power Supply Routing to Plant.”

The inspectors reviewed the change to determine whether it required an evaluation under 10 CFR 50.59 (d). The inspectors reviewed licensee Procedure 0PGP05-ZA-0002 and NRC-endorsed industry guidance for screening changes out of 10 CFR 50.59. The inspectors determined that there was an adverse effect on the offsite power system’s design function to provide reliable sources of offsite power for supplying plant auxiliary power systems for startup, shutdown, or any time power is unavailable from the unit’s main generator, and therefore, the change required an evaluation under 10 CFR 50.59.

Furthermore, the inspectors determined that the change met the criteria in 10 CFR 50.59 (c)(2)(ii). Specifically, the relocation of the shunt reactors resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of the offsite power system. The offsite power system is a system important to safety previously evaluated in the UFSAR. The relocation of the RT2 in 2014 placed it in close proximity to both offsite transmission circuits for Unit 1. This new location of RT2 resulted in its failure being able to cause a simultaneous loss of the #1 standby transformer and the Unit 1 main generator transformer, resulting in a LOOP to Unit 1 due to loss of both independent offsite transmission circuits.

Therefore, the inspectors determined that: (1) the change was a change to the facility as described in the UFSAR and 10 CFR 50.59 was applicable, (2) the relocation of the shunt reactors adversely affected the design function as described in the UFSAR, and as such, required a complete 10 CFR 50.59 evaluation, and (3) prior Commission approval was required prior to implementing the change, in accordance with 10 CFR 50.59 (c)(2)(ii).

However, the licensee failed to obtain Commission approval prior to implementing the change, and as a result, this change led to the LOOP event on July 24, 2024.

Corrective Actions: The licensee entered the condition into its corrective action program (CR 24-7216 and 26-2967) and initiated actions to address the root cause and contributing causes outlined in the root cause evaluation report. Immediate corrective actions included repairs to 345 kV transmission lines and affected transmission towers, as well as removal of RT2 from service to allow Unit 1 to resume operation. The licensee plans to relocate RT2 to a new location and utilize a different design. The licensee also completed revisions to procedures, developed and implemented written instructions on controls to govern transmission and distribution service provider changes, and completed and planned other actions. The licensee's corrective actions are expected to ensure the design of the plant matches the design and licensing bases and the UFSAR, and any changes to the plant, procedures, etc., will need to be evaluated in accordance with applicable processes, procedures, and regulations.

Corrective Action References: CR 24-7216 and CR 26-2967

Performance Assessment: The NRC determined this violation was associated with a performance deficiency. However, on November 19, 2025, Significance and Enforcement Review Panel (SERP) voting members were convened to determine whether to pursue only the traditional enforcement aspect, only the Reactor Oversight Process aspect, or both aspects in accordance with Inspection Manual Chapter 0612, Appendix B, "Issue Screening Directions." The NRC SERP voting members determined it was most appropriate to only pursue the traditional enforcement aspects of the violation in this case.

Enforcement: The ROP's significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to address this violation which impedes the NRC's ability to regulate using traditional enforcement to adequately deter non-compliance.

Severity: The NRC has determined the appropriate severity level to be Severity Level III for this violation. The relevant example from NRC Enforcement Policy, Section 6.1, "Reactor Operations," is example c.6, which states, "A licensee violates 10 CFR 50.59 by failing to obtain a license amendment for a change that has a consequence evaluated as having White, Yellow, or Red safety significance under the [Significance Determination Process (SDP)]." To determine the severity level, a regional Senior Reactor Analyst performed an evaluation in accordance with the NRC's SDP. The change had a consequence evaluated as having low safety significance (i.e. White). The SDP evaluation is attached.

Violation: Title 10 CFR 50.59, "Changes, Tests, and Experiments," Section (c)(2)(ii) requires, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change if the change would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated).

Title 10 CFR 50.59(d)(1) requires, in part, that a licensee shall maintain records of changes in the facility made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change does not require a license amendment pursuant to paragraph (c)(2) of this section.

Contrary to the above, on March 7, 2014, the licensee failed to obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety

previously evaluated in the UFSAR. Specifically, the licensee allowed a change to the offsite power system that included relocation of the shunt reactors and failed to conduct a written evaluation or provide a basis for the determination that the change did not require a license amendment in accordance with 10 CFR 50.59 (c)(2). The relocation of shunt reactor 2 increased the likelihood that a smoke plume of ionized gases from a fire would adversely affect adjacent overhead line conductors of the offsite power system, which is an SSC described in the UFSAR, and therefore increased the likelihood of a LOOP. As a result, on July 24, 2024, a shunt reactor 2 fire occurred, and the overhead C-phase conductor associated with the #1 standby transformer and the A-phase conductor of the Unit 1 main transformer were engulfed by the smoke plume. This condition resulted in phase-to-ground faults, which tripped the 345 kV north bus and caused the Unit 1 LOOP and reactor trip on July 24, 2024.

Enforcement Action: This is a Severity Level III violation, consistent with Section 6.1.c.6 of the Enforcement Policy.

This finding closes URI 05000498, 05000499/2024050-07.

EXIT MEETINGS AND DEBRIEFS

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

- On March 31, 2026, the inspectors presented the NRC inspection results to Robert Norris, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71153	Miscellaneous	CR 24-7216 Root Cause Evaluation	Loss of Offsite Power Resulting in a Unit 1 Automatic Reactor Trip and an Unplanned Power Reduction in Unit 2	1

South Texas Project, Unit 1
Loss of Offsite Power from Shunt Reactor Fire in Switchyard
Significance Determination Process Evaluation

Conclusion: The incremental conditional core damage probability (CCDP) for the loss of offsite power event, which occurred from a fire in shunt reactor 2 in the South Texas Project (STP) switchyard after its location was changed without properly evaluating that relocation per 10 CFR 50.59, was estimated to be 2.5E-6, or of low safety significance.

The analyst had high confidence in this estimate since this result was consistent with the results of the Final Accident Sequence Precursor (ASP) report for STP, Unit 1, "Loss of Offsite Power Resulting in Unit 1 Automatic Reactor Trip and Actuation of Emergency Diesel Generators and Auxiliary Feedwater Pumps" (ADAMS Accession Number ML25086A198), which calculated a mean CCDP of 3.6E-6. This ASP estimate and the CCDP estimated in this evaluation used slightly different inputs and methodologies, but yielded similar cutset, sequence, and numerical results.

Model Modifications: The analyst used a model based on STP SPAR model, version 8.80, run on SAPHIRE, version 8.2.12, that was modified by Idaho National Laboratory to include the following changes:

- Offsite power alignment. Safety buses E1A and E1C are normally supplied by the unit auxiliary transformer. The analyst added basic event ACP-TFM-FC-UT001A, "Failure of Unit Aux Transformer UT001A," and removed basic event ACP-TFM-FC-ST001A, "Failure of Standby Transformer ST001A," under fault trees ACP-STB1F-E and ACP-STB1H-E1C, to reflect this alignment. Also, under fault trees ACP-E1A, ACP-E1B, and ACP-E1C, basic event ACP-TFM-FC-ET001A, "Failure of Emergency Transformer ET001A," and associated human performance events ACP-XHE-XM-ET01ADA, "Operator Fails to Align Emergency Transformer ET001A to Bus E1A," ACP-XHE-XM-ET01ADB, "Operator Fails to Align Emergency Transformer ET001A to Bus E1B," and ACP-XHE-XM-ET01ADC, "Operator Fails to Align Emergency Transformer ET001A to Bus E1C" were added to credit use of the station's emergency transformer to power an engineered safety feature bus. The emergency transformer was ANDed with house event HE-SBO as the licensee only credits use of the emergency transformer during station blackout events.

- FLEX modeling. Because STP has developed Diverse and Flexible Coping (FLEX) Strategies, the analyst incorporated use of these FLEX strategies into the model. The analyst changed the basic event FLX-XHE-XM-ELAP, "Operators Fail to Declare ELAP when Beneficial," from a failure probability of 1.0 to 1.0E-2 to reflect nominal probabilistic success for the decision to employ these strategies. For the FLEX equipment failures, the analyst noted that the STP SPAR model used failure data for the FLEX equipment that was typical of installed permanent equipment at the plant. The analyst changed the failure rates to those from the Pressurized Water Reactor (PWR) Owners Group data. Finally, because STP has their FLEX diesel generators permanently installed and readily deployable, the analyst changed the failure rate for basic event for FLX-XHE-XM-4802 from TRUE to 5.0E-2.

- Feed-and-bleed success criteria. The success criteria for the feed-and-bleed strategy were adjusted to only requiring one of the two primary power-operated relief valves (PORVs) to successfully implement the strategy. This change in success criteria was made after the NRC Office of Nuclear Regulatory Research reviewed thermal-hydraulic analyses for the STP plants and considered those analyses consistent with those made in NUREG 2187, “Confirmatory Thermal-Hydraulic Analysis to Support Specific Success Criteria in the Standardized Plant Risk Models - Byron Unit 1,” which were used to adjust SPAR model success criteria. The analyst revised fault tree FAB to do this.

- ATWS pressure relief success criteria. The reactor coolant system pressure relief success criteria were adjusted to requiring two primary PORV or primary safety valve failures vice zero failures to fail the anticipated transient without scram (ATWS) pressure relief strategy. Analysts noted that this change in success criteria to the current revision of the SPAR model was needed after reviewing plant-specific calculations detailing the pressure relief valve capabilities for the STP units during ATWS events. This change appropriately eliminated conservatisms in the estimate of the CCDP by reducing the overestimation of the probabilities from core damage sequences containing ATWS events. The analyst revised fault tree RCSPRESS to do this.

- Crediting steam dumps and steam generator safety valves. The base SPAR model only credited the steam generator PORVs for steaming of the steam generators to reject heat after postulated events. The model was enhanced to recognize that the steam dump system would be available for events where the condenser remained available and to recognize that the steam generator safety valves would be available as an alternate heat rejection path.

- Alternate room cooling. The licensee informed the analyst that upon their loss of normal cooling, alternate room cooling for electrical equipment rooms would not be available during loss of offsite power events. The model was updated to disallow room cooling during loss of offsite power events by ANDing the loss of normal room cooling with operator action to establish alternate room cooling and house events which flag for losses of offsite power.

Internal Events: For the evaluation, the analyst assumed that the loss of offsite power resulted from the fire in shunt reactor 2. In the Event and Condition Assessment module of SAPHIRE for the modified model, the analyst ran an initiating events analysis for a loss of condenser heat sink, which sets the initiating event frequency for a switchyard centered loss of offsite power to 1.0 and all other initiating event frequencies to 0.0. Additionally, the analyst set the following basic events to 1.0E-3, corresponding to the human error probability derived for recovering offsite power to a safety bus in the Final ASP report for STP, Unit 1, “Loss of Offsite Power Resulting in Unit 1 Automatic Reactor Trip and Actuation of Emergency Diesel Generators and Auxiliary Feedwater Pumps” (ADAMS Accession Number ML25086A198):

- ACP-XHE-XM-ET01ADA, “Operator Fails to Align Emergency Transformer ET001A to Bus E1A”
- ACP-XHE-XM-ET01ADB, “Operator Fails to Align Emergency Transformer ET001A to Bus E1B”

- ACP-XHE-XM-ET01ADC, “Operator Fails to Align Emergency Transformer ET001A to Bus E1C”
- OEP-XHE-XL-NR01HSC, “Operator Fails to Recover Offsite Power in 01 Hour (Switchyard)”
- OEP-XHE-XL-NR02HSC, “Operator Fails to Recover Offsite Power in 02 Hours (Switchyard)”
- OEP-XHE-XL-NR04HSC, “Operator Fails to Recover Offsite Power in 04 Hours (Switchyard)”
- OEP-XHE-XL-NR04HSC1, “Operator Fails to Recover Offsite Power in 04 Hours (Switchyard Centered) Given Failure at 1 Hour”
- OEP-XHE-XL-NR06HSC, “Operator Fails to Recover Offsite Power in 06 Hours (Switchyard)”
- OEP-XHE-XL-NR08HSC, “Operator Fails to Recover Offsite Power in 08 Hours (Switchyard)”
- OEP-XHE-XL-NR08HSC1, “Operator Fails to Recover Offsite Power in 08 Hours (Switchyard Centered) Given Failure at 1 Hour”

The inputs resulted in an estimate of the CCDP from internal events from the performance deficiency of 2.53E-6. The analyst then subtracted out the core damage probability obtained from applying the nominal initiating event frequency for a switchyard center loss of offsite power (9.0E-3/year) to the model with all other failure probabilities being nominal, which yielded an incremental CCDP of 2.5E-6. The dominant sequence, which accounted for approximately 85 percent of the CCDP estimate, was a switchyard centered loss of offsite power in which offsite power was recovered and all auxiliary feedwater capability was lost along with the ability to perform reactor coolant system feed-and-bleed operations.

External Events: The performance deficiency resulted in a loss of offsite power initiating event. The probability of an external event occurring concurrently with the performance deficiency is extremely low and therefore the CCDP from such events would be negligible. The CCDP from internal events was determined to be the best representation of the risk of this event.

Conditional Large Early Release Probability: The analyst evaluated the Conditional Large Early Release Probability (CLERP) using Inspection Manual Chapter 0609, Appendix H, “Containment Integrity Significance Determination Process.” The top core damage sequences screened out as being risk significant using Table 6.1, “Phase 1 Screening – Type A Findings at Full Power.” From this, the analyst considered that the significance was best estimated by using the CCDP of the issue.

Licensee Results: The licensee informed the analyst that their estimate of the CCDP for this event using their model was of low safety significance with similar core damage sequences produced by the STP SPAR model and gave no further clarifications of specific values.

Sensitivities: The analyst estimated the CCDP for changes to key assumptions made in the evaluation. Upon review of the models, licensee PRA personnel and the analyst noted differences in failure data between their respective models. The analyst adjusted several events to appreciate their contribution.

- Use of the licensee value for tank failures: The failure rate for tanks in the SPAR model includes all tanks and not just condensate tanks. Licensee personnel expressed that use

of this failure rate overestimates the value for what was applicable to condensate tank failures. The analyst substituted the licensee's condensate tank failure value into the SPAR model to obtain an estimate of CCDP of 2.29E-6.

- Exclusion of pump volute failures: Unlike the SPAR model, the licensee model did not have the common cause failure event for auxiliary feedwater pump volutes. The analyst removed this input from the SPAR model results to obtain an estimate of CCDP of 2.36E-6.

- Use of licensee failure probability for operators failing to initiate feed-and-bleed: The licensee developed a new estimate for the human error probability for feed-and-bleed applicable to the event with a value of 5.52E-3. The analyst used this value in the SPAR model to obtain an estimate of CCDP of 2.22E-6.

- Use of licensee emergency diesel generator failure data: The licensee stated that they use site-specific data for failures of the emergency diesel generators in their model and not industry data. The analyst ran the licensee failure data in the SPAR model to obtain an estimate of CCDP of 9.88E-7.

- Use of licensee failure rate for the turbine driven auxiliary feedwater pump: The licensee noted that the failure rate for the basic event for the turbine driven auxiliary feedwater pump failing to run in the SPAR model was approximately twice as large as the value in their model. The analyst ran the sensitivity, which yielded an estimate of CCDP of 1.79E- 6.

Uncertainties: The analyst performed an uncertainties analysis using the Monte Carlo method with 5000 runs of the internal events estimate in SAPHIRE. The distribution of results was tight around the point estimate with a large majority of the results in the range of 1.0E-6 to 1.0E-5. The median CCDP estimate of the uncertainties was 2.29E-6; the mean CCDP estimate was 3.2E-6; and approximately 70 percent of the results were greater than 1.0E-6 (White).