UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

December 7, 2007

EA-07-194

D.J. Bannister, Site Director
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION - FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION; NRC INSPECTION REPORT 05000285/2007011

Dear Mr. Bannister:

The purpose of this letter is to provide you the final results of our significance determination of the preliminary Greater than Green finding identified in the subject inspection report. The NRC's final risk-informed conclusion is that the finding discussed in this letter is best characterized as a White finding. Our rationale for this conclusion is discussed below, as well as in Enclosure 2.

Our preliminary finding was discussed with your staff during an exit meeting on September 18, 2007. The inspection finding was addressed using the Significance Determination Process (SDP) and was preliminarily characterized as Greater than Green, a finding with more than very low safety significance, that may require additional NRC inspections. As described in Section 4OA2 of the subject inspection report, contamination containing dust and oil was found on the field flash relay auxiliary contact surfaces, which apparently caused the February 14, 2007, failure of the Train A emergency diesel generator. Our inspection determined there were multiple performance deficiencies in that: (1) maintenance personnel were applying an unapproved wet lubricant to the auxiliary contact sliding mechanisms, contrary to vendor recommendations and in the absence of procedural controls; (2) Fort Calhoun Station staff did not treat the February 14, 2007, emergency diesel generator failure as a significant condition adverse to quality; and (3) actions in response to applicable operating experience were not timely and did not prevent this condition from occurring.

The NRC's preliminary assessment of the safety significance of this inspection finding, which is documented in Attachment 2 of NRC Inspection Report 05000285/2007011 (ML072840428), resulted in an increase in core damage frequency (CDF) for internal events of 8.6E-6/year, or White for safety significance. This assessment was dependent on the influential assumption of a 14-day exposure time, and included external initiating events known to be potentially significant contributors to the overall significance of the finding. At the completion of our inspection, you had not completed your own final analysis of the significance of the finding. Pending completion of your analysis, and review by the NRC staff, we considered the significance of the inspection finding best characterized as Greater than Green.

At the request of the Omaha Public Power District (OPPD), a Regulatory Conference was held on November 8, 2007, to discuss OPPD's position on the safety significance of the finding and corrective actions taken in response to the failure of the auxiliary contacts in the emergency diesel generator field flash circuit. During the Regulatory Conference, OPPD agreed with the finding as characterized in NRC Inspection Report 05000285/2007011, and your staff described the corrective actions taken in response to the finding. Further, your staff asserted that the safety significance was low to moderate (i.e., White). Your staff's analysis and conclusions are included as an enclosure to the Regulatory Conference Meeting Summary (ML073210005), issued on November 13, 2007.

After careful consideration of the information developed during the inspection, and the information your staff provided at the Regulatory Conference, the NRC has concluded that the inspection finding is appropriately characterized as White. This was based upon an SDP Phase 3 analysis performed by the NRC staff using a Standardized Plant Analysis Risk (SPAR) model simulation of the failed Train A emergency diesel generator, as well as an assessment of the risk contributions to external initiators using insights and values provided by your staff prior to and during the Regulatory Conference. Based upon our assessment of the applicable information, the NRC staff found your assumptions and analyses of the applicable accident scenarios to be appropriate. Specifically, we agree with your assumed exposure time of T/2, or 14 days and your assumptions regarding external initiating events. Therefore, we estimated the change in core damage frequency associated with this condition, for both internal and external events, to be 5.4E-6/year, as discussed in Enclosure 2 to this letter. Your final significance of 5.1E-6/year, as shown in the enclosure to the aforementioned Regulatory Conference Meeting Summary, closely agreed with our result.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in the NRC Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 2, "Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)."

The NRC has also determined that the failure of the Train A emergency diesel generator field flash auxiliary contacts involved two violations of NRC requirements. These violations are described in detail in the referenced inspection report. The first was a violation of 10 CFR 50, Appendix B, Criterion XVI (Corrective Action), with two examples, for the failure to:
1) determine the cause of the February 14, 2007, emergency diesel generator failure, a significant condition adverse to quality, and take corrective action to preclude repetition;
and 2) promptly identify and correct a significant condition adverse to quality (high resistance on field flash circuit contacts) after determining that similar operating experience was applicable. The second was a violation of Technical Specification 5.8.1.a (Procedures) for failing to establish a procedure for proper lubrication of the auxiliary contact sliding mechanism, an activity that affected the performance of the emergency diesel generator. These violations are cited in the enclosed Notice of Violation (Notice). In accordance with the NRC Enforcement Policy, the Notice is considered escalated enforcement action because it is associated with a White finding. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

In addition, we will use the NRC Action Matrix, as described in NRC Inspection Manual Chapter 0305, “Operating Reactor Assessment Program,” to determine the most appropriate NRC response and any increase in NRC oversight. We will notify you by separate correspondence of that determination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC’s document system (ADAMS), accessible from the NRC Web site at www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction.

Sincerely,

/RA/

Elmo E. Collins,
Regional Administrator

Docket No. 50-285
License No. DPR-40

Enclosures:
1. Notice of Violation
2. Final Significance Determination

cc w/enclosures:
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SUNSI Review Completed: _gbm__ ADAMS: ☑ Yes ☐ No Initials: _gbm__
☑ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☑ Non-Sensitive

RIV:SRI:DRP/E	C:DRP/E	D:DRP	D:DRS	ACES
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NOTICE OF VIOLATION

Omaha Public Power District
Fort Calhoun Station

Docket No. 50-285
License No. DPR-40
EA-07-194

During an NRC inspection completed on September 17, 2007, two violations of NRC requirements were identified. In accordance with the NRC Enforcement Policy, the violations are listed below:

A. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. A condition that would cause a failure of an emergency diesel generator is a Significant Condition Adverse to Quality (SCAQ).

   Contrary to the above, prior to February 14, 2007, the licensee failed to promptly identify and correct a significant condition adverse to quality involving high resistance across the field flash contacts of a relay in the Train A emergency diesel generator (EDG) voltage regulator circuit. Specifically, on September 16, 2006, the licensee had determined that operating experience (OE) associated with an EDG failure was applicable to Fort Calhoun, but the licensee failed to promptly identify and correct high electrical resistance on the field flash relay 2CR auxiliary contacts (the same issue that the OE addressed). On February 14, 2007, the EDG failed during a surveillance test because of high resistance across the field flash contacts. In a second example of this violation, as of April 30, 2007, the licensee failed to determine the cause of the February 14, 2007, Train A EDG failure (a significant condition adverse to quality) and to take corrective actions to preclude repetition.

B. Fort Calhoun Technical Specification 5.8.1(a) states, in part, that written procedures shall be established, implemented and maintained covering the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A, 1978. Section 9 of this guide states, in part, that maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances.

   Contrary to the above, as of February 16, 2007, the licensee failed to provide a written procedure for maintenance that could affect the performance of safety-related EDG voltage regulator relay auxiliary contacts. Specifically, the licensee failed to establish a written procedure for the proper lubrication of the safety-related auxiliary contact sliding mechanisms.

These violations are associated with a White SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Omaha Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-07-194" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 7th day of December, 2007

Final Significance Determination

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," the analyst performed a Phase 3 analysis using the Standardized Plant Analysis Risk (SPAR) Model for Fort Calhoun, Revision 3.31, dated April 10, 2006, to simulate the failed Train A emergency diesel generator. Additionally, the analyst conducted an assessment of the risk contributions to external initiators using insights and/or values provided by the licensee's probabilistic risk assessment model and simplified fire probabilistic risk assessment.

<u>Assumptions</u>:

To evaluate the change in risk caused by this performance deficiency, the analyst made the following assumptions:

A. The vital batteries at Fort Calhoun will deplete after approximately 2.6 hours of full post-accident loads without an operating battery charger. However, if operators take actions to shed unnecessary loads from the vital dc buses, as required by Emergency Operating Procedure, Attachment 6, "Minimizing DC Loads," the batteries will last for 8 hours following loss of power to the chargers.

B. Idaho National Laboratories provided an appropriate update to Revision 3.31 of the SPAR model that reflected the 8-hour batteries, as described in the updated safety analysis report.

C. The Fort Calhoun SPAR, Revision 3.31 model, with the modification discussed in Assumption B, represents an appropriate tool for evaluation of the subject finding.

D. The field flash relay auxiliary contacts for Diesel Generator 1 were continuously degrading via an advanced aging process caused by the combination of dust continuously provided by forced ventilation and the migration of an unapproved lubricant applied by licensee craftsmen to the contactor actuator.

E. Diesel Generator 1 successfully started and loaded during a surveillance performed on January 17, 2007. The generator failed to load during a surveillance on February 14, 2007, because the field flash relay auxiliary contacts failed as discussed in Assumption D.

F. The exposure time used for evaluating this finding should be determined in accordance with Inspection Manual Chapter 0609, Appendix A, Attachment 2, "Site Specific Risk-Informed Inspection Notebook Usage Rules."

   Attachment 2 discusses the approach to establishing the exposure time that should be used for the significance determination process. Step 1.1 states:

   > "If the inception of the condition is unknown, then an exposure time of one half of the time period since the last successful demonstration of the component or function (*t/2*) should be used."

G. The appropriate exposure time (the time that Diesel Generator 1 was not functional) for use in this evaluation is 14 days.

The exact time at which the contacts discussed in Assumptions D and E were in a condition that would result in failure is unknown. Therefore, in accordance with Assumption F, Diesel Generator 1 would not have started and loaded upon demand for one half of the period from January 17 through February 14, or for a 14-day period.

H. Given the condition of the auxiliary contacts in the field flash relay and the complexity of identifying the condition, operators would not have been able to recover Diesel Generator 1 prior to postulated core damage.

I. The likelihood that Diesel Generator 2 would fail from degrading field flash relay auxiliary contacts, given a failure of Diesel Generator 1, (e.g., common cause failure) was 0.02 per demand.

The analyst reviewed the licensee's use of this beta factor for the common cause failure value. The analyst determined that, given the contactors on Diesel Generator 2 were 20 years newer than those on Diesel Generator 1, and that the Diesel Generator 2 contacts had not indicated degradation toward failure, this value was appropriate. Essentially, the failure rate for Diesel Generator 2 was doubled by using this beta factor.

J. The nominal likelihood for a loss of offsite power was unaffected by the subject finding.

K. Evaluating the risk contribution of this finding related to seismic events and internal flooding is appropriately conducted by adding an additional 9.3% to the change in risk for internal events quantified by the SPAR model.

The scope of the licensee's probabilistic risk assessment model includes the contribution to seismic events and internal flooding, initiators that are not evaluated by the SPAR. These external initiators represented 9.3% of the total risk affect for the loss of Diesel Generator 1. The analyst evaluated cutsets provided by the licensee's model and determined that this was an appropriate fraction for use in assessing these initiators.

L. The licensee's fire risk model is an appropriate tool for evaluation of the subject finding.

The analyst independently evaluated the risk change related to internal fires. These insights were then used to challenge and evaluate the results of the licensee's model. In all cases, the licensee's model covered the scenarios posed by the analyst and included a larger scope of fires than was feasible for the analyst to evaluate.

M. Those differences between the SPAR and the licensee's models that were not adjusted, as documented under Assumptions A through C above, were inconsequential.

The analyst, in reviewing the differences between the models, determined that there were several global differences including: the power-operated relief valve success criteria for once-through cooling; the probability of failure for certain load-shed scenarios; and the level of detail in modeling certain unique initiators. However, the analyst determined that these differences were not of consequence to this evaluation because the final results were within a factor of 2 of each other.

Internal Initiating Events:

The Senior Reactor Analyst used the SPAR model for Fort Calhoun Station, as modified, to estimate the change in risk associated with internal initiators that was caused by the finding. Average test and maintenance of modeled equipment was assumed, and a cutset truncation of 1.0×10^{-12} was used.

Consistent with guidance in the Risk Assessment Standardization Project Handbook, including NRC document, "Common-Cause Failure Analysis in Event Assessment, (June 2007)," the analyst modeled the condition by adjusting the following basic events in the SPAR model:

Basic Event	Original Value	Conditional Value
EPS-DGN-FS-1A	$5.0 \text{ X } 10^{-3}$	TRUE
EPS-DGN-FR-1A	$2.41 \text{ X } 10^{-2}$	IGNORE
EPS-DGN-CF-START	$6.55 \text{ X } 10^{-5}$	0.02

The SPAR baseline core damage frequency (CDF_{Base}) was 7.37×10^{-6}/year. The evaluation case for the above change set resulted in a conditional core damage frequency (CCDF) of 8.25×10^{-5}/year. The dominant core damage sequences were documented in the table below:

Initiating Event	Sequence	Top Failures	Frequency
Loss of Offsite Power	22-12	Failure of DG-2 with Battery Depletion at 8 hours	4.25×10^{-5}/year
	22-15	Failure of DG-2 with Battery Depletion at 2.6 hours	1.26×10^{-5}/year
	21	Failure of Turbine and Diesel-Driven Auxiliary Feedwater Pumps	1.06×10^{-5}/year

Final Significance Determination

The change in incremental conditional core damage frequency (ICCDF) was calculated as follows:

$$\begin{aligned} ICCDF &= CCDF - CDF_{base} \\ &= 8.25 \times 10^{-5}/\text{year} - 7.37 \times 10^{-6}/\text{year} \\ &= 7.51 \times 10^{-5}/\text{year} \end{aligned}$$

Therefore, the change in core damage frequency ($\Delta CDF_{Internal}$) caused by this finding and related to internal initiators was calculated as follows:

$$\begin{aligned} \Delta CDF_{Internal} &= ICCDF * \text{Exposure Period} \\ &= 7.51 \times 10^{-5}/\text{year} * 14 \text{ days} \div 365 \text{ days/year} \\ &= 2.88 \times 10^{-6} \end{aligned}$$

This result indicated that the finding was of low to moderate significance with respect to the risk of internal initiating events.

<u>External Initiating Events</u>:

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Step 2.2.5, "Screen for the Potential Risk Contribution from External Initiating Events," the increase in risk of the inspection finding related to fire, flooding, severe weather, seismic, or other initiating events that were considered by the licensee's Individual Plant Evaluation for External Events analysis must be accounted for in any Phase 3 evaluation. Experience with using the site specific risk-informed inspection notebooks has indicated that accounting for external initiators could result in increasing the risk significance attributed to an inspection finding by as much as one order of magnitude. Therefore, if the internal initiating events evaluation for an inspection finding represents an increase in risk of greater than or equal to 1×10^{-7}/year, a senior reactor analyst or other appropriate NRC risk analyst should perform a Phase 3 analysis to estimate the increase in risk from these initiators.

<u>Seismic and Internal Flooding</u>: As stated in Assumption K, the analyst assumed that the change in risk related to seismic events and internal flooding ($\Delta CDF_{Seismic}$) can best be quantified by adding 9.3% (Seismic Factor) to the risk related to internal initiators. Therefore, this additional risk was calculated as follows:

$$\begin{aligned} \Delta CDF_{Seismic} &= \Delta CDF_{Internal} * \text{Seismic Factor} \\ &= 2.88 \times 10^{-6} * 0.093 \\ &= 2.68 \times 10^{-7} \end{aligned}$$

Internal Fire: As documented in Assumption L, the analyst used the licensee's fire model results to quantify the change in risk related to internal fire. The result of the licensee's fire model for the failure of Diesel Generator 1 ($CCDF_{LicFire}$) was 8.05×10^{-5}/year. The baseline for this model ($CDF_{LicBase}$) was 2.21×10^{-5}/year. The dominant core damage sequences were documented in the table below:

Initiating Event	Top Failures
Fires Affecting Diesel Generator 2	Consequential Loss of Offsite Power, Battery Depletion, and Failure of Operators to Use Diesel-Driven Auxiliary Feedwater Pump after Battery Depletion
Various Motor Control Center Fires	Consequential Loss of Offsite Power, Battery Depletion, and Failure of Operators to Use Diesel-Driven Auxiliary Feedwater Pump after Battery Depletion
Various Transformer Fires	Consequential Loss of Offsite Power, Battery Depletion, and Failure of Diesel Generator 2

The change in incremental conditional core damage frequency was calculated as follows:

$$ICCDF_{LicFire} = CCDF_{LicFire} - CDF_{LicBase}$$

$$= 8.05 \times 10^{-5}\text{/year} - 2.21 \times 10^{-5}\text{/year}$$

$$= 5.84 \times 10^{-5}\text{/year}$$

Therefore, the change in core damage frequency ($\Delta CDF_{LicFire}$) caused by this finding and related to internal fires was calculated as follows:

$$\Delta CDF_{LicFire} = ICCDF_{LicFire} * \text{Exposure Period}$$

$$= 5.84 \times 10^{-5}\text{/year} * 14 \text{ days} \div 365 \text{ days/year}$$

$$= 2.24 \times 10^{-6}$$

Other External Initiators: The analyst reviewed the licensee's Individual Plant Evaluation for External Events analysis for initiators other than those previously discussed, including but not limited to high winds, external floods, transportation accidents, and external fires. Additionally, the analyst evaluated the impact of these initiators on the analysis previously performed and documented in NRC Inspection Report 05000285/2005/010, dated April 15, 2005. The analyst determined that the failure of Diesel Generator 1 would have negligible impact on the risk derived from these initiators and would not result in an increase in the final characterization of the finding.

External Events Summary: As documented above, the analyst determined that the external events important to the risk associated with the subject finding were seismic events, internal flooding and internal fire. The seismic and flood contributors were evaluated using a percentage of the internal events result. The analyst calculated a ΔCDF of 2.68×10^{-7} over the exposure period. Using the licensee's fire model results, the analyst calculated a fire ΔCDF of 2.24×10^{-6} over the exposure period. Therefore the risk of the subject finding related to external events ($\Delta CDF_{External}$) was the sum of the two, 2.51×10^{-6}. This result indicated that the finding was of low to moderate significance with respect to the risk of external initiating events.

Large, Early Release Frequency:

Using NRC Inspection Manual Chapter 0609, Appendix H, 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, only findings related to accident categories involving intersystem loss of coolant accidents and steam generator tube ruptures have the potential to impact the large, early release frequency. In addition, an important insight from the individual plant evaluation program and other probabilistic risk assessments was 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. Given that this finding is not related to intersystem loss of coolant accidents nor steam generator tube ruptures and that the core damage scenarios for this finding leave the reactor coolant system at high pressure, the analyst concluded that the large, early release frequency was not significantly affected by the subject finding.

Final Phase 3 Result:

The total risk contribution of the finding is expressed as the summation of the internal events contribution and the external events contribution. The change in internal event risk was estimated as 2.88×10^{-6} over the exposure period. The risk related to external initiators changed by 2.51×10^{-6}. Therefore total ΔCDF for the subject finding can be calculated as the following sum:

$$\Delta CDF = \Delta CDF_{External} + \Delta CDF_{Internal}$$

$$= 2.51 \times 10^{-6} + 2.88 \times 10^{-6}$$

$$= 5.39 \times 10^{-6}$$

This ΔCDF is the sole basis for the characterization of the subject performance deficiency because the finding did not affect the large, early release frequency in any significant manner. Therefore the finding is of low to moderate risk significance (White).