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

July 18, 2011

EA-11-025

David J. Bannister, Vice President
and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4
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 05000285/2011007

Dear Mr. Bannister:

The purpose of this letter is to provide you the final significance determination of the preliminary Yellow finding identified in our previous communication dated May 6, 2011, which included the subject inspection report. The inspection finding was assessed using the Significance Determination Process and was preliminarily characterized as a Yellow finding with substantial importance to safety that may result in additional NRC inspection and potentially other NRC action. This finding was associated with the June 14, 2010, failure of a reactor trip contactor (M2) in your reactor protection system.

At your request, a regulatory conference was held on June 2, 2011, to further discuss your views on this issue. During the regulatory conference, your staff described the Fort Calhoun Station's assessment of the significance of the finding and they provided a summary of the corrective actions, and insights from the root cause analysis of the finding. This material is documented in the NRC Meeting Summary (ML111660027) dated June 14, 2011. You also requested that the NRC reconsider its evaluation of the finding's risk significance based on four specific areas of consideration where differences exist between the NRC's preliminary significance determination and your staff's risk assessment. These are: 1) Shorter Exposure Time (T/2 + repair vs. T + repair); 2) Lower Failure Probability for Clutch Power Supply Breaker; 3) Common Cause Failure Determination; and 4) Higher Operator Reliability in Tripping the Reactor. Between June 6 and June 28, 2011, you provided supplemental information regarding follow-up questions asked by NRC staff at the conference. This additional material was docketed as ADAMS document ML111881131.

The NRC has reviewed your areas of consideration and our evaluation of each is provided in Enclosure 2 of this letter along with the revised NRC risk assessment. The NRC considered the information developed during the inspection, and the information that you provided at, and subsequent to, the conference. The NRC has concluded that the finding is appropriately characterized as White, a finding with low to moderate importance to safety and will result in additional NRC inspection and potentially other NRC actions.

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 NRC Inspection Manual Chapter 0609, Attachment 2. An appeal must be sent in writing to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd., Suite 400, Arlington, Texas 76011-4125.

The NRC has concluded that failure to assure that the cause of a significant condition adverse to quality was determined and failure to take corrective actions to preclude repetition of the condition, is a violation of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix B, Criterion XVI, "Corrective Action," as cited in the enclosed Notice of Violation. The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, the Notice of Violation is considered an escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter. Please follow the instructions specified in the enclosed Notice of Violation when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Because your current plant performance is in the Degraded Cornerstone (Mitigating Systems) Column, and this violation also impacts that cornerstone, the NRC will use the NRC Action Matrix to determine the most appropriate NRC response to this violation. The NRC 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 available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS). ADAMS is 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: 50-285
License: DPR-40

Enclosures:
1. Notice of Violation

2. Fort Calhoun Reactor Protection System Issue
   Final Significance Determination

cc w/Enclosures:

Distribution via Listserv

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ADAMS		☒ Yes		☒SUNSI Review Complete	Reviewer Initials: JAC
☒ Publicly Available		☐Non-publicly Available		☐Sensitive	☒ Non-sensitive

RIV/DRP:PBE	DRP:PBE	DRS-SRA	D:DRS	ACES
RVAzua	JAClark	DPLoveless	AVegel	RKellar
/RA/	/RA/	/RA/	/RA/	/RA/via email
07/08/11	07/08/11	07/14/11	07/14/11	07/07/11
Counsel	NRR/OE		D:DRP	ORA
MBarkman Marsh	NColeman		KMKennedy	EECollins
/RA/via email	/RA/via email		/RA/	/RA/
07/13/11	07/13/11		07/15/11	07/18/11

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NOTICE OF VIOLATION

Omaha Public Power District
Fort Calhoun Station

Docket No.: 05000285
License No.: DPR-40
EA-11-025

During an NRC inspection conducted from January 17 through April 15, 2011, one violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, 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 nonconformances 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.
>
> Contrary to the above, between November 3, 2008, and June 14, 2010, the licensee failed to assure that the cause of a significant condition adverse to quality was determined and corrective actions were taken to preclude repetition. Specifically, the licensee failed to preclude shading coils from repetitively becoming loose material in the M2 reactor trip contactor. The licensee failed to identify that the loose parts in the trip contactor represented a potential failure of the contactor if they became an obstruction; and therefore, failed to preclude repetition of this significant condition adverse to quality, that subsequently resulted in the contactor failing.

This violation is associated with a White significance determination process finding in the Mitigating Systems Cornerstone.

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 20555-0001 with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125, and a copy to the NRC Resident Inspector - Fort Calhoun Station, 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-11-025" 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, 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's website at 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.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 18th day of July 2011

During the regulatory conference held on June 2, 2011, the Fort Calhoun Station (FCS) staff described your assessment of the significance of the finding as summarized below. Specifically, your staff discussed four differences that existed between the NRC's preliminary significance determination and your risk assessment. These differences and our conclusions are as follows:

Item 1 – Shorter Exposure Time (T/2 + repair vs. T + repair)

Your staff stated that exposure time for this issue should not utilize "T" plus repair time, but use "T/2" plus repair time instead. This would result in a reduced exposure period from 64.0 days to 32.5 days. This was based on your analysis that a shading coil must fragment, due to wear, prior to a piece of it being able to jam the contactor in the closed position. You also stated this wear would likely take weeks or months. Therefore, you concluded that the fragmenting and jamming occurred at some unknown time between April 10, and June 14, 2010. This would indicate that the use of T/2 is more applicable to this case.

NRC staff determined that the provided failure modes and effects analysis for the shading coil was very comprehensive and understandable. However, there was no corresponding failure modes and effects analysis presented for the overall contactor (i.e., how the shading coil failure could cause the contactor failure). Definitive testing or evaluation of the jamming sequence for the contactor was not provided.

During discussions with your forensic specialist at the regulatory conference, NRC staff questioned the methods used to determine how the shading coil actually jammed the contactor. The specialist indicated that specific confirmation testing was not conducted, but that a shading coil fragment was likely repositioned during vibration, moved in an upward direction, and then jammed the contactor mechanism in its opening motion on June 14, 2010. Based on visual and physical evidence, NRC staff concluded that this was unlikely. The travel on the contactor mechanism, from full contact closure until the contacts open, was only approximately 1/8 inch. The NRC staff concluded it would be extremely difficult for a shading coil fragment to both enter the gap between the frame and the contactor slide and stop the contactor slide from moving in such a small amount of travel. However, when a contactor slide moves from the full open to the closed position, the travel is over 1/2 inch. The NRC staff believes it is more likely a whole shading coil or fragment was forced into the gap between the frame and the contactor slide during a closing action; specifically the April 10, 2010, closing prior to the June 14, 2010, failure. Therefore, the NRC concludes the applicable exposure time was 63 days, plus a 1 day repair time, for a total of 64 days.

Item 2 – Lower Failure Probability for Clutch Power Supply Breaker

Your staff stated that the generic breaker failure data used in the preliminary significance determination was not the best available information for vital breakers CB-AB and CB-CD. Instead your staff suggested that the NRC staff use generic data from NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," plus data developed using test results from testing the two breakers previously installed at Fort Calhoun. However, your final assessment indicated that you believed a Bayesian update of the test data, using a Jeffreys non-informative prior distribution would be the appropriate value.

The NRC staff determined that, to the extent the test data from the previously installed breakers represented the installed conditions of the breakers, this data should be used to update the generic data. However, the NRC staff concluded that the test data should not be used to update a Jeffreys non-informative prior distribution when existing generic priors were available that adequately represented the population of the breakers in question. The staff also concluded that data from NUREG/CR-6928 should not be used because the breakers in question were neither reactor trip breakers nor were they maintained and tested to the standards used for reactor trip breakers.

The NRC staff updated the priors used in the preliminary significance determination with the data obtained from the test results on vital breakers CB-AB and CB-CD. The NRC concluded that this approach represented the best available information. The calculated total failure probability for the breakers was 3.81×10^{-4} demand which is a change from 7.5×10^{-3} documented in the preliminary determination.

Item 3 – Common Cause Failure Determination

Your staff stated that there was no single clear path for analysis of common cause failure for this issue and recommended that the NRC staff use the definition of common cause failure documented in NUREG/CR-5500, Volume 10, "Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998." Additionally, your staff commented that the NRC staff made an incorrect reference to Revision 1.01 of the Risk Assessment of Operational Events handbook in our inspection report. Finally, your staff stated that the common cause observations in the inspection report under Assumption 7 may need to be updated based on new information provided in the Engineering Systems, Inc. report.

The NRC staff determined that the reference to Revision 1.01 of the handbook was incorrect. However, this definition was not used in the common cause methodology utilized in our analysis. The reasons for adjusting the common cause failure probability were best described in the inspection report Page A-4, Assumptions 7 and 8.

The NRC staff also determined that NUREG/CR-5500 provides a concise definition of a common cause failure. However, in the significance determination, the NRC staff did not assume that a common cause failure event had occurred. If a failure of Contactors M1 and M2 had occurred at the same time, the risk would have been significantly higher than our original estimates. The guidance contained in NUREG/CR-5500 was not intended to be used to evaluate a condition where the analyst believes that the common cause failure probability should be increased based on observed conditions. The NRC staff has determined that the approach used in the inspection report is the appropriate method to adjust common cause failure probabilities when components are maintained and operated under similar conditions.

The NRC staff reviewed Assumption 7 in the NRC inspection report in light of the findings documented in the report generated by the professional engineering consulting firm Engineering Systems, Inc. However, the only condition that may have changed based on the Engineering Systems, Inc. report was that, "subparts exhibited significant scratching and indentations." The NRC staff determined that despite such a change, the subject conditions, operation and maintenance history of the contactors still warranted adjustment of the common cause failure probability of contactor M1 given that contactor M2 failed.

Common cause failure probabilities are included in probabilistic risk assessment because analysts have long recognized that many factors, such as the poor maintenance practices indicated in the inspection report, which are not modeled explicitly in the models, can defeat redundancy or diversity and make failures of multiple similar components more likely than would be the case if these factors were absent. The effect of these factors on risk can be significant.

For practical reasons related to data availability, the common cause failure probabilities of similar components are estimated using data collected at the component level, without regard to failure cause.

Factors such as poor maintenance processes are often part of the environment in which the components are embedded and are not intrinsic properties of the components themselves. The NRC staff uses the failure memory approach in evaluating the significance of a performance deficiency. Observed failures are mapped into the probabilistic model, but successes are treated probabilistically. Thus, failure probabilities are left at their nominal values or are conditioned as necessary to reflect the details of the event.

To address this conditioning, the NRC staff has determined that there are three basic ground rules for treatment of common cause failure:

a. The shared cause is the deficiency identified in the inspection report which led to the observed equipment failure. In the case of the subject finding, the licensee's failure to identify the cause of the loose shading coils was the performance deficiency. The inspectors observed that at least one shading coil would easily come out of its recess on all contactors.

b. Common cause failures are of concern when they occur during the mission time of the probabilistic risk assessment, which for internal hazard groups is generally 24 hours. The common cause failure analysis methodology used and alpha vectors documented in the inspection report were developed to intrinsically incorporate this requirement into the common cause failure probabilities.

c. Credit for programmatic actions to mitigate common cause failure potential (staggering equipment modifications, etc.) should be applied qualitatively during the enforcement process and not incorporated into the numerical risk result. For the subject performance deficiency, this condition is moot. Inspection of components and records reviews indicated that all contactors had been handled in the same manner.

Therefore, the NRC concludes that the treatment of common cause failure probabilities for the reactor protection system contactors was appropriate and the conditional failure probability of the M1 contactor is best approximated as 3.59×10^{-2}/demand.

Item 4 – Higher Operator Reliability in Tripping the Reactor

Item 4a – Under Anticipated Transient Without Scram Conditions

Your staff indicated that follow-up operator actions, past the 10-minute point in the anticipated transient without scram (ATWS) scenario, should be credited. You provided an evaluation by Westinghouse of the expected Fort Calhoun Station plant response to this event. The evaluation indicated that, due to a large negative moderator temperature coefficient, power would automatically be reduced before the American Society of Mechanical Engineers (ASME) Level C pressure limit of 3200 psig was exceeded. This would indicate that further operator actions could be taken to trip the control rods without physical damage to key reactor components or systems.

NRC staff determined that the reactor response to a delayed tripping of the control rods in an ATWS scenario, especially the pressure response, is a critical aspect in preventing core damage. The details of the calculations and thermal-hydraulic runs of record are well established. NUREG-1780 states that pressure transients are unacceptable if the ASME Level C value of 3200 psig is exceeded. It further stated that a higher ASME service level was considered for

Babcock & Wilcox and Combustion Engineering plants, but was rejected on the basis that the reactor coolant system pressure boundary could deform to the point of inoperability.

Your evaluation showed a peak pressure of 3176 psia (approximately 3162 psig) during a run of the Combustion Engineering Nuclear Transient Simulator (CENTS) code. The NRC noted that similar thermal-hydraulic code runs, referenced in NUREG-1000 and NUREG-1780, were very sensitive to small variations or uncertainties in plant-specific parameters such as moderator temperature coefficient, reactor vessel volumes, and other physical parameters. Your analysis did not include sensitivities to variations or uncertainties in these parameters. For example, your analysis used the Fort Calhoun Station predicted beginning of life full power moderator temperature coefficient. However, you did not provide a sensitivity analysis for moderator temperature coefficient showing potential inaccuracies in this value or its variation with power. NUREG-1780 states that during the first part of the fuel cycle, below 100 percent power, the moderator temperature coefficient can be positive or insufficiently negative. If an ATWS occurs when the moderator temperature coefficient is either positive or insufficiently negative to limit reactor power, and the ATWS pressure increases, all subsequent mitigating functions are likely to be ineffective. NRC staff reviewed your predicted moderator temperature coefficient values over core life and at different power levels and concluded you also have positive or insufficiently negative values at lower powers.

It is the NRC's judgment that the 3176 psia outcome of your analysis is insufficient to assure the ASME Level C value is not actually exceeded, considering the potential inaccuracies and uncertainties of the analysis. Therefore, the NRC concluded the preliminary assessment time limitations for the ATWS response should still be used and no changes were made to the assessment for additional operator actions beyond 10 minutes.

<u>Item 4b - Manual Trip Probability</u>

Your staff pointed out that the failure of operators to push manual trip pushbutton No. 2 was not dependant on the success or failure of manual trip pushbutton No. 1. Based on your procedures the NRC staff concluded that, based on procedural guidance and operator training, the failure of operators to push manual trip pushbutton No. 2 would not likely be affected by the success or failure of manual trip pushbutton No. 1. Therefore, additional credit was given for the former probability under RPS-XHE-ERROR as shown in Table 1. However, the NRC did not use your suggested values (6×10^{-4}) for either manual pushbutton, as those values were based on additional time available to the operators in an ATWS scenario which the NRC staff determined should not be credited as discussed in Item 4a.

Summary

Table 1
Summary of Parameter Changes
Fort Calhoun Station Reactor Protector System Contactor Issue
Final Significance Determination

Parameter	Basic Event	SPAR Value	Preliminary Significance	Licensee Recommended	Final Significance
1 Shorter Exposure Time	N/A	N/A	64 days	32.5 days	64 days
2 Lower Failure Probability for Clutch Power Supply Breaker	RPS-BSN-FO-CBAB RPS-BSN-FO-CBCD	7.5×10^{-3}	7.5×10^{-3}	1.2×10^{-4}	3.81×10^{-4}
3 Common Cause Failure	RPS-RYT-CF-M12	2.4×10^{-6}	3.59×10^{-2}	2.4×10^{-6}	3.59×10^{-2}
3 Contactor Failure	RPS-RYT-CC-M1	1.2×10^{-4}	1.0	1.0	1.0
4a Operator Reliability Under ATWS Conditions (EOP-20)	N/A	N/A	N/A	1.4×10^{-3}	N/A
4b Manual Trip 1	RPS-XHE-XM-SCRAM	1×10^{-2}	1.5×10^{-3}	6.0×10^{-4}	1.5×10^{-3}
4b Manual Trip 2	RPS-XHE-ERROR	N/A	0.5	6.0×10^{-4}	6.0×10^{-3}

The NRC staff requantified the detailed model of the reactor protection system used in the preliminary significance determination using the modified parameters listed in Table 1. The revised internal change in core damage frequency was calculated to be 6.47×10^{-6}. Combining this with the external risk calculated in the preliminary determination the total change in core damage frequency was 7.14×10^{-6}.

The staff has considered the information you provided to the NRC regarding the significance of this issue and has concluded that the finding is appropriately characterized as being of low to moderate safety significance (White). The agency's preliminary evaluation, as documented in NRC Inspection Report 05000285/2011007, has been modified as shown above to reflect that the change in core damage frequency for the finding was 7.14×10^{-6} as compared with 2.6×10^{-5}.