UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 3, 2025

Delson C. Erb
Vice President, OPS Support
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: TRANSMITTAL OF SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2, EMERGENCY DIESEL GENERATOR 1B ACCIDENT SEQUENCE PRECURSOR REPORT (EPID L-2019-PMP-0033)

Dear Delson Erb:

The details of the Sequoyah Nuclear Plant, Units 1 and 2, emergency diesel generator 1B failure are provided in Nuclear Regulatory Commission (NRC) inspection report (IR) 05000327/2024090, "Sequoyah, Units 1 and 2 – NRC Inspection Report 05000327/2024090 and 05000328/2024090 and Preliminary Greater-than-Green Finding and Apparent Violation" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24066A197). As part of the accident sequence precursor (ASP) program, the NRC staff reviewed the event to determine if the increased core damage probability of the degraded condition exceeded the precursor threshold.

The preliminary ASP report was transmitted to the licensee on November 22, 2024, and a 60-day review period was allowed. The license provided comments on January 16, 2025 (ML25017A071). These comments are summarized along with the Office of Nuclear Regulatory Research responses in Appendix B of the final ASP analysis report, which is provided in the enclosure to this letter. The final ASP report is provided in an enclosure to this letter.

If you have any questions, please contact me at Perry.Buckberg@nrc.gov or (301) 415-1383.

Sincerely,

/RA/

Perry Buckberg, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosure:
As stated

cc: Listserv

ENCLOSURE

Final ASP Analysis
Sequoyah Nuclear Plant, Units 1 and 2
Failure of Emergency Diesel Generator 1B During Testing
050003272024090 - Precursor

Final ASP Analysis – Precursor

Accident Sequence Precursor Program – Office of Nuclear Regulatory Research			
Sequoyah Nuclear Plant, Units 1 and 2	Failure of Emergency Diesel Generator ‘1B’ During Testing		
Event Date: 9/19/2023	**LER:** None **IR:** 05000327/2024090	**Unit 1 ΔCDP =** 1×10^{-5} **Unit 2 ΔCDP =** 5×10^{-6}	
Plant Type:	Westinghouse Four-Loop Pressurized-Water Reactor (PWR) with Wet, Ice Condenser Containment		
Plant Operating Mode (Reactor Power Level):	Mode 1 (100% Reactor Power)		
Analyst: Christopher Hunter	**Reviewer:** Jeffery Wood	**Completion Date:** 1/27/2025	

1 EXECUTIVE SUMMARY

On September 19, 2023, a scheduled 24-hour run of emergency diesel generator (EDG) ‘1B’ was being performed. Approximately 4.5 hours into the run, multiple main control room (MCR) and local annunciator alarms including high crankcase pressure followed by low jacket water level were received. Operators immediately shutdown EDG ‘1B’ in accordance with the annunciator response procedure.

A preliminary greater-than-*Green* (GTG) finding was identified in inspection report (IR) 05000327/2024090, “Sequoyah, Units 1 and 2 – NRC Inspection Report 05000327/2024090 and 05000328/2024090 and Preliminary Greater-than-Green Finding and Apparent Violation,” (ML24066A197). The failure of EDG ‘1B’ was determined to be the loss of retention of an exhaust valve on the #14 cylinder. The exhaust valve fell into the cylinder and caused sufficient cylinder wall damage to lead to a loss of jacket water cooling. A subsequent root cause evaluation performed by the licensee determined that a material flaw had developed at a fatigued location in the rear outboard valve bridge lash adjuster spring. This root cause evaluation, along with a separate material evaluation, led the licensee to conclude that the combination of the material characteristics of the exhaust valve spring seat coupled with the unusual loads developed by the fatigue failed lash adjuster spring resulted in failure of EDG ‘1B’. An independent NRC review of the licensee evaluations determined that there was no performance deficiency. However, the NRC concluded that there was a minor violation associated with the licensee’s failure to adequately establish and implement maintenance instructions and practices. This minor violation could not be directly attributed to the failure of the EDG ‘1B’. See IR 05000327/2024091, “Sequoyah, Units 1 and 2 – Final Significance Determination – NRC Inspection Report 05000327/2024091 and 05000328/2024091,” (ML24145A085) for additional information. No licensee event report (LER) was issued for this degraded condition.

A preliminary detailed risk evaluation was performed by the NRC.[1] In addition, the licensee completed a separate risk evaluation.[2] However, since it was determined that no licensee performance deficiency was associated with this degraded condition, these evaluations were not finalized. Therefore, an independent accident sequence precursor (ASP) analysis was performed.

The mean increase in core damage probability (ΔCDP) for this degraded condition is 1×10^{-5} and 5×10^{-6} for Units 1 and 2, respectively. Therefore, this degraded condition is precursor for both units. The dominant hazards for this ASP analysis are internal events (57 and 54 percent, Unit 1 and Unit 2, respectively), high winds (23 percent for both units), internal fires (19 and 20 percent, Unit 1 and Unit 2, respectively), and seismic hazards (2 and 4 percent, Unit 1 and Unit 2, respectively). The risk contribution from tornados and internal/external floods were negligible for both units.[3]

2 EVENT DETAILS

2.1 Event Description

On September 19, 2023, a scheduled 24-hour run of EDG '1B' was being performed. Approximately 4.5 hours into the run, multiple MCR and local annunciator alarms including high crankcase pressure followed by low jacket water level were received. Operators immediately shutdown EDG '1B' in accordance with the annunciator response procedure. The licensee completed repairs and restored EDG '1B' operability after successful post-maintenance testing on September 29, 2023. Additional information is provided in IRs 05000327/2024090 and 05000327/2024091. No LER was issued for this degraded condition.

2.2 Cause

The failure of EDG '1B' was determined to be the loss of retention of an exhaust valve on the #14 cylinder. The exhaust valve fell into the cylinder and caused sufficient cylinder wall damage to lead to a loss of jacket water cooling. A licensee root cause evaluation determined that a material flaw had developed at a fatigued location in the rear outboard valve bridge lash adjuster spring. This root cause evaluation, along with a separate material evaluation, resulted in the licensee concluding that the combination of the material characteristics of the exhaust valve spring seat coupled with the unusual loads developed by the fatigue failed lash adjuster spring resulted in failure of EDG '1B'.

3 MODELING

3.1 SDP Results/Basis for ASP Analysis

The Accident Sequencer Precursor (ASP) Program uses SDP results for degraded conditions when available (and applicable). A preliminary GTG finding was identified in IR 05000327/2024090. However, the NRC later determined that there was no licensee

1 The NRC's detailed risk evaluation associated with the preliminary GTG finding calculated increases in core damage frequency (CDF) of 2.7×10^{-5} (Unit 1) and 2.1×10^{-5} (Unit 2) per year.

2 The licensee's risk evaluation calculated increases in CDF in the 10^{-6} range and 10^{-7} range per year for Units 1 and 2, respectively.

3 These hazards contributed less than one percent of the total ΔCDP for each unit.

performance deficiency (see IR 05000327/2024091 for additional information) and, therefore, no finalized detailed risk evaluation was performed for this degraded condition. Therefore, an independent ASP analysis was performed to determine the risk significance associated with the failure of EDG '1B'. A search of Sequoyah Nuclear Plant LERs did not reveal any "windowed" events.

3.2 Analysis Type

A degraded condition analysis was performed using a test and limited use revision of the version 8.82 SPAR model for Sequoyah Nuclear Plant created in May 2024. Note that this model is a Unit 1 model that is being used to evaluate the risk of this degraded condition for both units individually. This SPAR model was revised based on the review of the licensee's final integrated plant for the FLEX mitigation strategies. The following key FLEX modeling changes were included:

- The credit for the FLEX mitigation strategies was activated for SBO scenario for which an extended loss of AC power (ELAP) is declared.
- The requirement for operators to perform a deep load shed of the direct current (DC) buses to extend the safety-related batteries depletion time during a postulated SBO to allow for the implementation of the FLEX mitigation strategies was added to the SBO-ELAP event tree.
- Credit for offsite power recovery was limited to sequence-specific depletion time of the safety-related batteries, core uncovery, or 24 hours, whichever occurs first.
- Because of the large uncertainty in modeling assumptions related to availability and reliability of components and strategies for mission times that are well beyond 24 hours and the unclear basis for requiring AC power recovery within 72 hours, the 72-hour AC power requirement was eliminated from the SBO-ELAP event tree.
- Credit is provided for the plant to reach a safe and stable end state without offsite power recovery if either a turbine-driven auxiliary feedwater (AFW) pump or the FLEX steam generator (SG) makeup maintains adequate SG inventory and reactor coolant system (RCS) makeup is initiated prior to the onset of reflux cooling.[4]
- Credit for EDG repair was limited to the ELAP declaration time (i.e., 1 hour) because (a.) the operators' focus would switch from EDG troubleshooting activities to implementing the FLEX mitigation strategies and (b.) the DC load shedding activities could preclude recovery of EDGs.

This SPAR model includes the following hazards:

- Internal events,
- Internal and external floods,
- Internal fires

[4] There is potential to prevent core damage if RCS makeup is restored after the onset of reflux cooling and therefore, this modeling assumption is potentially conservative. A review of the Unit 1 analysis results reveals the accident sequences that are affected by this assumption account for approximately 7 percent of the ΔCDP. Similar results would be expected for Unit 2 because of the similar accident sequences and cut sets. Therefore, any potential changes would have a minimum effect on the analysis results and, therefore, was not evaluated further.

- Seismic events, and
- High winds (including tornados).

3.3 SPAR Model Modifications

The following additional SPAR model modifications were made to support this analysis:

- *Use of Common-Cause Failure (CCF) Parameters Derived from Component-Specific Priors*. The existing CCF parameters used in the SPAR models are derived using a single generic prior that includes CCF events from all components, failure modes, and causes. Since different component types and different failure causes could lead to drastically different alpha factors, the above usage of a single generic prior could result in significant uncertainty in the CCF parameter estimates. Given the uncertainties associated with using a single generic CCF prior in estimating the existing CCF parameters, prior distributions were derived for five component categories—(a.) pumps, (b.) valves, (c.) strainers, (d.) EDGs, and (e.) other equipment (e.g., transformers, breakers, fans, heat exchangers, etc.). A new set alpha factors for each applicable component and failure mode type are then calculated via a Bayesian update using the latest 15-year period of CCF data.[5] The existing alpha factors for the EDG CCF to run basic events (EPS-DGN-CF-FTR, EPS-DGN-CF-FRU1, and EPS-DGN-CF-FRU2) were substituted using those derived from the EDG component-specific prior in the base Sequoyah SPAR model as shown in the following table:

Table 1. Applicable CCF Parameters Derived from EDG Component-Specific Prior

CCF Template Event	CCCG	Alpha Factor	2020 CCF Parameters Generic Prior			2020 CCF Parameters Component-Specific Priors			%Δ
			a	b	Mean	a	b	Mean	Mean
EPS-EDG-FR-02A01	2	α_1	1.33E+02	1.85E+00	9.86E-01	1.71E+02	1.87E+00	9.89E-01	0.1%
EPS-EDG-FR-02A02	2	α_2	1.85E+00	1.33E+02	1.38E-02	1.87E+00	1.71E+02	1.08E-02	-10.8%
EPS-EDG-FR-04A01	4	α_1	3.09E+02	4.12E+00	9.87E-01	4.80E+02	4.05E+00	9.92E-01	0.5%
EPS-EDG-FR-04A02	4	α_2	2.21E+00	3.11E+02	7.06E-03	2.01E+00	4.82E+02	4.16E-03	-41.1%
EPS-EDG-FR-04A03	4	α_3	1.42E+00	3.12E+02	4.55E-03	1.57E+00	4.82E+02	3.25E-03	-28.6%
EPS-EDG-FR-04A04	4	α_4	4.82E-01	3.13E+02	1.54E-03	4.62E-01	4.83E+02	9.55E-04	-38.0%

  A review of the analysis results indicated that changes to additional CCF parameters to other components and/or failure modes would result in minimal changes to the ΔCDP for the degraded condition for both units and, therefore, no additional modifications were made.

- *Treatment of Potential Cross-Unit CCF of the EDGs*. The base SPAR model has three EDG common cause component groups (CCCGs) per failure mode (i.e., failure to run and failure to start).[6] One of these CCCGs include EDGs across units. A review of the

[5] These revised alpha factors (including comparisons with the existing CCF parameters used in the SPAR model) are provided on the publicly available Reactor Operational Experience Results and Databases Webpage.

[6] The use of multiple CCCGs can result in the overestimation of the impact of potential CCFs due to double counting.

existing CCF data for CCF events of similar components that have occurred relatively close in time to each other at multi-unit sites reveals a substantially smaller number of cross-unit CCF events (as compared to total number of CCF events).[7] And although the data review has some limitations, it is clear the CCF coupling of similar components across units is weaker than the CCF coupling of redundant components within the same system. Therefore, the use of the existing alpha factors across units is unlikely to provide best estimate of the potential for cross-unit CCF in most cases. To provide a better estimate of the potential for CCF between the Unit 1 and Unit 2 EDGs, the CCF parameters (as shown in Table 1) were multiplied by a factor of 0.1.[8] Note that these changes also reduce the double-counting effects. Although these changes are judged to be best estimate given the current CCF data and modeling, they are identified as key uncertainty associated with this analysis, which is discussed further in Section 4.4. The NRC is planning an additional review and research activities of the existing CCF data to determine an updated approach to the treatment of cross-unit CCF in the SPAR models.

3.4 Exposure Time

EDG '1B' successfully passed its previous surveillance tests prior to the failure on September 19, 2023. However, the nature of the failure mechanism makes it likely that EDG '1B' would not have been able to fulfill its safety function for its probabilistic risk assessment (PRA) mission time of 24 hours for some time. Therefore, the run history for EDG '1B' was used to estimate the exposure period (see the following table). Based on the run history, it has been determined that EDG '1B' was unable to fulfill its safety function from January 25th until September 29th, a period of 246 days. This exposure time estimate is identified as key uncertainty associated with this analysis, which is discussed further in Section 4.4.

Table 2. EDG '1B' Run History

Date	Description	Approximate Run Duration	Cumulative Run Time
9/29/2023	EDG '1B' is repaired; operability is restored	—	—
9/19/2023	Failed 24-hour endurance test	4 hours	4 hours
8/23/2023	Successful surveillance test	2 hours	6 hours
7/25/2023	Successful surveillance test	2 hours	8 hours
6/20/2023	Successful surveillance test	2 hours	10 hours
5/22/2023	Successful surveillance test	2 hours	12 hours
4/24/2023	Successful surveillance test	2 hours	14 hours
3/17/2023	Successful surveillance test	2 hours	16 hours
2/21/2023	Successful surveillance test	2 hours	18 hours
1/25/2023	Successful surveillance test	6 hours	24 hours

7 The existing CCF database has a total of 372 CCF events that occurred during 1989–2021 period. An initial review identified 10 CCF events for similar components across units that occurred close in time.

8 A review of the CCF data observed that the number of cross-unit CCF events were less than redundant CCF events in the same system for the applicable components (e.g., circulating water/service water strainers, EDGs, battery chargers, service water pumps, boiling water reactor safety relief valves, and safety injection pumps) by factors ranging from approximately 0.05 to 0.1. The selection of 0.1 factor reduction for the cross-unit EDG CCF to run is judged to be a more reasonable estimation of the potential for cross-unit CCF at this time.

3.5 Analysis Assumptions

The following modeling assumptions were determined to be significant for this analysis:

Unit 1

- Basic event EPS-DGN-FR-1B (*diesel generator 1B fails to run*) was set to TRUE because EDG '1B' failed during testing on September 19th.

Unit 2

- Basic event EPS-DGN-FR-2B (*diesel generator 2B fails to run*) was set to TRUE to act a surrogate event for the EDG '1B' failure that occurred during testing on September 19th. Because the SPAR model is a Unit 1 model, this is equivalent to setting an opposite unit EDG to failed to represent the impact of a failed Unit 1 EDG on Unit 2.

4 ANALYSIS RESULTS

4.1 Results

The mean ΔCDP for this analysis is calculated to be 1.2×10^{-5} and 4.5×10^{-6} for Units 1 and 2, respectively. The ASP Program threshold is 1×10^{-6} for degraded conditions; therefore, this event is a precursor for both units. The parameter uncertainty results are provided in the tables below:

Table 3. Parameter Uncertainty Results for Unit 1

5%	Median	Pt. Estimate	Mean	95%
2.0×10^{-6}	8.4×10^{-6}	1.1×10^{-5}	1.2×10^{-5}	3.5×10^{-5}

Table 4. Parameter Uncertainty Results for Unit 2

5%	Median	Pt. Estimate	Mean	95%
7.4×10^{-7}	3.0×10^{-6}	4.0×10^{-6}	4.5×10^{-6}	1.3×10^{-5}

4.2 Dominant Hazards[9]

The following table provides the breakdown of the risk impact for this degraded condition from the hazards that are included in the Sequoyah SPAR model.

Table 5. Hazard Breakdown

Hazard	Unit 1		Unit 2	
	ΔCDP	%	ΔCDP	%
Internal Events	6.0E-06	56.8%	2.1E-06	53.6%
Internal Fires	2.0E-06	18.7%	8.0E-07	20.0%
High Winds	2.4E-06	22.5%	9.0E-07	22.6%
Seismic	2.0E-07	1.9%	1.5E-07	3.8%
Tornados	0.00E+00	0.0%	0.00E+00	0.0%
Internal Flood	8.3E-09	0.1%	2.5E-09	0.1%
External Flood	5.5E-09	0.1%	2.1E-09	0.1%

9 The ΔCDPs presented in Sections 4.2, 4.3, and 4.4 are point estimates.

Total	1.1E-05		4.0E-06	

4.3 Dominant Sequences

Unit 1

The dominant accident sequence is weather-related LOOP sequence 17-06 (ΔCDP = 1.1×10^{-6}), which contributes approximately 10 percent of the total ΔCDP. The sequences that contribute at least 5 percent to the total ΔCDP are provided in the following table. The event tree with the dominant sequence is shown graphically in Figures A-1 and A-2 of Appendix A.

Table 6. Unit 1 Dominant Sequences

Sequence	ΔCDP	%	Description
LOOPWR 17-06	1.1E-06	10.4%	Weather-related LOOP initiating event occurs; emergency power system failure results in SBO; AFW is successful; RCP seal LOCA occurs; and operators fail to restore AC power within 4 hours results in core damage.
LOOPWR 15	8.7E-07	8.3%	Weather-related LOOP initiating event occurs; emergency power system is successful; AFW fails; and feed and bleed cooling fails resulting in core damage.
HWD-NON-TOR 2-17-12	7.4E-07	7.0%	High winds result in a LOOP initiating event; emergency power system failure results in SBO; AFW fails; and operators fail to restore AC power within 1 hour results in core damage.
HWD-NON-TOR 2-17-06	6.5E-07	6.2%	High winds result in a LOOP initiating event; emergency power system failure results in SBO; AFW is successful; RCP seal LOCA occurs; and operators fail to restore AC power within 4 hours results in core damage.
LOOPWR 17-12	6.2E-07	5.9%	Weather-related LOOP initiating event occurs; emergency power system failure results in SBO; AFW fails; and operators fail to restore AC power within 1 hour results in core damage.
LOOPGR 15	5.3E-07	5.0%	Grid-related LOOP initiating event occurs; emergency power system is successful; AFW fails; and feed and bleed cooling fails resulting in core damage.

Unit 2

The dominant accident sequence is weather-related LOOP sequence 17-06 (ΔCDP = 4.4×10^{-7}), which contributes approximately 11 percent of the total ΔCDP. The sequences that contribute at least 5 percent to the total ΔCDP are provided in the following table. The event tree with the dominant sequence is shown graphically in Figures A-1 and A-2 of Appendix A.

Table 7. Unit 2 Dominant Sequences

Sequence	ΔCDP	%	Description
LOOPWR 17-06	4.4E-07	11.1%	Weather-related LOOP initiating event occurs; emergency power system failure results in SBO; AFW is successful; RCP seal LOCA occurs; and operators fail to restore AC power within 4 hours results in core damage.
HWD-NON-TOR 2-17-12	2.9E-07	7.4%	High winds result in a LOOP initiating event; emergency power system failure results in SBO; AFW fails; and operators fail to restore AC power within 1 hour results in core damage.

Sequence	ΔCDP	%	Description
HWD-NON-TOR 2-17-06	2.7E-07	6.9%	High winds result in a LOOP initiating event; emergency power system failure results in SBO; AFW is successful; RCP seal LOCA occurs; and operators fail to restore AC power within 4 hours results in core damage.
LOOPWR 17-12	2.5E-07	6.3%	Weather-related LOOP initiating event occurs; emergency power system failure results in SBO; AFW fails; and operators fail to restore AC power within 1 hour results in core damage.
LOOPWR 15	2.3E-07	5.8%	Weather-related LOOP initiating event occurs; emergency power system is successful; AFW fails; and feed and bleed cooling fails resulting in core damage.

4.4 Key Uncertainties

A review of the analysis assumptions and results reveal the following key uncertainties:

- *Internal Fire Modeling*. The internal fire modeling in the Sequoyah SPAR model is largely based on the results of the individual plant examination for external events (IPEEE) that was completed in 1995, which is a key uncertainty for this analysis. The evaluation of internal fires in the IPEEE studies are known to have significant conservatisms. However, some assumptions and techniques used in the IPEEE could have led to the underestimating of some fire scenarios. A review of the internal fire results for this analysis revealed an issue with one of the dominant fire scenarios. Specifically, the scenario involving fires in the auxiliary control room (FRI-AUX-LOOP1) could be underestimated because the existing modeling assumes this event to only results in a plant-centered LOOP. However, a review of the information provided in the IPEEE shows that this scenario enveloped fires from several zones, some of which include failures of key equipment (e.g., turbine-driven AFW pump, safety-related electrical buses, etc.). Therefore, the risk impact from this scenario could be underestimated. The revision of this fire modeling will take a significant level of effort, which is beyond the scope of this analysis. Given any potential changes to the fire modeling is unlikely to result in this degraded condition to exceed the significant precursor threshold (i.e., ΔCDP greater than or equal to 10^{-3}), this issue was not investigated further. The modeling of internal fires in the Sequoyah SPAR model will be updated based on information from the licensee's fire probabilistic risk assessment (PRA) model in the future.

- *Exposure Time*. The exposure time that EDG '1B' was assumed to be unable to fulfil the PRA mission time of 24 hours was estimated based on the past testing run time. However, there are uncertainties associated with this estimation because this assumes that the failure mechanism behaved linearly during the entire exposure time. In addition, testing conditions are not identical to the conditions experienced by the EDG during a demand (e.g., different loading). If the exposure time was the maximum of 1 year, the ΔCDP associated with this degraded condition would remain in the same range and with no additional insights. The exposure time would need to be less than 30 days for Unit 1 and less than 60 days for the Unit 2 for the ΔCDP to decrease below the precursor threshold.

- *CCF Modeling Assumptions*. As stated in Section 3.3, the use of modified alpha factors across units (or systems) is subject to significant uncertainty. For the purposes of this analysis, sensitivity analyses were performed to bound the effects of the cross-unit CCF or the EDGs by (a.) eliminating the cross-unit CCF completely and (b) using the full alpha factors derived from the EDG component-specific prior for the cross-unit CCF. The elimination of the cross-unit CCF of the EDGs results in a lower ΔCDP 9.9×10^{-6} (6-percent decrease) and 7.2×10^{-6} (15-percent decrease) for Units 1 and 2, respectively. The use of the full alpha factors for the cross-unit CCF of the EDGs results in an upper ΔCDP 1.6×10^{-5} (51-percent increase) and 9.3×10^{-6} (134-percent increase) for Units 1 and 2, respectively.

- *No Credit for EDG '1B' Run Time*. Depending on when a postulated demand of EDG '1B' during the 246-day exposure time would occur, EDG '1B' may have run for some portion of the PRA mission time of 24 hours. However, ASP analyses use the "failure memory" approach in which successful operation of equipment is not credited. Past NRC condition assessment have sometimes credited additional time for AC power recovery given the run history of failed EDG in some cases (either in the best estimate or as a sensitivity analysis). The inclusion of this credit has resulted in a range of ΔCDP reductions (from minimal to up to 40 percent decrease).

Appendix A: Key Event Trees

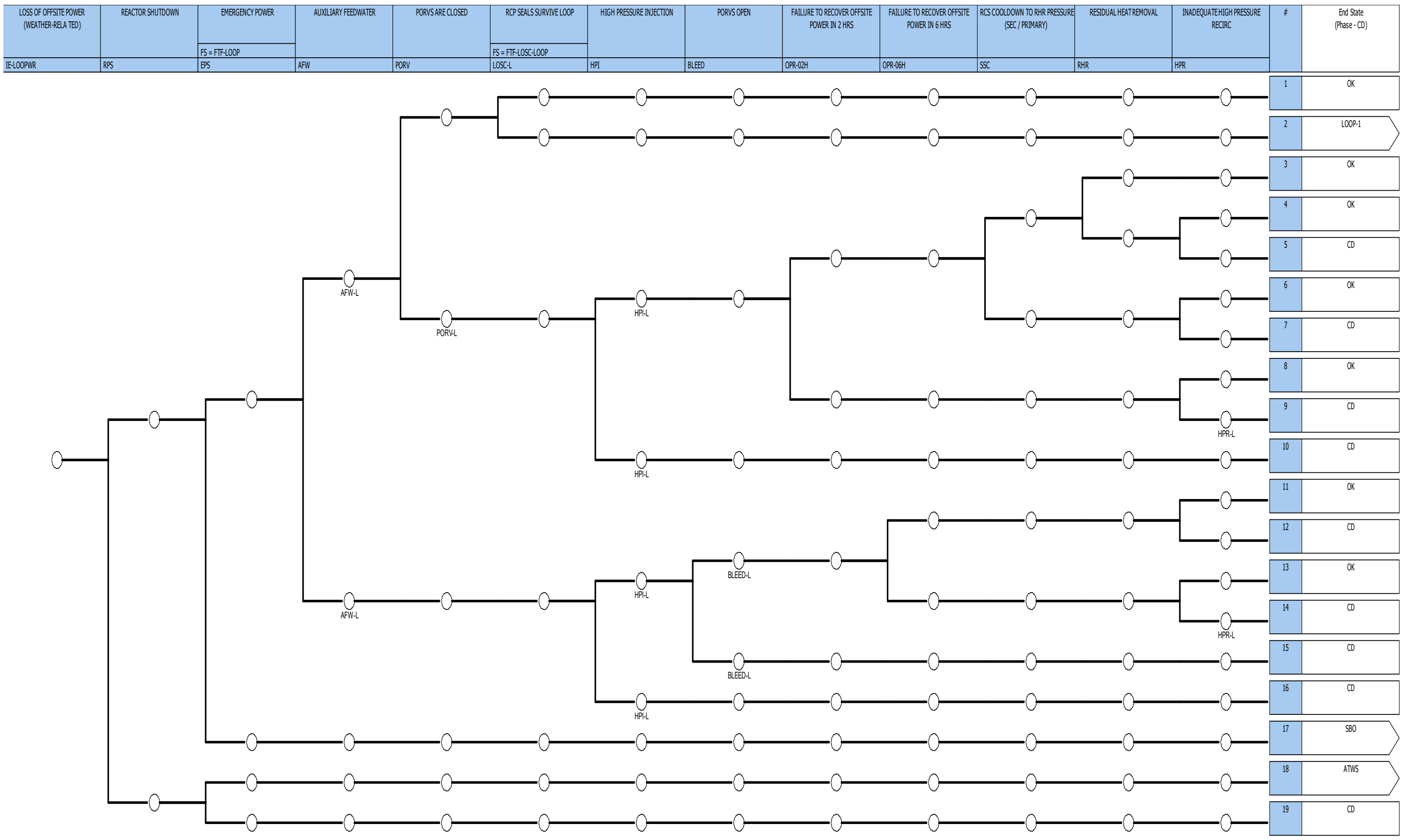

Figure A-1. Weather-Related LOOP Event Tree

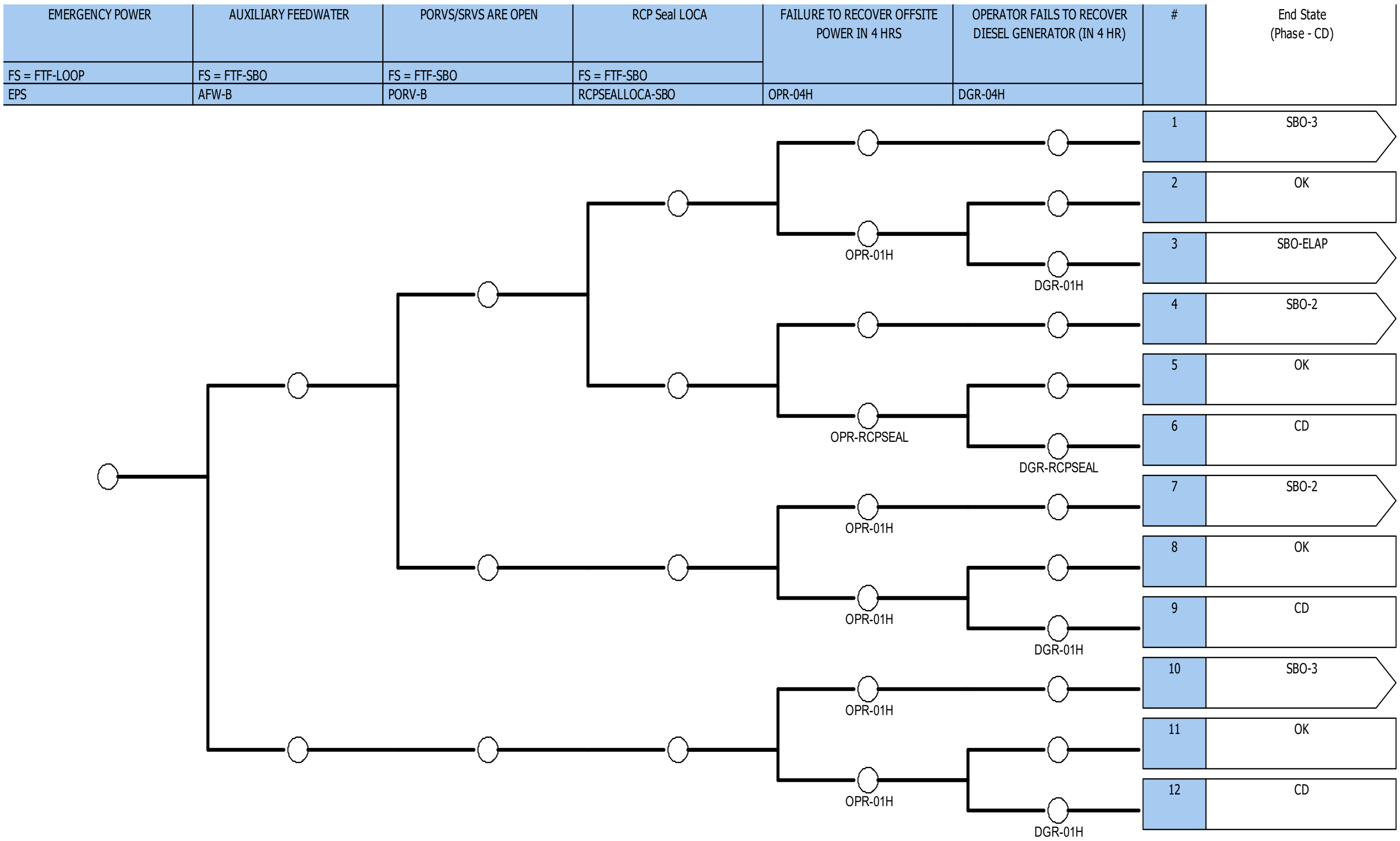

Figure A-2. SBO Event Tree

Appendix B: NRC Response to Licensee Comments

The NRC provided the licensee, Tennessee Valley Authority (TVA), the preliminary ASP analysis for an opportunity to provide any feedback before the analysis was finalized. TVA provided comments on the preliminary ASP analysis on January 16, 2025 (ML25017A071). The Office of Nuclear Regulatory Research response is provided for the comments below.

- **Comment 1 –** The Executive Summary and Section 2 of the analysis report was revised with the updated information provided in IR 05000327/2024091 associated with the failure of EDG '1B'.
- **Comment 2 –** The Executive Summary was revised to state that the licensee performed a risk evaluation and provided this risk evaluation to NRC Region 2 staff.
- **Comment 3 –** See response to Comment 2. In addition, the language was clarified that the NRC's detailed risk evaluation to support the SDP was not finalized.
- **Comment 4 –** The Sequoyah SPAR model used for this ASP analysis includes credit for the 480-volt FLEX diesel generators. However, credit is only provided for one of these diesel generators because it is believed insufficient time exists for operators to connect and start the other diesel generator prior to battery depletion should the initial diesel generator fail. A sensitivity analysis crediting both 480-volt FLEX diesel generators (in addition to one 6.9-kilovolt FLEX diesel generator) results in a decrease in ΔCDP of approximately 5 percent. Similar results would be expected for Unit 2 because of the similar accident sequences and cut sets. Given the potential small changes to the risk impact of the failure EDG '1B' with no change in the insights, credit for the addition 480-volt FLEX diesel generator was not evaluated further.

  The licensee states that the emergency raw cooling water and high-pressure fire protection systems would be available to provide inventory makeup to the SGs upon a loss of AFW. A review of the analysis accident sequences and cut sets indicates the auxiliary feedwater failure are immediate and, therefore, less than 1 hour would be available to connect these systems. Discussions with Region 2 staff indicate that these alternative systems would take longer to connect for these postulated scenarios. Therefore, credit for these systems was not provided.

  The licensee notes that restoration of RCS makeup would be possible after onset of reflux cooling for some time to prevent core damage and that the current treatment in the SPAR model is potentially conservative. The NRC agrees that the SPAR modeling assumption is potentially conservative. Language was added to Section 3.2 stating that the SPAR model assumption is potentially conservative. However, a review of the analysis accident sequences and cut set show that the potential impact is limited and, therefore, was not evaluated further.

SUBJECT: TRANSMITTAL OF SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2, EMERGENCY DIESEL GENERATOR 1B ACCIDENT SEQUENCE PRECURSOR REPORT (EPID L-2019-PMP-0033) DATED FEBRUARY 3, 2025

DISTRIBUTION:
PM Reading File
RidsACRS_MailCTR Resource
RidsNrrDorlLpl2-2 Resource
RidsNrrLAABaxter Resource
RidsNrrPMSequoyah Resource
RidsRgn2MailCenter Resource
CHunter, RES/DRA

ADAMS Accession Nos.:
Package ML25029A029
Letter ML25029A025
ASP Report ML25023A124

OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DORL/LPL2-2/BC
NAME	PBuckberg	ABaxter	DWrona
DATE	02/03/2025	02/03/2025	02/03/2025

OFFICIAL RECORD COPY