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

May 29, 2007

EA-07-047

R. T. Ridenoure
Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION; NRC INSPECTION REPORT 05000285/2006018 - FORT CALHOUN STATION

Dear Mr. Ridenoure:

The purpose of this letter is to provide you the final results of our significance determination of the preliminary White finding identified in the subject inspection report. The NRC's final risk-informed conclusion is that the violation of an NRC requirement 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 February 13, 2007. The inspection finding was addressed using the Significance Determination Process and was preliminarily characterized as White, a finding with low to moderate increased importance to safety that may require additional NRC inspections. This finding involved the improper installation of the valve disk of Containment Spray Header Isolation Valve HCV-345. The improper installation resulted in a condition in which the actual position of the valve was nearly opposite of the indicated position.

This finding also represented a violation of NRC requirements. The violation involved the conduct of maintenance activities on valve HCV-345 without procedures or work instructions appropriate to the circumstances and the failure to have appropriate work instructions to conduct adequate post-maintenance testing prior to returning the valve to service. This violation resulted in an inoperable train of the containment spray system for an entire operating cycle and also provided a reactor coolant system diversion flow path if shutdown cooling (SDC) was initiated following certain postulated accident conditions.

The NRC's preliminary assessment of the safety significance of this inspection finding, which is documented in Attachment 2 of NRC Inspection Report 05000285/2006018 (ML070640155) resulted in an increase in core damage frequency (CDF) for internal events of 5.7E-6/year, or White for safety significance. The NRC's assessment of this issue was conducted using the

NRC's Risk-Informed Inspection Notebook for Fort Calhoun Station, the NRC's probabilistic risk assessment (PRA) Standardized Plant Analysis Risk (SPAR) model for Fort Calhoun Station, and the SPAR-H Human Reliability Analysis Method (NUREG/CR-6883). Our preliminary assessment included, but was not limited to, the following assumptions: Containment Spray Train B was functional with the internals of Valve HCV-345 installed incorrectly; the exposure time for the condition was 454 days; and consideration that if an accident occurred which involved operators initiating SDC prior to a recirculation actuation signal, then reactor coolant would be diverted from the reactor coolant system. The NRC and your staff agreed upon the risk models and the accident sequences. However, there were differences between your evaluation and the NRC's preliminary significance determination analysis in regard to performance shaping factors (PSFs) used in the PRA of this issue. Additionally, your data and analysis of the external initiators/events were not available for our evaluation at the time of the issuance of the report.

At the request of the Omaha Public Power District (OPPD), a Regulatory Conference was held on April 16, 2007, to discuss OPPD's position on the safety significance of the finding and corrective actions taken in response to the improperly installed valve disk. During the Regulatory Conference, OPPD agreed with the apparent violation as characterized in NRC Inspection Report 05000285/2006018, and your staff described the corrective actions taken in response to the finding. However, your staff asserted that the safety significance was very low, or at a Green level. Your staff's analysis and conclusions are included as an enclosure to the Regulatory Conference Meeting Summary (ML071160393), issued on April 26, 2007.

In response to questions from NRC staff, OPPD provided additional information on April 23, 2007. This information included: "HRA Review Comments and Recommendations on OPPD HCV-345 Mispositioning SDP, Revision 1," (Scientech letter SEA-JFG-07-005) and, "Additional Fire Perspectives for HCV-345 Containment Spray Valve" (ML071320002). The NRC staff considered the additional information provided by your staff in performing the final significance determination for this issue.

After careful consideration of the information developed during the inspection, the information your staff provided at the Regulatory Conference, and the revised information you provided on April 23, 2007, the NRC has concluded that the inspection finding is appropriately characterized as White. We estimate the change in core damage frequency associated with this condition to be 4.6E-6/year, as discussed in Enclosure 2 to this letter. The NRC staff agrees with most of your assumptions and analyses of the applicable accident scenarios. However, several differences account for the variances between OPPD's significance evaluation and the NRC's final analyses. These differences include the assumptions in the PSFs of three human error probabilities (HEPs), the method of performing a dependency analysis for the HEPs, and the treatment of the fire-induced loss of offsite power (LOOP) events multiplication factor.

Regarding the PSFs utilized in your analysis, the NRC determined that your evaluation gives more credit for successful operator actions than is considered reasonable based on guidance in NUREG/CR-6883. The NRC concluded that "Nominal" credit should be given to these PSFs based upon the complexity of the modeled event and the level of education and training of operators. In giving "Nominal" credit, the NRC acknowledges that recognizing a postulated reactor coolant system leak would not be difficult to perform and would have little ambiguity, and that your staff had an adequate amount of formal schooling and instruction to ensure they were proficient in day-to-day operations and had been exposed to abnormal conditions. Detailed descriptions of the NRC's conclusions regarding the PSFs are provided in Enclosure 2. For the HEP dependency analysis for adding additional high-pressure safety injection flow, your staff determined that there was "low" dependency for this HEP because of "additional cues" that were available. However, the NRC concluded there is a "moderate" dependency because crediting the reactor vessel level monitoring system as an additional cue may provide insufficient time to prevent core damage. For the external contributor of fire-induced LOOP events, your staff stated that a multiplication factor is not required. However, based on the guidance in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," the NRC staff concluded that the use of a multiplication factor of 10 is appropriate. The NRC concluded that these fire scenarios could involve events in which significant amounts of smoke would be generated, offsite fire fighting capability could be needed, and multiple concurrent activities affecting plant operators would be occurring. The NRC considered that this scenario is consistent with the NUREG/CR-6850 philosophy of increasing the HEP by a factor of 10 to account for a "minor" increase in operator workload.

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 improper installation of the Valve HCV-345 is a violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," as 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 TPGwynn for/

Bruce S. Mallett
Regional Administrator

Enclosures: (see next page)

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

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

cc w/enclosures:
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Maier - WAM
Chamberlain - DDC
Powers - DAP
Dricks - VLD
Carpenter - CAC
Ashley - MAB
Trocine - LXT
Fuller - KSF
Howell - ATH
Caniano - RJC1
Kirkland - JCK3
Hanna - JDH1
OEMAIL
Haire - MSH2

SUNSI Review Completed: _JAC__ADAMS: ✓ Yes☐ No Initials: _JAC___
✓ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ✓ Non-Sensitive

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RIV:RI:DRP/E	SRA:DRS	D:DRS	D:DRP	
JAClark	RLBywater	DDChamberlain	ATHowell	
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5/10/07	5/09/07	5/09/07	5/11/07	

ACES	D:ACES	OE	NRR	DRA	RA
MHaire	KFuller	DSolorio	MAshley	PGwynn	BMallett
gmv for	***/RA/***	***/RA/***	***/RA/***	***/RA/***	***/RA TPGwynn for/***
5/9 /07	5/14 /07	5/22 /07	5/18/07	5/29/07	5/29/07

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NOTICE OF VIOLATION

Omaha Public Power District
Fort Calhoun Station

Docket No. 50-285
License No. DPR-40
EA-07-047

During an NRC inspection completed on February 13, 2007, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, this violation is listed below:

> 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.
>
> Contrary to the above, in May of 2005, Fort Calhoun Station personnel accomplished maintenance activities without procedures appropriate to the circumstances. Specifically, the licensee performed maintenance and post-maintenance activities on Containment Spray Header Isolation Valve HCV-345 using procedures that were not appropriate to the circumstances because the procedures did not require actions to verify the correct orientation of the valve. As a result, the valve was installed in the wrong orientation during maintenance, and post-maintenance testing did not detect the improper reassembly prior to returning the valve to service. This failure caused one train of the Containment Spray system to be inoperable from May 11, 2005 to September 9, 2006.
>
> This violation is 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 20555-0001 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at 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-07-047" and should include: (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 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 29th day of May 2007

Final Significance Determination
Fort Calhoun Station
Containment Spray Header Isolation Valve HCV-345

The NRC reviewed the information provided by the licensee during the Regulatory Conference held on April 16, 2007. In response to questions from the staff, the licensee provided additional information on April 23, 2007. This information included: "HRA Review Comments and Recommendations on OPPD HCV-345 Mispositioning SDP, Revision 1," (Scientech letter SEA-JFG-07-005); and "Additional Fire Perspectives for HCV-345 Containment Spray Valve." Using the additional information provided by the licensee, the NRC staff performed a final significance determination by modifying the original evaluation, as appropriate. The documentation that follows is not a stand-alone evaluation; the reader must be familiar with the preliminary significance determination documented in NRC Inspection Report 05000285/2006018 (ADAMS ML070640155).

The NRC concluded that the preliminary significance determination (that the finding was of low-to-moderate safety significance) remained unchanged.

The following summarizes the NRC staff's review of the licensee's information provided at the Regulatory Conference and afterwards.

I. Internal Events:

a. The NRC staff reviewed the licensee's approach of binning the initiating events of interest into four categories. Category 3 (dry containment sump at time of shutdown cooling initiation) and Category 4 (wet containment sump at time of shutdown cooling initiation) were potential contributors to core damage.

   The NRC staff concluded the licensee's frequency estimates were reasonable. The frequency of Category 3 events was approximately 1.69E-2/year and the frequency of Category 4 events was approximately 1.06E-2/year.

b. The NRC staff reviewed the licensee's approach of developing event trees to evaluate the Category 3 and Category 4 events using the four top events: COGEARLY, COMCOG, HPSIFLOW, and XSPRAYVALVE. These top events were reasonable representations of operator response to indications of a flow diversion through the incorrectly positioned containment spray valve. They provided a framework for evaluation using the NRC's SPAR-H method of estimating human error probabilities (HEPs).

c. The NRC staff reviewed and performed independent evaluations of the licensee's HEP estimates of the four top events. A summary of this evaluation is included below.

COGEARLY

For top event COGEARLY (representing operator failure to detect and diagnose that a loss of coolant accident (LOCA) has been caused by flow diversion given a dry containment sump initially), the licensee's estimate of the the total HEP of COGEARLY was 1.1E-3. The NRC staff concluded this was a reasonable estimate.

COMCOG

For top event COMCOG (representing operator failure to detect and diagnose that a LOCA has been caused by the flow diversion given a wet containment sump) the licensee's estimate of the HEP was 5E-6. The NRC staff disagreed with the licensee's performance shaping factor (PSF) levels for Complexity and Experience/Training.

For Complexity, the licensee concluded that the PSF level was "Obvious Diagnosis" but the NRC staff concluded the PSF level was best represented as "Nominal." Using the guidance in NUREG/CR-6883, "The SPAR-H Human Reliability Analysis Method," the NRC concluded that determining a reactor coolant system leak had occurred was “not difficult to perform and had little ambiguity.” These are the criteria for a Nominal PSF level result. The NRC concluded the cues for a LOCA would not be so compelling in this context to warrant an order of magnitude reduction that a PSF level of "Obvious Diagnosis" would provide.

For Experience/Training, the licensee concluded that the PSF level was "High" but the NRC concluded the PSF level was best represented as "Nominal." The NRC concluded that operating crews had an adequate amount of formal training and instruction to ensure they were proficient in day-to-day operations and had been exposed to abnormal conditions. These are the criteria for a Nominal PSF level result. The "High" level was considered but required demonstrated master-level experience. Because of the extended duration of the postulated event, and the need to consider the performance capabilities of an average crew, Nominal was chosen.

The NRC concluded the HEP estimate for COMCOG was 1E-4, as shown below:

	COMCOG	
PSF	**Diagnosis Multiplier (Base = 0.01)**	**Action Multiplier (Base = 0.001)**
Available Time	Expansive Time (0.01)	N/A
Stress	High (2.0)	N/A
Complexity	Nominal (1.0)	N/A
Experience/Training	Nominal (1.0)	N/A

	COMCOG	
PSF	**Diagnosis Multiplier (Base = 0.01)**	**Action Multiplier (Base = 0.001)**
Procedures	Diagnostic/symptom oriented (0.5)	N/A
Ergonomics/HMI	Nominal (1.0)	N/A
Fitness for Duty	Nominal (1.0)	N/A
Work Processes	Nominal (1.0)	N/A
SUBTOTAL	1.0E-4	N/A
TOTAL	1E-4	

HPSIFLOW

For top event HPSIFLOW (representing operator failure to increase high-pressure safety injection flow on a loss of reactor coolant system inventory), the licensee's estimate of the HEP was 1.5E-5. For this HEP, the NRC staff agreed with the licensee's PSF level evaluation for Complexity in the Diagnosis component. In this context, the decision to add additional safety injection flow given that HPSI was already inservice was considered an "Obvious Diagnosis" level. But, the NRC disagreed with the licensee's PSF level for Experience/Training. In the NRC's view this was best represented as "Nominal" in both the Diagnosis and Action components of the HEP. The basis for this change was the same as described previously for COMCOG.

The NRC concluded the HEP estimate for HPSIFLOW was 3E-5, as shown below:

	HPSIFLOW	
PSF	**Diagnosis Multiplier (Base = 0.01)**	**Action Multiplier (Base = 0.001)**
Available Time	Expansive Time (0.01)	>=50x time required (0.01)
Stress	High (2.0)	High (2.0)
Complexity	Obvious Diagnosis (0.1)	Nominal (1.0)
Experience/Training	Nominal (1.0)	Nominal (1.0)
Procedures	Diagnostic/symptom oriented (0.5)	Nominal (1.0)
Ergonomics/HMI	Nominal (1.0)	Nominal (1.0)
Fitness for Duty	Nominal (1.0)	Nominal (1.0)
Work Processes	Nominal (1.0)	Nominal (1.0)
SUBTOTAL	1.0E-5	2.0E-5

	HPSIFLOW	
PSF	**Diagnosis Multiplier (Base = 0.01)**	**Action Multiplier (Base = 0.001)**
TOTAL	3E-5	

XSPRAYVALVE

For top event XSPRAYVALVE (representing operator failure to isolate the flow diversion path), the licensee's estimate of the HEP was 4.1E-4. The NRC disagreed with the licensee's PSF level for Complexity in the Diagnosis component of the HEP. The NRC concluded the licensee's selected PSF level of "Obvious Diagnosis" was overly optimistic and was best represented as "Nominal."

The NRC concluded the HEP estimate for XSPRAYVALVE was 5E-4, as shown below:

	XSPRAYVALVE	
PSF	**Diagnosis Multiplier (Base = 0.01)**	**Action Multiplier (Base = 0.001)**
Available Time	Expansive Time (0.01)	>=50x time required (0.01)
Stress	High (2.0)	High (2.0)
Complexity	Nominal (1.0)	Nominal (1.0)
Experience/Training	Nominal (1.0)	Nominal (1.0)
Procedures	Diagnostic/symptom oriented (0.5)	Incomplete (20.0)
Ergonomics/HMI	Nominal (1.0)	Nominal (1.0)
Fitness for Duty	Nominal (1.0)	Nominal (1.0)
Work Processes	Nominal (1.0)	Nominal (1.0)
SUBTOTAL	1.0E-4	4.0E-4
TOTAL	5E-4	

d. The NRC staff reviewed the licensee's method of performing a dependency analysis for the HEPs and disagreed with the licensee's dependency decision tree evaluation for HPSIFLOW. The NRC determined that "Additional Cues" should not be credited. The licensee credited the shift technical advisor periodically checking the reactor vessel level monitoring system (RVLMS) as an additional cue; but, the NRC concluded that using the RVLMS may not allow sufficient time for action between indication of a lowering water level and core damage. Therefore, the NRC's dependency decision tree evaluation resulted in "moderate" dependency rather than "low" dependency. This resulted in a conditional HEP of 0.14 for HPSIFLOW following failure to isolate the shutdown cooling flow diversion rather than 0.05.

Based on the above review, the NRC staff estimated that the Joint HEP for Category 3 events was 1.51E-7 and that the Joint HEP for Category 4 events was 1.37E-4.

The increase in core damage frequency for each category of initiating event of concern can be expressed as:

Category 3: 1.69E-2/year * 1.51E-7 = 2.55E-9/year

Category 4: 1.06E-2/year * 1.37E-4 = 1.45E-6/year

Therefore, Category 3 events had negligible contribution and the increase in core damage frequency due to internal events was estimated as that from Category 4 events alone.

II. External Initiators

a. The NRC staff reviewed the licensee's approach for evaluating the contribution to risk significance of the finding due to external initiating events and concluded that only fire events were potentially significant.

b. The licensee referenced use of NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," for estimating HEPs following fire initiating events. With respect to the events involving a fire-induced loss of component cooling water, the licensee identified four fire scenarios of interest. Two of these involved fires in the auxiliary building and two involved fires in the control room. For each of the fire scenarios, the licensee multiplied the joint HEP for these events by a factor of 10. NUREG/CR-6850 recommends applying this factor to HEPs from an internal events PRA when a specific set of conditions are met. The multiplication factor is to account for the effects of potential fire brigade interaction and other minor increased workload and/or distraction issues. The NRC noted that using this factor for control room fires was a nonconservative application. The NUREG states that the HEP should be set to 0.1 or multiplied by a factor of 10, whichever is greater, based upon the complexity. Control room fires may result in control room evacuation or other significant additional complications that may require additional detailed analysis, and would indicate setting the HEP valve to 0.1 is correct. However, the frequency of loss of reactor coolant pump seal cooling events resulting from a control room fire-induced loss of component cooling water was very low (3.93E-6/year). Even if the 0.1 value were used, then the change in risk would be less than 3.93E-7/year. Therefore, the NRC concluded that the contribution of these fire scenarios to the overall risk significance would be very small. As a result, the NRC still agreed with the licensee's overall conclusion that the significance contribution from loss of component cooling water events resulting from fire was still very small.

c. With respect to the events involving a fire-induced loss of offsite power (LOOP), the licensee stated that use of the NUREG/CR-6850 multiplication factor of 10 was not required. The fire scenarios of concern that would cause a LOOP event are all fires involving transformers outside the plant in the transformer yard. The licensee stated that because these initiating events occurred outside the plant and several hours would elapse before operators would initiate shutdown cooling, a factor of 10 increase in the HEP was not necessary. The NRC staff disagreed with this conclusion and believed that a factor of 10 increase in the HEP was still appropriate. These fire scenarios would typically be major events involving significant amounts of smoke, possibly involving the response of offsite fire fighters, and a coincident LOOP with operators performing a plant cooldown and depressurization using emergency diesel generators. The NRC considered that this scenario was consistent, at a minimum, with the NUREG/CR-6850 philosophy of increasing the HEP by a factor of 10 to account for a <u>minor</u> increase in operator workload. Therefore, using the licensee's frequency of a fire-induced LOOP of 2.31E-3/year, and the NRC's revised joint HEP multiplied by a factor of 10 (1.37E-4 * 10 = 1.37E-3), the increase in core damage frequency resulting from fire-induced LOOP scenarios was:

2.31E-3/year * 1.37E-3 = 3.16E-6/year

III. Final Significance Determination

The overall safety significance of a performance deficiency with respect to core damage frequency is expressed as the summation of the increase in core damage frequency from internal and external initiating events. Therefore, the total increase in core damage frequency is estimated as: 1.45E-6/year (Internal Events) + 3.16E-6/year (External Events) = 4.61E-6/year. This result is of low-to-moderate safety significance (White).