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

June 13, 2024

MEMORANDUM TO: John D. Monninger, Regional Administrator

THRU: Geoffrey B. Miller, Director
Division of Operating Reactor Safety

Hay, Michael signing on behalf of Miller, Geoffrey on 06/11/24

FROM: Patricia J. Vossmar, Branch Chief
Projects Branch A
Division of Operating Reactor Safety

Signed by Vossmar, Patricia on 06/11/24

SUBJECT: MANAGEMENT DIRECTIVE 8.3 EVALUATION FOR SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION, UNIT 2 AUTOMATIC REACTOR TRIP ON MAY 12, 2024

Pursuant to Regional Office Policy Guide 0801, "Management Directive 8.3 and Inspection Manual Chapter 0309 Reactive Team Inspection Decisions, Implementation, and Documentation for Power Reactors," the enclosed table provides the Management Directive 8.3 evaluation for South Texas Project Electric Generating Station, Unit 2, for an automatic reactor trip due to interruption of power from the unit auxiliary transformer caused by an apparent electrical relay failure. Staff performed this evaluation to determine the risk significance of the event to determine the appropriate level of NRC response. Based on this evaluation, the staff recommends a special inspection be performed for follow-up of this event.

Concur with Recommendation:

Signed by Monninger, John on 06/24/24

John D. Monninger
Regional Administrator

Date

Enclosures:
MD 8.3 Decision Documentation Form
(Deterministic and Risk Criteria Analyzed)

CONTACT: Patricia Vossmar, DORS/PBA
817-200-1144

MANAGEMENT DIRECTIVE 8.3 EVALUATION FOR SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION, UNIT 2 AUTOMATIC REACTOR TRIP ON MAY 12, 2024 - DATED JUNE 13, 2024

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DOCUMENT NAME: MANAGEMENT DIRECTIVE 8.3 EVALUATION FOR SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION, UNIT 2 AUTOMATIC REACTOR TRIP ON MAY 12, 2024

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x SUNSI Review By: RLB	ADAMS þ Yes o No	þ Non-Sensitive o Sensitive	þ Publicly Available o Non-Publicly Available	Keyword: NRR-123

OFFICE	SPE:DORS/A	SRA:DORS	C:DORS/A	D:DORS	RIV:RA
NAME	RBywater	RDeese	PVossmar	GMiller	JMonninger
SIGNATURE	**RLB**	***/RA/***	***/RA/***	***/RA/***	***/RA/***
DATE	6/11/2024	6/11/2024	6/11/2024	6/11/2024	6/13/2024

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MANAGEMENT DIRECTIVE 8.3
DECISION DOCUMENTATION FORM
(Deterministic and Risk Criteria Analyzed)

PLANT:	**South Texas Project, Unit 2**	**EVENT DATE:**	**May 12, 2024**
RESPONSIBLE BRANCH CHIEF:	**Patricia Vossmar**	**EVALUATION DATE:**	**May 30, 2024**

BRIEF DESCRIPTION OF THE SIGNIFICANT OPERATIONAL EVENT OR DEGRADED CONDITION:

On May 12, 2024, at 4:41 pm CDT, South Texas Project (STP), Unit 2, automatically tripped from approximately 15 percent power. The trip was caused by a loss of power from the unit auxiliary transformer due to actuation of its lockout relay, which resulted in a loss of power to the reactor coolant pumps and a reactor protection system actuation. Instrumentation and Controls (I&C) technicians were completing an unrelated work order in the same cabinet in the control room that contained the relay at the time of the event. Earlier in the day, the unit had been starting up from a planned refueling outage, and the turbine-generator was offline at the time of the event.

The interruption of power from the unit auxiliary transformer resulted in a loss of power to non-safety-related equipment including the reactor coolant pumps and the main condenser. Consistent with the STP electrical distribution design, the loss of power from the unit auxiliary transformer also caused an interruption of power to the A and C engineered safety features buses. Emergency diesel generators (EDG) 21 and 23 automatically started and provided power to the A and C buses, respectively. The B engineered safeguards bus was aligned to the standby transformer offsite power source and remained energized from offsite power throughout the event.

The operations crew responded to the event and stabilized the unit in a safe condition using natural circulation of the reactor coolant system. Decay heat removal was provided by the steam generator power-operated relief valves (PORVs).

Due to the reactor protection system actuation (RPS) while critical, this event was reported as a four-hour, non-emergency notification per 10 CFR 50.72(b)(2)(iv)(B). This event was also reported per 10 CFR 50.72(b)(3)(iv)(A) as an event that resulted in a valid actuation of the emergency diesel generators. The NRC received this report, and it is documented as event notification (EN) 57124 on the NRC's public website.

There were a couple of equipment problems that occurred during the event. Control room operators noticed that the 480 V load center E2A feeder breaker failed to automatically close on a load sequencer signal and provide power to 480 V motor control centers (MCC) E2A1 and E2A3. The operators unsuccessfully attempted to close the breaker manually from the control room, placed the breaker hand switch in pull-to-lock, and requested electrical maintenance support. These MCCs provide power to several A-train motor-operated valves, and essential cooling water pump 2A and EDG 21 essential support components. Electrical maintenance performed a visual inspection and reported there was no obvious deficiency with the breaker. At 5:07 pm CDT, the operators removed the breaker hand switch from the pull-to-lock position and the breaker closed automatically, restoring power to the loads.

MANAGEMENT DIRECTIVE 8.3
DECISION DOCUMENTATION FORM
(Deterministic and Risk Criteria Analyzed)

PLANT: South Texas Project, Unit 2	EVENT DATE: May 12, 2024

Additionally, when control room operators transferred steam generator PORVs from automatic to manual control to use them for decay heat removal, the C PORV failed to the full-open position when the operator began to manually open the valve. The operator closed the valve using manual control and tried a second time with the same result. The operator closed the C PORV and continued decay heat removal using the remaining three steam generator PORVs. Main steam isolation valves were manually closed due to a loss of the condenser as an available heat sink. Auxiliary feedwater was used to feed the steam generators.

During the event, reactor coolant pump 2B developed a high seal leakoff flow rate, indicating a degraded seal. While one stage of the multi-stage seal had failed, integrity of the pump seal was maintained as a result of the other intact seal stages. The licensee decided it was necessary to continue the cooldown to cold shutdown conditions to replace the reactor coolant pump seal.

Shortly after the event occurred, the licensee worked to realign available offsite power sources to supply the safety and non-safety electrical buses. Standby bus 2F was reenergized from the Unit 2 standby transformer at 5:29 pm CDT, and the safety related 2A bus was transferred off the diesel generator and realigned to receive power from standby bus 2F at 11:33 pm CDT. Standby bus 2H was reenergized from the Unit 1 standby transformer at 5:47pm CDT, and the safety related 2C bus was realigned from the diesel generator to receive power from standby bus 2H at 5:30 am CDT the next morning, May 13, 2024. Normal decay heat removal was placed in service using the residual heat removal pumps and the unit entered cold shutdown on May 14.

Y/N	DETERMINISTIC CRITERIA
N	**Involved operations that exceeded, or were not included in, the design bases of the facility**
	Remarks: The event was within the design basis of the facility.
N	**Involved a major deficiency in design, construction, or operation having potential generic safety implications**
	Remarks: The event did not reveal any major deficiencies, nor did it have generic safety implications.
N	**Led to a significant loss of integrity of the fuel, primary coolant pressure boundary, or primary containment boundary of a nuclear reactor**
	Remarks: There was not a loss of any barriers during this event
Y	**Led to the loss of a safety function or multiple failures in systems used to mitigate an actual event**
	Remarks: There were multiple failures in the systems used to mitigate an actual event. The C steam generator PORV failed to the full-open position when the licensee attempted to manually open it and the A-train load center output breaker providing power to MCC E2A1 and MCC E2A3 failed to close.
N	**Involved possible adverse generic implications**
	Remarks: The trip did not have generic safety implications
N	**Involved significant unexpected system interactions**
	Remarks: The trip did not involve significant unexpected system interactions.
Y	**Involved repetitive failures or events involving safety-related equipment or deficiencies in operations**
	Remarks: Prior to this Unit 2 event involving a failed safety-related steam generator PORV, there were three failures of steam generator PORVs in 2024 on Unit 1 (one failure of A PORV and two failures of the C PORV). The NRC issued a Green NCV in Inspection Report 498;499/2024001 for inadequate corrective actions for the failure of the Unit 1, C PORV.
N	**Involved questions or concerns pertaining to licensee operational performance**
	Remarks: Operators responded appropriately to the event.
N	**Involved circumstances sufficiently complex, unique, or not well enough understood, or involved safeguards concerns, or involved characteristics the investigation of which would best serve the needs and interests of the Commission.**
	Remarks: This event did not involve circumstances sufficiently complex, unique, or not well enough understood; nor involve safeguards concerns. This event did not involve such unusual characteristics requiring an investigation to serve the needs and interests of the Commission.

Y/N	DETERMINISTIC CRITERIA
N	**Emergency Preparedness, Radiation Protection, and/or Security/Safeguards Deterministic Criteria**
	Remarks: None of the emergency preparedness, radiation protection, and/or security/safeguards deterministic criteria could be answered in the affirmative for this event.

CONDITIONAL RISK ASSESSMENT

IF IT IS DETERMINED THAT A RISK ANALYSIS IS NOT REQUIRED - ENTER NA BELOW AND CONTINUE TO THE DECISION BASIS BLOCK

RISK ANALYSIS BY: Rick Deese	DATE: May 30, 2024

Brief description for the basis of the assessment (may include assumptions, calculations, references, peer review, or comparison with licensee's results):

For this risk assessment, the analyst used the test and limited model South Texas Project SPAR model TLU1, Version 8.80, run on SAPHIRE, Revision 8.2.10. The following modifications were made to the model to better model the plant conditions and failures:

1. The success criteria for the feed and bleed strategy were adjusted to only requiring 1 of the 2 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 South Texas Project 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 complemented pertinent basic events in fault tree FAB to do this.
2. Because of the low decay heat load at the time of the event resulting from the unit's recent refueling outage, the analyst adjusted basic event HPI-XHE-XM-FAB, Operator Fails to Initiate Feed and Bleed Cooling. The performance shaping factor for available time was changed from barely adequate time to nominal time, resulting in a change of the failure probability of the basic event from 2.0E-2 to 2.0E-3.
3. The reactor coolant system pressure relief success criteria were adjusted to requiring 2 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 South Texas Project units during ATWS events. This change appropriately eliminated conservatisms in the estimate of the conditional core damage probability by reducing the overestimation of the probabilities from core damage sequences containing ATWS events. The analyst complemented pertinent basic events in fault tree RCSPRESS to do this.

CONDITIONAL RISK ASSESSMENT

4. The failure probability of basic event FLX-XHE-XM-ELAP, Operators Fail to Declare ELAP when Beneficial, was adjusted from 1.0 to 1.0E-2 to give credit for use of mitigating strategies using FLEX equipment.
5. At the time of the event, safety buses E1A and E1C were supplied by the unit auxiliary transformer. The analyst added basic event ACP-TFM-FC-UT002A, Failure of Unit Aux Transformer UT002A, and removed basic event ACP-TFM-FC-ST001A, Failure of Standby Transformer ST001A, under fault trees ACP-E1A1-SB1F and ACP-E1C-1H, to reflect this alignment.
6. The analyst created new basic event ACP-CRB-OO-LCE1A1, Feeder Breaker for Load Center E1A1 Fails to Close after Load Shed, to fault tree ACP-E1A-480V, to account for the observed failure of the breaker during the event. This basic event was ANDed with a new basic event ACP-XHE-XM-E1A1RESTORE, Operators Fail to Restore Load Center after Load Shed, to allow for recovery from the failure of the breaker to close. Inclusion of this recovery basic event recognized a range of possibilities of operators being able to reclose the breaker, including the actual time it took during the event. The basic event was created by using nominal ratings for all performance shaping factors except for time available for which extra time was credited.

The event was run as an initiating events analysis using the Events and Conditions Assessment module of SAPHIRE. To estimate the risk of the event in this module, the following assumptions were made:

1. The event was run as a loss of condenser heat sink event since functionality of the main condenser as a heat sink was lost for event mitigation. The initiating event frequency for a loss of condenser heat sink was set to 1.0 and the initiating event frequency for all other initiating events was set to 0.0.
2. Basic event ACP-TFM-FC-UT002A, Failure of Unit Aux Transformer UT002A, was set to TRUE to model the loss of offsite power to safety buses E1A and E1C.
3. Basic Event ACP-CRB-OO-LCE1A1, Feeder Breaker for Load Center E1A1 Fails to Close after Load Shed, was set to TRUE to reflect the initial failure of the breaker to close.
4. The failure probability of the PORV steam generator C was doubled from 1.61E-2 to 3.22E-2, to reflect the possibility that the internal malfunction the licensee discovered in troubleshooting the PORV could have prevented its functioning under some conditions. The SPAR model made automatic adjustments to account for the possibility of the other PORVs experiencing similar failures to common cause.
5. To account for the increased reactor coolant pump seal leakage in reactor coolant pump 2B, the analyst set basic event RCS-MDP-LKBP2, RCP Seal Stage 2 Integrity (Binding/Popping Open) Fails, to TRUE. This adjustment made little impact on the estimate of conditional core damage probability because the risk increase from a reactor coolant pump seal LOCA was diminished by the licensee’s past plant modification which installed low leakage seals.
6. Per NRC practice for performing Management Directive 8.3 risk assessments, the analyst set all test and maintenance failure probabilities to 0.0.

CONDITIONAL RISK ASSESSMENT

Applying these assumptions and model modifications led to an estimate for the conditional core damage probability of 2.1E-5. The dominant accident sequence leading to core damage was a loss of condenser heat sink event where the steam generator PORVs fail to function and the feed and bleed strategy for decay heat removal fails.

The analyst discussed these results with the licensee in a phone call on May 23, 2024. The licensee performed a risk assessment after adjusting their base PRA model to credit successful recovery of the closing function of the feeder breaker for load center E2A1 and estimated a conditional core damage probability of 8.0E-7 for the event. Some modeling differences were noted which contributed to differences in the NRC's and licensee's estimates of conditional core damage probability. Notably, the licensee modeled the event as a plant-centered loss of offsite power in their PRA model which is not consistent with how NRC risk assessment methodologies would model the actual plant event. Treatment of the event as a loss of offsite power in the NRC SPAR model would also inappropriately invoke full credit for recovery from loss of offsite power events when offsite power was not considered lost. Also, the licensee used lower failure rates in their PRA model for establishing alternate room cooling for sequences which involve probabilistic loss of room cooling.

This analysis was reviewed and concurred on by a senior reactor analyst from the Office of Nuclear Reactor Regulation/Division of Risk Assessment.

THE ESTIMATED INCREMENTAL CONDITIONAL CORE DAMAGE PROBABILITY (CCDP) IS:	2.1×10^{-5}
WHICH PLACES THE RISK IN THE RANGE OF:	SIT/AIT Overlap

RESPONSE DECISION

USING THE ABOVE INFORMATION AND OTHER KEY ELEMENTS OF CONSIDERATION AS APPROPRIATE, DOCUMENT THE RESPONSE DECISION TO THE EVENT OR CONDITION, AND THE BASIS FOR THAT DECISION

DECISION AND DETAILS OF THE BASIS FOR THE DECISION:

Region IV staff concluded that two of the deterministic criteria were met and the estimated incremental conditional core damage probability was in the range of 2.1×10^{-5}, and as a result, a reactive inspection should be considered. Considering the risk result, the types of equipment failures experienced, the perceived complexity of the event, and the need for inspection staff resources with electrical engineering expertise, the Region IV staff determined that a reactive inspection was recommended. In consultation with headquarters staff, Region IV determined that a Special Inspection Team rather than an Augmented Inspection Team is the appropriate response due to the limited number of inspection items requiring NRC follow-up, the uncertainty in the NRC risk model and its assumptions, and the lack of generic implications associated with this event.

BRANCH CHIEF REVIEW: Patricia Vossmar	DATE: May 30, 2024
DIVISION DIRECTOR REVIEW: Geoffrey Miller	DATE: May 30, 2024
ADAMS ACCESSION NUMBER: EVENT NOTIFICATION REPORT NUMBER (as applicable): 57124 E-mail to NRR_Reactive_Inspection@nrc.gov	