UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

June 1, 2009

Mr. Adam C. Heflin, Senior Vice
President and Chief Nuclear Officer
AmerenUE
P.O. Box 620
Fulton, MO 65251

Subject: CALLAWAY PLANT - NRC INTEGRATED INSPECTION REPORT 05000483/2009007

Dear Mr. Heflin

On April 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Callaway Plant. The enclosed inspection report documents the inspection findings, which were discussed on April 30, 2009, with Mr. Scott Sandbothe, Manager, Regulatory Affairs, and other members of your staff. This report documents baseline inspection activities related to a failure of emergency diesel generator Train B due to a jacket water leak which occurred on December 24, 2008. Region IV management decided to document this inspection in a separate report because the underlying performance deficiencies were complex and the preliminary significance of associated findings appeared to be of greater than very low safety significance. The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents an NRC-identified finding of very low safety significance (Green). This finding was determined to involve violations of NRC requirements. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this finding as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violation or the significance of the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Callaway Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the Callaway Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Vincent G. Gaddy, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure:
NRC Inspection Report 05000483/2009007
w/Attachments: Supplemental Information
Final Significance Determination Evaluation

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000483
License:	NPF-30
Report:	05000483/2009007
Licensee:	AmerenUE
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, MO
Dates:	January 29 through April 30, 2009
Inspectors:	D. Dumbacher, Senior Resident Inspector J. Groom, Resident Inspector
Reactor Analyst:	D. Loveless, Senior Reactor Analyst
Approved By:	V. Gaddy, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000483; 01/29/2009 – 04/30/2009; Callaway Plant, Operability Evaluations.

The report covered a three month long period of announced focused baseline inspection by a senior resident inspector and a resident inspector. One Green noncited violation of significance was identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after the licensee failed to adequately select suitable replacement parts essential to the operation of emergency diesel generator Train B. On December 24, 2008, during performance of Procedure OSP-NE-0001B, “Standby Diesel Generator ‘B’ Periodic Tests,” Callaway operations personnel identified that the emergency diesel generator Train B had an approximately 0.82 gallon per minute jacket water leak resulting in operators declaring the equipment inoperable. Upon removal, the gasket was found to be soft and extruding from the flange edge. The licensee originally concluded the gasket failed due to vibrations associated with engine shutdown but altered that conclusion after discussions with the resident inspectors and additional investigation. The licensee ultimately determined that the cause of the failure was due to incorrect gasket material being used during Job W200773 performed on October 16, 1999. The gasket was 1/8” thick versus 1/16” thick which resulted in a lack of compression. Since the gaskets are composed of an aramid fiberous material, the lack of compression allowed the gasket to absorb water and soften. The leak identified on December 24, 2008, developed once the gasket softened sufficiently to extrude from the flange edge. This issue has been entered into the licensee’s corrective action program as Callaway Action Request 200812985.

  This finding was greater than minor because it was associated with the mitigating systems cornerstone attribute of design control and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” this finding was determined to represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time. When evaluated per Manual Chapter 0609 Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," and the Callaway Plant Phase 2 pre-solved table item “Diesel Generator Fails to Run after Start,” the inspectors determined this finding to be potentially risk significant. This finding was forwarded to a senior reactor analyst for review. The results of the senior reactor analyst’s Phase 3 analysis determined the finding to be of very low safety significance. This finding did not have a crosscutting aspect since it was not a performance deficiency indicative of current licensee performance (Section 1R15).

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Mitigating Systems

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issue:

- January 29, 2009, jacket water gasket leak on emergency diesel generator Train B (CAR 200812985)

The inspectors selected this potential operability issue based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that Technical Specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the Technical Specifications and Final Safety Analysis Report to the licensee's evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. The inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one operability evaluations inspection sample as defined in Inspection Procedure 71111.15-04.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after the licensee failed to adequately select suitable replacement parts essential to the operation of emergency diesel generator Train B.

Description. On December 24, 2008, during performance of Procedure OSP-NE-0001B, "Standby Diesel Generator 'B' Periodic Tests," Callaway operations personnel identified that the emergency diesel generator Train B had an approximately 0.82 gallon per minute jacket water leak. Since the leak exceeded the licensee's acceptance criteria for jacket water leakage, operations personnel immediately declared the diesel engine inoperable. The source of the leakage was from a two-bolt flanged connection on the jacket water return line from the right turbocharger bank. Upon removal, the gasket was found to be soft and extruding from the flange edge. The failed gasket was replaced and emergency diesel generator Train B was successfully tested on December 25, 2008.

The licensee's initial past operability determination concluded that vibrations associated with engine shutdown contributed to the gasket failure observed on December 24, 2008. The inspectors reviewed the licensee's analysis and questioned if vibrations associated with engine shutdown are different than those associated with normal engine operation. Additionally, the inspectors questioned if other diesel generator gaskets were susceptible to vibration induced failure since they are subjected to similar operating conditions.

The licensee performed additional investigation and determined that vibrations did not contribute to the gasket failure. Their investigation concluded that gasket material used in the failure location was not original equipment material as specified by the diesel vendor. The failed gasket was 1/8" thick as opposed to 1/16" thick and had been installed during Job W200773 performed on October 16, 1999. The thicker gasket was generically approved for use by the licensee in Request for Resolution 4327A which inappropriately concluded the increased thickness was insignificant and would allow the system to seal under the same pressure as the thinner material. Consultation with the gasket vendor's technical support revealed that the increased thickness resulted in a lack of compression. Specifically, the gasket requires a minimum of 2500 psi of compression to seal. An estimate provided by the vendor determined that in the two-bolt configuration using the increased thickness gasket, the gasket was compressed only to approximately 2100 psi. Since the gaskets are composed of an aramid fiberous material, the lack of compression allowed the gasket to absorb water and soften. The leak identified on December 24, 2008, developed once the gasket softened sufficiently to extrude from the flange edge.

Since the emergency diesel generators rely on jacket water to remove heat from the engine during operation, the licensee performed a past operability determination to establish the impact of the failed gasket. Analysis by the licensee determined that once the level in the jacket water system was reduced to below the outlet of the cylinder heads, heat would begin to accumulate within the diesel engine and jacket water system. If water temperature exceeded 195°F the diesel engine would trip. Automatic makeup to the jacket water system would not be available during a design basis accident since the source of makeup is via the non-safety demineralized water system. Assuming no makeup, the licensee determined that the 0.82 gallon per minute leak would adversely affect engine performance after approximately 130 minutes of operation. In this scenario, the seven day mission time requirement of emergency diesel generator Train B would not be met.

Analysis. The performance deficiency associated with this finding involved the licensee's failure to adequately select suitable replacement parts essential to the operation of emergency diesel generator Train B. This finding was greater than minor because it was associated with the mitigating systems cornerstone attribute of design control and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," this finding was determined to represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time. The finding required a Phase 2 analysis. When evaluated per Manual Chapter 0609 Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," and the Callaway Plant Phase 2 pre-solved table item "Diesel Generator Fails to Run after Start," the inspectors determined this finding to be potentially risk significant. The finding was forwarded to a senior reactor analyst for review. The senior reactor analyst performed the Phase 3 analysis, which is contained in Attachment 2 of this report, which determined that the finding was of very low safety significance.

This finding did not have a crosscutting aspect since it was not a performance deficiency indicative of current licensee performance.

Enforcement. Title 10 of the Code of Federal Regulations Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for the selection and review for suitability of application of materials and parts that are essential to the safety related functions of structures, systems, and components. Contrary to the above, from October 16, 1999, through December 24, 2008, the licensee failed to ensure the suitability of repair parts essential to the safety-related function of emergency diesel generator Train B. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as CAR 200812985, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000483/2009007-01, Failure to Ensure Suitable Replacement Parts Essential for Emergency Diesel Generator Train B.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Mitigating Systems

.1 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors recognized a corrective action item documenting a potential non-compliance with ASME Section III, Class 3 code requirements. Specifically, CAR 200902095 documents that jacket water supply and return lines are two-bolt flanged connections versus four-bolt flanged connections as required by code. The licensee's Final Safety Analysis Report details the classification of structures, systems and components and lists piping and valves associated with diesel generator support systems and ASME Section III, Class 3, thereby requiring four-bolt flanged connections. Following discussions with the licensee and the NRC Office of Nuclear Reactor Regulation, the inspectors learned that the particular two-bolt flanged connections associated with jacket water cooling to the turbocharger are considered integral and internal to the diesel engine and are therefore exempt from the ASME Section III, Class 3 requirement to be a four-bolt configuration.

These activities constitute completion of one in-depth problem identification and resolution sample as defined in Inspection Procedure 71152-05.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On April 30, 2009, the inspectors presented the inspection results to Mr. Scott Sandbothe, Manager, Regulatory Affairs, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: Supplemental Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

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G. Bradley, Manager, Operations
T. Elwood, Supervising Engineer, Regulatory Affairs/Licensing
S. Maglio, Assistant Manager, Regulatory Affairs
K. Mills, Manager, Plant Engineering
S. Petzel, Engineer, Regulatory Affairs
S. Sandbothe, Manager, Regulatory Affairs
R. Wissel, System Engineer

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

050002009007-01	NCV	Failure to Ensure Suitable Replacement Parts Essential for Emergency Diesel Generator Train B (Section 1R15)

LIST OF DOCUMENTS REVIEWED

1R15: Operability Evaluations

JOBS

W200773	08009762	09000341

CALLAWAY ACTION REQUESTS

200812985

PROCEDURES

E-0	Reactor Trip or Safety Injection	Revision 12
ES 0.1	Reactor Trip Response	Revision 9
EC Supp Guide	Emergency Coordinator Supplemental Guideline	Revision 7
ODP-ZZ-00001 Addendum 1	Annunciator Response	Revision 2
OTA-KJ-00122	Annunciator Response Procedure Diesel Generator NE02 Control Panel	Revision 15
OTA-RK-00016	Addendum 23D, Diesel Generator NE02 Trouble	Revision 0
OTO-ZZ-00001	Control Room Inaccessibility	Revision 31

DRAWINGS

M-22KJ04 (Q)	Piping and Instrumentation Diagram Diesel Generator B Cooling Water System	Revision 19

M-22KJ05 (Q)	Piping and Instrumentation Diagram Standby Diesel Generator B Intake Exhaust, F.O. and Start Air System	Revision 24
M-22KJ06 (Q)	Piping and Instrumentation Diagram Standby Diesel Generator B Intake Lube Oil System	Revision 17

MISCELLAENOUS

RFR 4327A

PRA Evaluation Request 08-327	SDP Evaluation of “B” EDG Jacket Water Leak Failure	Revision 0
Calculation ZZ-267	Callaway IPE/PRA Sequence Qualification	Revision 0

4OA2: Identification and Resolution of Problems

CALLAWAY ACTION REQUESTS

200601311	200704436	200801270	200804210
200808567	200812985	200900121	200902095

REQUESTS FOR RESOLUTION

200903533

FINAL SIGNIFICANCE DETERMINATION EVALUATION

Callaway Plant
Improper Jacket Water Gasket Installed
Significance Determination Basis

A. Statement of Performance Deficiency

Maintenance and work control personnel failed to properly evaluate repair parts for Emergency Diesel Generator B, therefore, installing a jacket water system gasket of incorrect thickness in 1999. As a result, the gasket failed prematurely during a surveillance run causing a loss of functionality of the diesel generator.

B. Significance Determination Basis

1. Phase 1 Screening Logic, Results and Assumptions

In accordance with NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," the analyst determined that the failure to install the correct gasket was a licensee performance deficiency. The issue was more than minor because it was similar to Example 5.c in Manual Chapter 0612, Appendix E, and it met the "not minor if" requirement because the gasket was installed and the diesel generator was returned to service in the degraded configuration.

The analyst evaluated the issue using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones provided in Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." Although this finding affected multiple cornerstones, the analyst determined that the Mitigating Systems Cornerstone best reflected the dominant risk of the finding. The analyst determined that the finding represented an actual loss of safety function of Emergency Diesel Generator B for longer than the Technical Specification allowed outage time. Therefore, a Phase 2 estimation was conducted in accordance with Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations."

2. Phase 2 Risk Estimation

In accordance with Manual Chapter 0609, Appendix A, Attachment 1, "User Guidance for Phase 2 and Phase 3 Reactor Inspection Findings for At-Power Situations," the Senior Reactor Analyst evaluated the subject finding using the "Risk-Informed Inspection Notebook for Callaway Nuclear Generating Station, Unit 1," Revision 2.1a. The following assumptions were made:

a. The identified performance deficiency occurred in 1999 when Emergency Diesel Generator B was returned to service following the replacement of the subject jacket water system gasket. However, the deficiency only began to affect plant risk upon estimated time of failure.

b. The failure was identified during a diesel run on December 24, 2008. The last successful surveillance that placed the subject gasket in service was completed on November 24, 2008. Emergency Diesel Generator B was repaired and returned to service on December 25, 2008.

c. In accordance with Manual Chapter 0609, Appendix A, Attachment 2, “Site Specific Risk-Informed Inspection Notebook Usage Rules,” Rule 1.1, “Exposure Time,” the analyst evaluated the time frame over which the finding impacted the risk of plant operations. Because the exposure period over which the performance deficiency affected plant risk was unknown, the analyst selected one-half the time from the last successful test as the appropriate exposure time. Therefore, the exposure time used to represent the time that the performance deficiency affected plant risk in the Phase 2 estimation was between 3 and 30 days.

d. In accordance with Appendix A, Attachment 1, Step 2.1.3, “Find the Appropriate Target for the Inspection Finding in the Pre-solved Table,” the analyst determined that the appropriate target for evaluating this performance deficiency was “Diesel Generator Fails to Run after Start.” Therefore, the analyst utilized the pre-solved table associated with the SDP notebook to perform the estimation.

e. The analyst gave no operator action credit as discussed in Manual Chapter 0609, Appendix A, Attachment 1, Table 4, "Remaining Mitigation Capability Credit." The requirements to have procedures in place and to have trained the operators in recovery under similar conditions for such credit were not met.

The dominant sequences from the notebook were documented in Table A2-1 below:

TABLE A2-1

Failure of Emergency Diesel Generator B

Phase 2 Sequences

Initiating Event	Sequence	Mitigating Functions	Results
Loss of Offsite Power	4	LOOP-EAC-REC8	6
	8	LOOP-EAC-TDAFW-REC1	7
	11	LOOP-EAC-SEAL-REC4	7
Loss of ac Power	1	LEAC-PORV-LPR	7

Using the pre-solved worksheet, the result from this estimation indicated that the finding was of low to moderate safety significance (WHITE). However, the analyst determined that this estimate did not include a full coverage of the risk related to the failure identified, particularly because the diesel would have continued to run without intervention for approximately 3 hours following gasket failure. Therefore, a Phase 3 evaluation was conducted to better assess the risk of the finding related to internal initiators and fully assess the risk related to external initiators.

3. **Phase 3 Risk Analysis**

The analyst conducted a Phase 3 analysis of the risk, in accordance with Manual Chapter 0609, Attachment 1, Phase 3, “Risk Significance Estimation Using Any Risk Basis That Departs from the Phase 1 or 2 Process.” In accordance with Manual Chapter 0609, Appendix A, the analyst performed a Phase 3 analysis using the Standardized Plant Analysis Risk (SPAR) Model for Callaway, Revision 3.45, dated September 2008, to simulate failure of Emergency Diesel Generator 2 after a 2.8 hour period. Additionally, the analyst conducted an assessment of the risk contributions from external initiators using insights and/or values provided by the licensee’s Individual Plant Evaluation for External Events (IPEEE).

Assumptions:

In evaluating the risk related to the subject performance deficiency, the analyst made the following assumptions:

1. The best-available information indicated that the jacket water gasket reached a point in degradation that would result in failure during operation some time between the successful surveillance test on November 24, 2008, and its failure on December 24, 2008.

2. In accordance with Manual Chapter 0609, Appendix A, Attachment 1, Usage Rule 1.1, “Exposure Time,” the analyst determined that the exposure time should be estimated by using one half the time from the previous successful test to the failure because the time of inception of imminent failure is unknown.

3. Diesel Generator NE02 operated for 1.8 hours on December 24, 2008, and would have continued to operate for at least an additional hour prior to failing.

4. In addition to the exposure time in Assumption 2, Diesel Generator NE02 was known to be in a failed condition from the time of discovery on December 24, 2008 until it was returned to a functional status on December 25, 2008.

5. Given the nature of the leak, operators could have recovered Diesel Generator NE02 either prior to or after the engine tripped on high temperature by filling the jacket water surge tank with water from the fire water system.

6. The recovery documented in Assumption 5 is possible provided operators diagnose the condition, identify the availability of the recovery, and do not take actions to repair the machine that would make it unavailable for recovery.

7. Given the conditions, location of the gasket, and complexity of the required actions, it is unlikely that repair would have been completed prior to postulated core damage.

Method:

The analyst determined that the change in core damage frequency (ΔCDF) could best be quantified by evaluating the exposure period over which the performance deficiency impacted plant risk (EXP), the baseline frequency of a station blackout at Callaway ($\lambda_{SBO\text{-}BASE}$) and conditional core damage probability ($CCDP_{BASE}$), the frequency of a station blackout at Callaway with Diesel Generator NE02 failing ($\lambda_{SBO\text{-}CASE}$) and the associated CCDP ($CCDP_{CASE}$), and the likelihood that Diesel Generator NE02 would not be recovered before postulated core damage ($P_{DGA\text{-}NR}$). The change in risk would then be calculated as follows:

$$\Delta CDF = [(\lambda_{SBO\text{-}CASE} * CCDP_{CASE}) - (\lambda_{SBO\text{-}BASE} * CCDP_{BASE})] * EXP * P_{DGA\text{-}NR}$$

Exposure Period:

Using Assumptions 1 through 3, the analyst calculated an exposure time of 362 hours over which Diesel Generator NE02 would have failed in approximately 2.8 hours. Assumption 4 indicates that Diesel Generator NE02 was unavailable for use for an additional 28.3 hours.

Station Blackout Frequency:

The analyst quantified the likelihood of a station blackout (λ_{SBO}) occurring at Callaway for each of two conditions. The loss of offsite power frequency (λ_{LOOP}) for Callaway, provided by the Idaho National Laboratory is 3.59×10^{-2}/year. Using the SPAR model, the analyst solved for the probability of Diesel Generator NE01 failure ($P_{DG\text{-}A}$) as 7.65×10^{-2}. Therefore, for the portion of time where Diesel Generator NE02 was unavailable ($P_{DG\text{-}B} = 1.0$), the station blackout frequency was calculated as follows:

$$\begin{aligned}\lambda_{SBO\text{-}0} &= \lambda_{LOOP} * P_{DG\text{-}A} * P_{DG\text{-}B}\\ &= 3.59 \times 10^{-2}/\text{year} * 7.65 \times 10^{-2} * 1.0\\ &= 2.75 \times 10^{-3}/\text{year}\end{aligned}$$

The analyst then quantified the likelihood of a station blackout when Diesel Generator NE02 would have run for approximately 2.8 hours. The analyst determined that the probability that offsite power would not be recovered in 2.8 hours ($P_{LOOP\text{-}NR}$) was 2.34×10^{-1}. The probability that Diesel Generator NE01 would not be recovered in 2.8 hours ($P_{DGA\text{-}NR}$) was 7.75×10^{-1}. Therefore, for the portion of the exposure time that Diesel Generator NE02 would have failed in approximately 2.8 hours was calculated as follows:

$$\begin{aligned}\lambda_{SBO\text{-}2.8} &= \lambda_{LOOP} * P_{DG\text{-}A} * P_{DG\text{-}B} * P_{LOOP\text{-}NR} * P_{DGA\text{-}NR}\\ &= 3.59 \times 10^{-2}/\text{year} * 7.65 \times 10^{-2} * 1.0 * 2.34 \times 10^{-1} * 7.57 \times 10^{-1}\\ &= 4.86 \times 10^{-4}/\text{year}\end{aligned}$$

The baseline station blackout frequency ($\lambda_{SBO\text{-}BASE}$) was quantified using the SPAR model to be 2.41×10^{-4}/year.

Conditional Core Damage Probability:

In accordance with Assumption 3, using the plant-specific SPAR model for Callaway Plant, Revision 3.45, the analyst determined that the baseline CCDP for a station blackout was 4.0×10^{-2}. The analyst noted that this was the same CCDP for a station blackout when Diesel Generator NE02 failed at time zero. The CCDP for Diesel Generator NE02 failing at time 2.8 hours was calculated to be 1.59×10^{-1}.

Recovery:

The analyst used an event tree to quantify the probability of successful recovery of Diesel Generator NE02 following a postulated loss of offsite power and subsequent jacket water leak. Using fault-tree logic and/or the SPAR-H human reliability analysis method, the analyst determined the following approximate split fractions for this event tree:

Diagnosis	1.1×10^{-2}
Early Makeup	3.1×10^{-1}
Trip	1.5×10^{-3}
Choice	5.5×10^{-1}
Late Makeup	2.4×10^{-2}
Repair	1.0

The resulting nonrecovery probability was 1.86×10^{-1}.

Internal Events Results:

As stated in the Methods section, the analyst calculated the change in risk for the period of time that Diesel Generator NE02 would have failed in approximately 2.8 hours as follows:

$$\begin{aligned}\Delta CDF &= [(\lambda_{SBO\text{-}CASE} * CCDP_{CASE}) - (\lambda_{SBO\text{-}BASE} * CCDP_{BASE})] * EXP * P_{DGA\text{-}NR} \\ &= [(4.86 \times 10^{-4}/year * 1.59 \times 10^{-1}) - (2.41 \times 10^{-4}/year * 4.0 \times 10^{-2}) \\ &\qquad * (362\ hours \div 8760\ hours/year) * 1.86 \times 10^{-1} \\ &= 5.20 \times 10^{-7}\end{aligned}$$

The analyst then calculated the change in risk for the period of time that Diesel Generator NE02 was unavailable as follows:

$$\begin{aligned}\Delta CDF &= [(\lambda_{SBO\text{-}CASE} * CCDP_{CASE}) - (\lambda_{SBO\text{-}BASE} * CCDP_{BASE})] * EXP * P_{DGA\text{-}NR} \\ &= [(2.75 \times 10^{-3}/year * 4.0 \times 10^{-2}) - (2.41 \times 10^{-4}/year * 4.0 \times 10^{-2}) \\ &\qquad * (28.3\ hours \div 8760\ hours/year) * 1.0 \\ &= 3.23 \times 10^{-7}\end{aligned}$$

Because the two cases were selected specifically to make them independent, the analyst calculated the total internal events ΔCDF by summing the two cases (8.43×10^{-7}).

Internal Fire:

The analyst reviewed the licensee's IPEEE dated June 30, 1995. Given the short exposure period, the analyst determined that the impact on most of the risk-significant fire areas, documented in Table 4.2.1-1, "Summary of Fire Modeling Results," was negligible for this evaluation. However, the analyst noted that Diesel Generator NE02 was the only protected source of ac power during a main control room abandonment scenario. Therefore, this scenario was further evaluated.

Like the evaluation of internal initiators, the analyst determined that the change in risk could be calculated using the following equation:

$$\Delta CDF = [(\lambda_{SBO\text{-}CASE} * CCDP_{CASE}) - (\lambda_{SBO\text{-}BASE} * CCDP_{BASE})] * EXP * P_{DGA\text{-}NR}$$

The analyst noted that, upon control room abandonment, the licensee's procedures cause a station blackout prior to placing Diesel Generator NE02 in service. Therefore, the station blackout frequency is equal to the control room abandonment frequency ($\lambda_{Abandon}$). The analyst calculated this by combining the fire ignition frequency (λ_{Fire}) and the abandonment nonsuppression probability (P_{NS}) as follows:

$$\begin{aligned} \lambda_{SBO\text{-}Fire} &= \lambda_{Abandon} \\ &= \lambda_{Fire} * P_{NS} \\ &= 1.01 \times 10^{-3} * 3.40 \text{ X } 10^{-2} \\ &= 3.43 \times 10^{-5} \end{aligned}$$

The approximate baseline CCDP for a control room abandonment was 0.1 as documented in the IPEEE. The analyst made the bounding assumption that, without the protected power source, the CCDP would be 1.0, indicating that the postulated event would lead to core damage.

The analyst reassessed the nonrecovery probability used in the internal events assessment. Given that Diesel Generator NE02 would be the only qualified power source, the analyst assumed that the licensee would not attempt to remove the machine from service as long as a viable recovery existed. Requantifying the probability of nonrecovery provided a value of 0.1219.

The analyst then calculated the change in risk related to the subject performance deficiency as follows:

$$\begin{aligned} \Delta CDF &= [(\lambda_{SBO\text{-}CASE} * CCDP_{CASE}) - (\lambda_{SBO\text{-}BASE} * CCDP_{BASE})] * EXP * P_{DGA\text{-}NR} \\ &= [(3.43 \times 10^{-5}/\text{year} * 1.0) - (3.43 \times 10^{-5}/\text{year} * 0.1)] \\ &\qquad * 16 \text{ days} \div 365 \text{ days/year} * 0.1219 \\ &= 1.65 \times 10^{-7} \end{aligned}$$

Seismic Events:

The analyst reviewed the IPEEE and determined that the licensee had used the seismic margins analysis method in assessing the plant at a 0.3g peak ground acceleration review level earthquake. All safe shutdown paths were determined to have a high confidence of survival at the review level earthquake. The analyst noted that the Risk Assessment of Operational Events Handbook, Volume 2, "External Events," Revision 1.01, provided a frequency of exceedance for a 0.3g earthquake at Callaway of 7.035×10^{-6}. Given a 16-day exposure period, the analyst determined that the frequency of a seismic event that could be impacted by the performance deficiency beyond the initiating event alone would be 3.1×10^{-7}. This value would have to be combined with the CCDP for each seismic bin, making the overall risk much lower. Therefore, the analyst determined that the change in risk from seismic events would be negligible for the subject performance deficiency.

High Winds, Floods and Other External Events:

The analyst reviewed the IPEEE and determined that no other credible scenarios initiated by high winds, floods, external fire, and other external events could initiate a LOOP and directly cause failure of other risk significant components important to the subject finding. Therefore, the analyst concluded that external events other than seismic events were not significant contributors to risk for this finding.

Risk Contribution from Large Early Release Frequency (LERF):

Using Manual Chapter 0609 Appendix H, "Containment Integrity Significance Determination Process," the analyst determined that this was a Type A finding (i.e., LERF contributor) for a large dry containment. For pressurized water reactor plants with large dry containments (like Callaway), only findings related to accident categories of intersystem loss of coolant accidents and steam generator tube ruptures have the potential to impact LERF. In addition, an important insight from the individual plant evaluation program and other probabilistic risk assessment studies is that the conditional probability of early containment failure is less than 0.1 for core damage scenarios that leave the reactor coolant system at high pressure (>250 psi) at the time of reactor vessel breach. The analyst noted that none of the cutsets were from steam generator tube rupture or intersystem loss of coolant accident sequences. Therefore, the analyst determined that the change in risk related to the subject performance deficiency was insignificant with respect to LERF.

C. Final Significance Determination

Because the initiators reviewed for internal events and external initiators are independent by nature, the analyst calculated the total ΔCDF by summing the two results (1.0×10^{-6}). Although this result was on the threshold of very low safety significance and low to moderate safety significance the analyst determined that the fire risk estimate was a bounding evaluation rather than best estimate. However, further detailed evaluation was not warranted to show quantitatively that the ΔCDF was less than the threshold. The analyst also showed that this finding was not significant with respect to the LERF. Therefore, the analyst determined that this finding was of very low risk significance (Green).