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

June 23, 2025

EAF-RIV-2025-0126

Joseph Sullivan, Site Vice President
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC INSPECTION REPORT 05000382/2025092 AND PRELIMINARY WHITE FINDING

Dear Joseph Sullivan:

On May 30, 2025, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Waterford Steam Electric Station, Unit 3. On June 17, 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.

The enclosed report discusses a preliminary White finding (i.e., a finding with low to moderate safety significance that may require additional NRC inspections), with an associated apparent violation. As described in the enclosed report, NRC inspectors determined the failure to develop and implement a replacement preventive maintenance strategy for relays in the emergency diesel generators was a performance deficiency that was within the licensee's ability to foresee and correct.

As described in Section 71111.12 of the enclosed report, on December 11, 2024, during surveillance testing of emergency diesel generator (EDG) A, operators observed oscillations in output voltage and frequency and declared the diesel inoperable. Subsequent troubleshooting through December 22, 2024, revealed faults associated with the automatic voltage regulator K4 relay. In 2004, the site received industry forewarning of obsolescence issues related to the EDG voltage regulator and associated relays; in 2016, the licensee contracted a third party who identified vulnerabilities with the voltage regulator and associate relays. However, the licensee failed to adequately establish a preventive maintenance strategy for the replacement of the K4 relay, which resulted in the inoperability of EDG A and incurred unplanned risk during the extended outage. The NRC assessed the significance of the finding using the significance determination process (SDP) and readily available information. The final resolution of this finding will be conveyed in separate correspondence.

The finding has an associated apparent violation which is being considered for escalated enforcement in accordance with the NRC Enforcement Policy, which can be found on the NRC website at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. The apparent violation involves the failure to adequately establish a preventive maintenance strategy for the inspection or replacement of parts that have a specific lifetime, in accordance with Technical Specification 6.8.1.a, "Procedures and Programs."

In accordance with NRC Inspection Manual Chapter 0609, we intend to complete our evaluation using the best available information and issue our final significance determination and enforcement decision, in writing, within 90 days from the date of this letter. The NRC's significance determination process encourages an open dialogue between your staff and the NRC; however, the dialogue should not impact the timeliness of our final determination.

Before we make a final decision on this matter, we are providing you with an opportunity to either: (1) attend a regulatory conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a regulatory conference, it should be held within 40 days of the receipt of this letter, and we encourage you to submit supporting documentation at least one week prior to the conference to make the conference more efficient and effective. The focus of the regulatory conference is to discuss the significance of the finding and not necessarily the root cause(s) or corrective action(s) associated with the finding. If a regulatory conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 40 days of your receipt of this letter.

If you choose to respond in writing, it should be clearly marked as a "Response to Apparent Violation in NRC Inspection Report 05000382/2025092; EAF-RIV-2025-0126" and should include for the apparent violation: (1) the reason for the apparent violation or, if contested, the basis for disputing the apparent violation; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken; and (4) the date when full compliance will be achieved. Your response may reference or include previously docketed correspondence if the correspondence adequately addresses the required response. 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.

Additionally, your written response should be sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001 with a copy to the Director, Division of Operating Reactor Safety, U.S. Nuclear Regulatory Commission, Region IV, 1600 East Lamar Blvd., Arlington, Texas 76011-4511, and the NRC Resident Inspector at Waterford Steam Electric Station, Unit 3, and emailed to R4Enforcement@nrc.gov, within 40 days of the date of this letter. If an adequate response is not received within the time specified or an extension of time has not been granted by the NRC, the NRC will proceed with its enforcement decision or schedule a regulatory conference.

Please contact John Dixon at (817) 200-1574 within 10 days from the issue date of this letter to notify the NRC of your intention to attend a regulatory conference or provide a written response. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision.

If you decline to request a regulatory conference or to submit a written response, you relinquish your right to appeal the final significance determination process determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of NRC Inspection Manual Chapter 0609.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room and from the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html.

If you have any questions concerning this matter, please contact John Dixon of my staff at (817) 200-1574.

Sincerely,

Signed by Miller, Geoffrey on 06/23/25

Geoffrey B. Miller, Director
Division Of Operating Reactor Safety

Docket No. 05000382
License No. NPF-38

Enclosures:
1. Inspection Report
2. Detailed Risk Evaluation

cc w/ encl: Distribution via LISTSERV

WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC INSPECTION REPORT 05000382/2025092 AND PRELIMINARY WHITE FINDING – DATED JUNE 23, 2025

DISTRIBUTION:

RidsOeMailCenter Resource
RidsSecyMailCenter Resource
RidsEdoMailCenter Resource
RidsOiMailCenter Resource
RidsRgn3MailCenter Resource
JMonninger, RA
GMiller, DORS
NTaylor, DORS
JDixon, DORS
ASanchez, DORS
KChambliss, DORS
VDricks, ORA
TSmith, ORA
LReyna, DORS
KCook-Smith, DORS
AGarmoe, NRR
RidsNrrMailCenter Resource
RidsOcaMailCenter Resource
EDO_Managers
RidsRgn1MailCenter Resource
RidsOcfoMailCenter Resource
RAlexander, ORA
BAlferink, ORA
ARoberts, ORA
DCylkowski, ORA
TSteadham, OEDO
AMoreno, OCA
LWilkins, OCA
CYoung, DORS
RidsOgcMailCenter Resource
RidsOigMailCenter Resource
RidsNsirMailCenter Resource
RidsRgn2MailCenter Resource
RidsOpaMail Resource
R4Enforcement
R4-DORS-IPAT
DPelton, OE
JCai, OE
JPeralta, OE
DBradley, OE
EBrothman, NRR
MKhanna, NRR
RFelts, NRR
JDrake, NRR
SVasavada, NRR

DOCUMENT NAME: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – NRC INSPECTION REPORT 05000382/2025092 AND PRELIMINARY WHITE FINDING
ADAMS ACCESSION NUMBER: **ML25168A320**

☒ SUNSI Review By: ACR		☒ Non-Sensitive ☐ Sensitive		☒ Publicly Available ☐ Non-Publicly Available	
OFFICE	SRI:DORS/PBD	RI:DORS/PBD	SPE:DORS/PBD	BC:DORS/PBD	SRA:DORS
NAME	KChambliss	KCook-Smith	ASanchez	JDixon	CYoung
SIGNATURE	***/RA/***	***/RA/***	***/RA/***	***/RA/***	***/RA/***
DATE	06/18/25	06/18/25	06/18/25	06/18/25	06/18/25
OFFICE	TL:ACES	RC	OE	D:DORS	
NAME	BAlferink	DCylkowski	DBradley	GMiller	
SIGNATURE	***/RA/***	***/RA/***	***/RA/***	***/RA/***	
DATE	06/18/25	06/18/25	06/20/25	06/23/25	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000382
License Number:	NPF-38
Report Number:	05000382/2025092
Enterprise Identifier:	I-2025-092-0000
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Killona, LA 70057
Inspection Dates:	December 11, 2024, to May 30, 2025
Inspectors:	K. Chambliss, Senior Resident Inspector K. Cook-Smith, Resident Inspector A. Sanchez, Senior Project Engineer C. Young, Senior Reactor Analyst
Approved By:	John L. Dixon, Jr., Branch Chief Reactor Projects Branch D Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a NRC inspection at Waterford Steam Electric Station, Unit 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Preventive Maintenance Planning Failed to Prevent Voltage Regulator Relay Failure Resulting in Emergency Diesel Generator A Inoperability			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Preliminary White EAF-RIV-2025-0126 AV 05000382/2025092-01 Open	[H.14] - Conservative Bias	71111.12
The inspectors reviewed a self-revealed finding of preliminary White significance and an associated apparent violation of Technical Specification 6.8.1.a for the licensee's failure to implement a maintenance procedure as listed in Regulatory Guide 1.33, "Quality Assurance Program Requirements," revision 2, Appendix A, Section 9.b. Specifically, the licensee failed to develop and implement preventive maintenance strategies for relays in the emergency diesel generators in accordance with procedure EN-DC-324, "Preventive Maintenance Program," revision 17.			

Additional Tracking Items

None.

INSPECTION SCOPES

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

REACTOR SAFETY

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) emergency diesel generator A voltage and frequency oscillations event, December 11, 2024

INSPECTION RESULTS

Preventive Maintenance Planning Failed to Prevent Voltage Regulator Relay Failure Resulting in Emergency Diesel Generator A Inoperability			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Preliminary White EAF-RIV-2025-0126 AV 05000382/2025092-01 Open	[H.14] - Conservative Bias	71111.12
The inspectors reviewed a self-revealed finding of preliminary White significance and an associated apparent violation of Technical Specification 6.8.1.a for the licensee’s failure to implement a maintenance procedure as listed in Regulatory Guide 1.33, “Quality Assurance Program Requirements,” revision 2, Appendix A, Section 9.b. Specifically, the licensee failed to develop and implement preventive maintenance strategies for relays in the emergency diesel generators in accordance with procedure EN-DC-324, “Preventive Maintenance Program,” revision 17.			
Description: On December 11, 2024, emergency diesel generator (EDG) A was declared inoperable due to oscillations in output voltage and frequency that were observed during a planned surveillance test. The voltage regulator relay was initially identified as the cause of the oscillations and was replaced. During the subsequent run of EDG A on December 11, 2024, voltage and frequency oscillations slowed, however, rated voltage was not able to be achieved. Further investigations led to the replacements of the remote gate firing module, silicon-controlled rectifiers, various relays, and the automatic voltage regulator from December 13 through 15, 2024:			

1. On December 13, 2024, Dewetron data pointed to the silicon-controlled rectifiers being suspect, and one of the rectifiers along with the K7 and field flash (FF) relays were replaced
2. On December 14, 2024, the automatic voltage regulator was replaced with a spare
3. On December 15, 2024, the automatic voltage regulator and remote gate firing module were replaced

With the exception of the original automatic voltage regulator (which had a failed R4 potentiometer), all these removed parts were bench tested satisfactorily after their removal. Further data collection during runs of the diesel were conducted from December 16 through December 20, 2024. During this troubleshooting, the emergency diesel generator began tripping on generator differential trip signals. A heightened unplanned “yellow” risk window was entered on December 21, 2024, due to entering Risk-Informed Completion Time. On December 22, 2024, the licensee conducted checks of the manual circuits and discovered that multiple normally closed contacts were indicated as open on either the K3 or K4 relays. These relay assemblies were replaced, and the diesel was retested and ran satisfactorily. The diesel was then declared operable after an operability run on December 24, 2024. Analysis of the components found that the K4 relay had contacts that failed to change state as expected and was the cause of the failure of the diesel. During this troubleshooting, EDG A was started and stopped, or started and tripped, a total of 15 times during the forced outage of the diesel.

Subsequent causal analysis of the relays found that on November 11, 2001, a similar failure occurred on the same emergency diesel generator and a replacement preventive maintenance strategy was put in place to replace the voltage regulator and K7 relay for both EDGs A and B. On June 1, 2016, the licensee contracted a third party vendor to conduct a study that identified vulnerabilities with the voltage regulator and associated relays; the previous preventive maintenance strategies were expanded to include replacement of the FF, K1, K2, K3, K4, and linear reactor relays on September 22, 2016.

In accordance with EN-DC-324, “Preventive Maintenance Program,” revision 17, first time implementation should be less than 2 years for high critical, non-outage preventive maintenance tasks with a greater than 5-year periodicity. Contrary to that procedure, the first-time implementation was not scheduled until 2023. Between 2023 and 2024 multiple attempts were made to schedule this maintenance, however the licensee rescheduled it multiple times due to a perceived lack of parts availability though parts were available throughout the industry. On May 7, 2024, the preventive maintenance tasks were retired due to the misconception that parts were unavailable due to obsolescence. This justification was inadequate to retire the preventive maintenance tasks, and the relays were never replaced. Consequently, the K4 relay failed rendering EDG A inoperable.

Corrective Actions: The licensee initiated a condition report to enter this issue into the corrective action program. The condition report initiated an adverse condition analysis with the following corrective actions: 1) reestablish the preventive maintenance to replace all of the relays associated with the voltage regulator circuit, 2) document the correct availability status and obsolescence status of the relays, voltage regulators, remote gate firing module, silicone--controlled rectifiers, and diodes, and 3) perform an analysis and implement performance improvements to determine and correct factors that played into not setting the proper due date for the relay replacements and the unjust cancellation of the preventive maintenance.

Corrective Action References: The licensee entered this issue into the corrective action program with condition report CR-WF3-2024-05830.

Performance Assessment:

Performance Deficiency: The failure to develop and implement a replacement preventive maintenance strategy in accordance with EN-DC-324, “Preventive Maintenance Program,” for the relays in emergency diesel generator A and B was a performance deficiency. Specifically, the failure to develop and implement the preventive maintenance strategy to replace the relay resulted in EDG A failing its surveillance test and being inoperable.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to properly develop and implement replacement preventive maintenance strategies resulted in the failure of the K4 relay, and the subsequent inoperability of emergency diesel generator A.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The inspectors determined the finding impacted the Mitigating Systems cornerstone and used Exhibit 2 to evaluate the condition. The inspectors determined the finding represented a loss of probabilistic risk assessment (PRA) function of one train of a multi-train technical specifications (TS) system for greater than its TS allowed outage time. Specifically, a loss of function of EDG A existed for greater than the allowed outage time specified by TS 3.8.1.1. Therefore, a detailed risk evaluation was performed by a regional senior reactor analyst.

This evaluation is provided as Enclosure 2. Based on the results of this evaluation, the finding was determined to have a preliminary significance of low-to-moderate safety significance (White).

Cross-Cutting Aspect: H.14 - Conservative Bias: Individuals use decision making-practices that emphasize prudent choices over those that are simply allowable. A proposed action is determined to be safe to proceed, rather than unsafe to stop. Specifically, the licensee did not use prudent decision-making practices in implementing the replacement preventive maintenance strategy, which resulted in the maintenance not being completed within two years as required by licensee procedure. Assumptions were made relating to the obsolescence of the relays, which lead to the delay in acquiring replacement relays and completion of the preventive maintenance. Conservative decision-making was not used in the justification for the eventual cancellation of the preventive maintenance.

Enforcement:

Violation: Technical Specification 6.8.1.a, “Procedures and Programs,” states in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures listed in Regulatory Guide 1.33, “Quality Assurance Program Requirements,” revision 2, Appendix A, February 1978.

Regulatory Guide 1.33, revision 2, Appendix A, Section 9.b, “Procedures for Performing Maintenance,” states, in part, that preventive maintenance schedules should be developed to specify replacement of parts that have a specific lifetime. The licensee established procedure

EN-DC-324, “Preventive Maintenance Program,” revision 17, in part, to meet this requirement.

Contrary to the above, from 2016 to December 11, 2024, the licensee failed to adequately develop and implement a preventive maintenance strategy for inspection or replacement of parts that have a specific lifetime. Specifically, the licensee failed to adequately develop and implement a preventive maintenance strategy for the replacement of the emergency diesel generator relays, which resulted in the inoperability of EDG A and incurred unplanned risk during the extended outage.

Enforcement Action: This violation is being treated as an apparent violation pending a final significance (enforcement) determination.

EXIT MEETINGS AND DEBRIEFS

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

- On June 17, 2025, the inspectors presented the NRC inspection results to Joseph Sullivan, Site Vice President and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.12	Corrective Action Documents	CR-WF3-YYYY-XXXXX	2024-05830	
71111.12	Procedures	EN-DC-324	Preventative Maintenance Program	17

Detailed Risk Evaluation
Plant Name/Unit Number: Waterford 3
Inspection Report #: 2025-092
Enforcement Action #: EAF-RIV-2025-0126

BACKGROUND

On December 11, 2024, the emergency diesel generator (EDG) train A was declared inoperable due to oscillations in output voltage and frequency that were observed during scheduled testing. Subsequent troubleshooting through December 22, 2024, revealed faults associated with the automatic voltage regulator K4 relay. Following corrective maintenance, the EDG A was restored to an operable condition on December 24, 2024.

PERFORMANCE DEFICIENCY

The failure to develop and implement a replacement preventive maintenance strategy for relays in the EDGs was a performance deficiency.

IMPACT ON SAFETY FUNCTION(S)

The analyst assumed that the performance deficiency resulted in a degraded condition where the EDG A would not have been capable of performing its design basis function for its PRA mission time of 24 hours. The safety function provided by the EDG would be necessary to mitigate any design basis event that includes a loss of offsite power (LOOP) condition.

EXPOSURE TIME

The analyst reviewed results of previous surveillance testing of the EDG A conducted on November 20, 2024. The analyst assumed that reasonable assurance existed as of November 20, 2024, that the capability of the EDG A to perform its required safety function would not have been affected by the degraded condition that was subsequently revealed during the December 11, 2024, testing. The analyst assumed that a “T/2” exposure time for the degraded condition would be applicable for the total “T” time of 21 days from November 20 to December 11, 2024. The analyst determined that a total exposure time of 10.5 days (i.e., a “T/2” period) plus 13 days of repair time would be applicable for this analysis.

INFLUENTIAL ASSUMPTIONS

- The analyst assumed that the performance deficiency resulted in the EDG A being in a condition where it would have failed to load and run for its required mission time during an applicable design basis event during the 10.5-day “T/2” exposure time discussed above, with applicable impacts on common cause failure.
- For station blackout (SBO) sequences involving this EDG A failure condition, the analyst assumed that the safety function of the EDG A would not be recoverable.
- The analyst assumed that the use of Diverse and Flexible Coping (FLEX) strategies SBO events should be credited.

MODELING APPROACH

The Waterford SPAR Model version TLU11 (based on version 8.81) along with SAPHIRE software version 8.2.11 were used for the evaluation. Using the events and conditions assessment (ECA) workspace, the analyst determined that the risk impact of degraded condition described above would be most appropriately modeled by setting the basic event EPS-DGN-LR-DG3A (Diesel Generator 3A-S Fails to Load Run) to TRUE for the 10.5-day “T/2” exposure time discussed above and by setting the basic event EPS-DGN-TM-DG3A (DG 3A-S Unavailable Due to Test and Maintenance) to TRUE for the actual 13-day outage period involving troubleshooting and corrective maintenance. To credit the use of FLEX during both time periods, 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 cases.

The analyst also modified the fault trees EPS-DG3AS (Failure of Diesel Generator 3A-S) and EPS-DG3BS (Failure of Diesel Generator 3B-S) to include an additional baseline common cause failure event EPS-DGN-CF-LR (Common Cause Failure of Diesel Generators to Load Run). Given the troubleshooting and corrective maintenance involved with restoring the EDG A to a functional condition following this failure, the analyst determined that no recovery credit would be applicable to model for the case of a design basis event.

Based on a review of licensee operating procedures that do not allow the station’s Permanent Temporary Emergency Diesel (PTED) to be in a non-functional condition for maintenance or testing purposes concurrently with either the EDGs A or B, the analyst also added event tree post-processing rules to identify and eliminate cutset results containing mutually exclusive testing/maintenance conditions consisting of the basic event EPS-DGN-TM-TEDG (Temporary Emergency Diesel Generator Unavailable Due To Test and Maintenance) together with either EPS-DGN-TM-DG3A or EPS-DGN-TM-DG3B.

The analyst noted that the basic events EPS-DGN-FS-TEDG (Temporary Emergency Diesel Generator Fails to Start), EPS-DGN-FR-TEDG (Temporary Emergency Diesel Generator Fails to Run), and EPS-DGN-LR-TEDG (Temporary Emergency Diesel Generator Fails to Load Run) included failure data consistent with EDG failure data from Table 61 of Reference 1. The analyst determined that the modeled PTED failure probabilities should more appropriately be consistent with failure data from Table 69 for SBO generators. The analyst implemented the applicable updates to the failure probabilities reflected in the PTED failure basic events.

The analyst also noted that the basic events FLX-DGN-FS-DG1 (FLEX DG1 Fails to Start) and FLX-DGN-FR-DG1 (FLEX DG1 Fails to Run) included failure data consistent with Portable Diesel Generator failure data from Table 6-1 of Reference 2. The analyst considered the configuration of the “N” FLEX DG of being permanently installed in the reactor auxiliary building, as well as licensee maintenance and testing practices, and determined that the modeled “N” FLEX DG failure probabilities should more appropriately be consistent with failure data from Table 69 of Reference 1 for SBO generators. The analyst implemented the applicable updates to the failure probabilities reflected in these FLEX DG failure basic events.

INTERNAL EVENTS RESULTS

With the assumptions and modeling approach described above, the analyst obtained an increase in average annual core damage frequency (delta-CDF) result of 6.34E-7/year associated with internal events (including internal flooding events).

EXTERNAL EVENTS RESULTS

The increase in risk associated with external events was also evaluated. Using the same modeling approach as described above for internal events, the analyst evaluated the risk contribution from the external events modeled in SPAR, which included hurricane (HCN), tornado, high straightline winds (HWD), and seismic events. The analyst obtained a total result of 7.79E-7/year for delta-CDF associated with these external events. The analyst noted that 95 percent of this total external event risk contribution was coming from HCN events (4.23E-7, or 54 percent) and HWD events (3.21E-7, or 41 percent). External event risk contributions are further addressed in the qualitative considerations and sensitivity analyses sections below.

Table 1.

External Risk Category	Delta-CDF (/yr)	%
Hurricane Events	4.23E-7	54%
HWD Events	3.21E-7	41%
Tornado + Seismic Events	3.53E-8	5%
External Event Total	7.79E-7	100%

Since the SPAR model did not include modeling of fire events, the analyst determined that best available information associated with the risk attributable to fire events would be obtained from the analyst's review of the licensee's fire PRA model results for this condition. Based on this review, as further discussed below in the Licensee Results section, the analyst concluded that the increase in risk for this condition associated with fire events was best estimated to be a delta-CDF of 6.17E-7/year.

DOMINANT SEQUENCES

Dominant sequences contributing to the delta-CDF results for internal events involved LOOP events with failure of all EDGs (i.e., SBO events) and failures to recover either offsite electrical power (OEP) or EDG function within 2 hours (i.e., extended loss of AC power (ELAP) events) and either: 1) FLEX strategy failures, with and without failures to recover the OEP function within 24 hours, or 2) failure of the emergency feedwater function. For external events, hurricane and high wind sequences resulted in additional LOOP frequency being added to the same dominant sequence types noted above.

LARGE EARLY RELEASE FREQUENCY

The significance of the impact of the finding on large early release frequency (LERF) was also evaluated. The analyst evaluated the increase in LERF using Inspection Manual Chapter (IMC) 0609, Appendix H, "Containment Integrity Significance Determination Process." The finding was treated as a Type A finding because it could influence the likelihood of accidents leading to core damage as well as being a contributor to LERF. Additionally, the finding was evaluated for potential increase in the likelihood of a consequential steam generator tube rupture (C-SGTR). The analyst reviewed all sequences contributing to delta-CDF for any elements affecting LERF. The analyst identified sequences in which the emergency feedwater function failed as representing additional core damage sequences (i.e., in addition to the plant's corresponding baseline CDF risk) that involve "High-Dry-Low" (HDL) conditions, which have the potential to result in a C-SGTR. The analyst assumed a C-SGTR conditional probability of 2E-1 would be applicable for this category of sequences, consistent with guidance from NUREG-2195. Application of an assumed LERF factor of 1.0 per IMC 0609 Appendix H screening guidance would result in delta-LERF results of 3.65E-8/year for internal events and 2.08E-8/year for external events (not including HWD events or fire events). The analyst assumed that 1E-1[1] would be a more appropriate screening value for an average LERF factor that would be applicable for this category of sequences involving a SGTR condition. Application of this factor to the total delta-CDF associated with the applicable sequences described above yielded delta-LERF results of 3.65E-9/year for internal events and 2.08E-9/year for external events (not including HWD events or fire events). The analyst determined that best available information regarding LERF risk from fire events and for internal events would be obtained from a review of the licensee's LERF modeling results, as discussed below in the Licensee Results section. Overall, the analyst determined that risk attributable to LERF was not a dominant metric in the significance determination for this finding.

QUALITATIVE CONSIDERATIONS

The analyst reviewed licensee operations procedure OP-901-521, "Severe Weather and Flooding," Revision 344, which contains guidance to commence a plant shutdown to Mode 3 between 16 and 12 hours prior to the projected arrival of hurricane conditions onsite. The analyst noted that this procedure further directs that, following completion of the plant shutdown and at the discretion of plant management, a plant cooldown to Mode 4 with shutdown cooling in service may also be performed. The analyst noted that SPAR at-power PRA modeling applies for plant conditions in Modes 1, 2, 3, and 4, except for when reactor coolant system temperature conditions allow for entry to shutdown cooling (SDC) in Mode 4. The analyst noted that a recent example of a hurricane response occurred in August 2021 with Hurricane Ida making landfall near the plant's location. In that event, Waterford 3 achieved shutdown conditions but experienced a LOOP event prior to entry into SDC mode. The analyst considered that risk associated with dominant sequences in this case (i.e., LOOP) would likely be somewhat lower in Mode 3 versus full power operations. To the extent that the licensee may successfully achieve Mode 4 conditions with SDC in service prior to the onset of a hurricane event, that corresponding fraction of the LOOP consequences attributable to the event could be evaluated with a shutdown risk PRA model in lieu of at-power risk. The analyst noted that shutdown risk in this scenario, particularly the risk associated with loss of power to support SDC functions, could still represent substantial risk, especially in an early period of higher decay heat load following a transition to Mode 4 and SDC conditions. A sensitivity analysis is explored below to consider the portion of total risk that is attributable to HCN-induced events for this condition.

The analyst also considered the basis for the current initiating event frequency (IEF) modeling for the LOOP weather-related (LOOPWR) category of LOOP initiating events in the SPAR internal events model, which includes subcategories of a) extreme weather events such as events involving high wind conditions, and b) severe weather events which includes other types of weather events. The analyst noted that LOOPWR sequences represented a significant

---

[1] A basis for the selection of this LERF factor screening value is further discussed in Enclosure 3 of NRC Inspection Report 05000382/2024013 (ADAMS ML24228A261).

portion of internal event dominant sequences for this evaluation. The analyst also noted that external event sequences of HWD and HCN also constituted a significant portion of dominant sequences for this evaluation. The HWD and HCN sequences effectively contributed additional risk due to a LOOPWR initiating event because of applying conditional LOOPWR probabilities to the associated IEFs for HWD and HCN events. The analyst determined that this combination of all currently modeled initiating events potentially overestimates the frequency of high wind-related extreme weather-induced LOOP events. After consulting with staff from Idaho National Laboratory, the analyst determined that an applicable interim adjustment to offset this potential double-counting concern would be to exclude the currently modeled HWD external event contribution from the total risk estimate. This approach is reflected in a sensitivity analysis discussed below and is also reflected in the best estimate results for this evaluation.

SENSITIVITY ANALYSES

Sensitivity #1:
As discussed above, the analyst determined that the risk contribution from the HWD external event likely represents a double-counting concern relative to a portion of the events that are reflected currently modeled IEF for LOOPWR internal events. The analyst determined that the exclusion of HWD events from the external event modeling reduced the total external event risk result for delta-CDF from 7.79E-7/year to 4.58E-7/year.

Sensitivity #2:
Additionally, pursuant to the qualitative consideration discussed above regarding the potential risk increase due to hurricane HCN events, the analyst determined that the total risk contribution from HCN events for this evaluation was a delta-CDF of 4.23E-7/year, which represents 92 percent of the revised total external event risk after exclusion of HWD events. As discussed above, the analyst considered that some fraction of this risk contribution could more appropriately be evaluated as shutdown risk versus at-power risk, corresponding to the likelihood that the station may achieve shutdown conditions with decay heat removal being provided by the shutdown cooling function prior to being subject to the impacts of an HCN event. The analyst determined that a corresponding potential reduction in total risk attributable to HCN events would not, by itself, have the potential to reduce the total risk for this evaluation below the White significance threshold.

Sensitivity #3:
The analyst noted that credit for offsite power recovery (OPR) in the event of an SBO is provided if accomplished within the first 2 hours of the event, prior to a transition to ELAP/FLEX and SBO coping strategies. Based on a review of the licensee's SBO, ELAP, and FLEX implementation procedures, the analyst determined that the opportunity for OPR in this scenario could potentially exist for up to 12 hours (contingent on the probability of timeliness and success of deep load shedding actions for one selected train as well as no other associated failures precluding this potential extended DC power availability for the selected train). The analyst evaluated the potential decrease in risk associated with a bounding case involving decreasing the following 2-hour OPR failure probabilities to reflect 12-hour failure probabilities:

OEP-XHE-XL-NR02H (Operator Fails to Recover Offsite Power in 2 Hours)
OEP-XHE-XL-NR02HGR (Operator Fails to Recover Offsite Power in 2 Hours, Grid-Related)
OEP-XHE-XL-NR02HPC (Operator Fails to Recover Offsite Power in 2 Hours, Plant-Centered)
OEP-XHE-XL-NR02HSC (Operator Fails to Recover Offsite Power in 2 Hours, Swyd-Centered)
OEP-XHE-XL-NR02HWR (Operator Fails to Recover Offsite Power in 2 Hours, Weather-related)

Table 2.

	Delta-CDF Results (/yr)	
Risk Category	**Before**	**With OPR Adjustment**
Internal Events	6.34E-07	4.03E-07
HCN Events	4.23E-07	4.23E-07
EQK and TOR Events	3.53E-08	3.53E-08
Totals	**1.09E-06**	**8.61E-07**

The analyst noted that the lack of impact of this OPR failure probability shifting on external event results is likely attributable to the fact that OPR failure probabilities modeled for the dominant LOOPWR sequences are relatively independent of time available to perform the action, for the applicable time intervals being considered in this analysis.

The analyst determined that a potential reduction in total risk attributable to this consideration would not, by itself, have the potential to reduce the total risk for this evaluation below the White significance threshold. Of the maximum possible reduction in delta-CDF results for internal events of 2.31E-7/yr attributable to this consideration as shown in Table 2 above, the analyst qualitatively assumed that a reduction of 1.5E-7/yr would be included in the best estimate results for this analysis, as reflected in the results cited below in the Conclusion section. This assumption was based on a qualitative assessment of the factors referenced above. Specifically, the time period available to implement OPR, relative to the maximum potential available time of 12 hours, would be reflective of the degree to which all associated actions involving ELAP declaration and SBO coping (e.g., deep load shedding) would be both timely and successful, as well as a dependency on no other failures or complications occurring affecting the extended availability of the selected train of DC power.

LICENSEE RESULTS

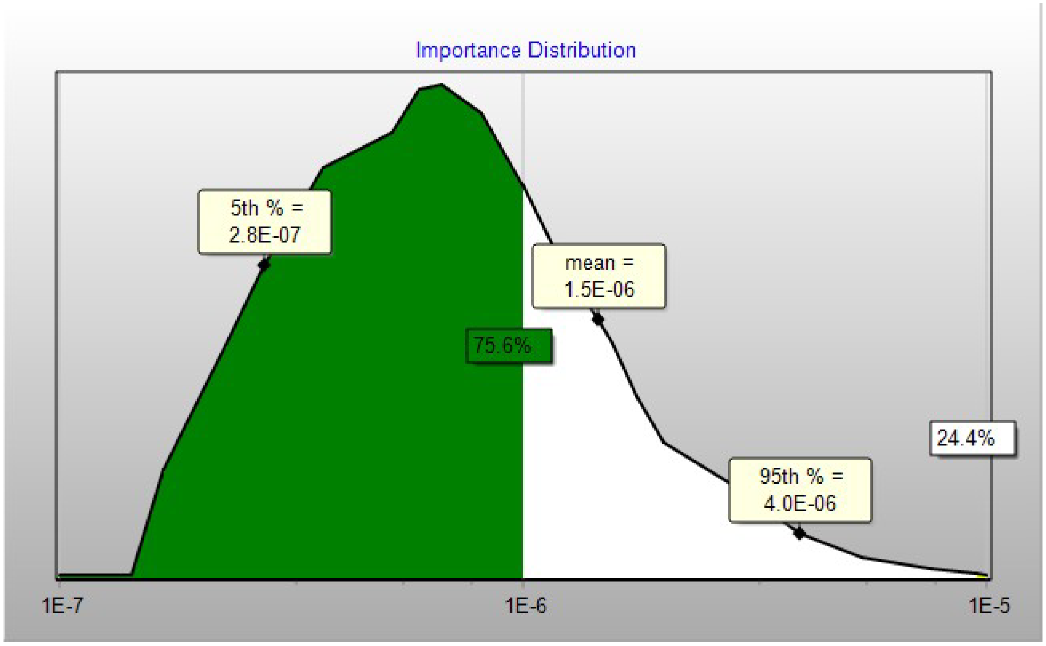
The analyst reviewed risk assessment results provided by the licensee from the use of the licensee's PRA model for the categories of internal events (internal flood) and fire events. These results, summarized in Table 3 below, reflected a failure-to-start condition for the A EDG for an exposure time of 10.5 days (with applicable common cause failure implications) together with a test/maintenance condition for the A EDG for an exposure time of 13 days. The analyst determined that the fire risk results obtained from the licensee's PRA model represented best available information for use in this analysis.

Table 3.

Risk category	Delta-CDF (/yr)	Delta-LERF (/yr)
Internal Events	8.23E-8	2.77E-10
Fire Events	6.17E-7	1.55E-9
Totals	6.99E-7	1.83E-9

UNCERTAINTY ANALYSIS

The analyst performed an uncertainty analysis using the Monte Carlo method with a sample size of 3,775 on the internal and external events results using the SAPHIRE ECA workspace.



5th %	Median	Point Estimate	Mean	95th %
2.76E-7	9.95E-7	1.45E-6	1.45E-6	3.95E-6

The above results reflect a simplified approach of combining the two exposure periods of 10.5 days and 13 days into one 24-day period of a failure-to-load-run condition for purposes of performing a single overall uncertainty estimation that reflects the approximate total risk estimate for this analysis. These results do not include the significant risk contribution from fire events (which was estimated separately based on review of licensee modeling results). The results represented above do include risk contribution from HWD events to serve as an approximate surrogate to offset the fire risk contribution that is not included in SPAR modeling, in an effort to estimate the uncertainty distribution associated with the approximate best-estimate total risk for this evaluation (it can be noted that the resulting point estimate reflected above is just slightly lower than the actual best-estimate total risk (delta-CDF) reported in Table 4 below).

CONCLUSION

The analyst concluded that the overall preliminary risk significance of the finding was determined to be low to moderate safety significance (White), based on best estimates of 1.56E-6/year for total delta-CDF and 3.91E-9/year for total delta-LERF, as detailed in the summary table below. The source of “best available information” for the best estimate results below for fire risk (delta-CDF and delta-LERF) and delta-LERF risk for internal events was determined to be the analyst’s review of the licensee’s modeling results. The results in the remaining risk categories are based on the analyst’s use of the NRC SPAR model.

Table 4.

Risk category	Delta-CDF (/yr)	Delta-LERF (/yr)
Internal Events	4.84E-7	2.77E-10
External Events minus HWD	4.58E-7	2.08E-9
Fire	6.17E-7	1.55E-9
Totals:	1.56E-6	3.91E-9

References:

1) INL/EXT-21-65055, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants: 2020 Update,” November 2021.

2) PWROG-18042-NP, “FLEX Equipment Data Collection and Analysis,” Revision 1, February 2022.