Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

LIC-12-0160
October 27, 2012

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Reference: Docket No. 50-285

Subject: Licensee Event Report 2012-004, Revision 2, for the Fort Calhoun Station

Please find attached Licensee Event Report 2012-004, Revision 2, dated October 27, 2012. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B).

No commitments are being made in this letter.

If you should have any questions, please contact me.

Sincerely,

Louis P. Cortopassi
Vice President and CNO

LPC/rjr

Attachment
c: E. E. Collins, Jr., NRC Regional Administrator, Region IV
L. E. Wilkins, NRC Project Manager
J. C. Kirkland, NRC Senior Resident Inspector
INPO Records Center

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Fort Calhoun Station	05000285	1 OF 4

4. TITLE
Inadequate Analysis of Drift Affects Safety Related Equipment

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	29	2012	2012	- 004 -	2	10	27	2012	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 5

10. POWER LEVEL: 0

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Erick Matzke	402-533-6855

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

While investigating industry operating experience, it was determined that Fort Calhoun Station is subject to similar conditions where Static "0" Ring pressure switches with certain housing styles exhibit a setpoint shift when exposed to a change in temperature if the switch body is not vented. Fort Calhoun Station pressure switches that provide signals for high containment pressure to the reactor protection system and engineered safeguards actuation circuitry may have this configuration. The impact of the potential drift was evaluated and it was initially determined that neither reactor protection system nor the engineered safeguard circuitry may actuate at the required containment pressure of 5 psig. A subsequent evaluation of actual data concluded that safety analysis limits were not exceeded. However, two Technical Specification limits were not protected by the calibration procedure nominal trip setpoint when applying the additional uncertainty.

The Apparent Cause was determined to be poor vendor documentation which led to Engineering personnel to improperly interpret and apply the information contained in the Static "O" Ring vendor manual. Corrective actions were initiated to remove the vent caps, revise the affected calculations to the temperature correction factor and drift. Additional actions to revise and re-perform surveillance testing were initiated.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Fort Calhoun Station	05000285	2012	- 004 -	2	2 OF 4

NARRATIVE

EVENT DESCRIPTION

While investigating industry operating experience, it was determined that Fort Calhoun Station (FCS) is subject to similar conditions. The operating experience documented a condition where Static "0" Ring (SOR) pressure switches with housing styles N6 or RT and conduit seal option JJ, exhibit a setpoint shift when exposed to a change in temperature if the switch body is not vented.

Engineering performed an equipment database search to determine where the switches in question were installed at FCS. Several switches that input into specific Reactor Protection System (RPS) and Engineered Safeguards Actuation Circuitry (ESF) loops were in the affected population. Specifically these pressure switches provide safety-related signals for high containment pressure to the RPS and ESF.

Field inspection revealed that the vent plugs were installed on the subject SOR switches and therefore the temperature effect is applicable. No other interim actions were required at the time of discovery as the affected switches are not required in the current plant mode.

The impact of the potential drift was evaluated and it was determined that neither RPS nor the engineered safeguard circuitry may actuate at the required containment pressure of 5 psig. A preliminary evaluation determined that the actuation may occur at a slightly higher value than the required pressure. A subsequent analysis of actual data concluded that safety analysis limits were not exceeded and that only two TS Limits were not protected by the calibration procedure nominal trip setpoint when applying the additional uncertainty.

On May 2, 2012, an eight (8)-hour report was made per 10 CFR 50.72(b)(3)(ii)(B) to the NRC Headquarters Operation Office (HOO) at 1802 CDT (Event Number (EN) 47892) as an unanalyzed condition, which was retracted on October 16, 2012. As a result of the analysis discussed above, this LER is amending the previous reporting criteria. This report is being made in accordance with the requirements of 10 CFR 50.73(a)(2)(i)(B). The reporting requirements of 10 CRF 50.73(a)(2)(v) and 10 CFR 50.73(a)(2)(ix)(A) have been deleted.

CONCLUSION

The condition was initially determined to be reportable under 10 CFR50 72(b)(3)(ii)(B), Unanalyzed Condition, based on a conservative assumption that the error introduced violated not only the Technical Specification (TS) limits on Tables 2-1 and 2-11 (5.0 psig) but also the safety analysis limit of 5.4 psig, Updates Safety Analysis Repot (FSAR) Table 14.1-1. A subsequent evaluation of actual data concluded that safety analysis limits were not exceeded and that two TS Limits were not protected by the calibration procedure nominal trip setpoint when applying the additional uncertainty.

The cause was determined to be poor vendor documentation which led Engineering personnel to improperly interpret and apply the information contained in the SOR vendor manual.

CORRECTIVE ACTIONS

FCS is currently conducting a Causal Analysis as part of the Restart Checklist Item 3.d.1. The results of this analysis will be provided separately to the NRC as part of the Confirmatory Action Letter (CAL) closure process.

Corrective actions specific to this condition were initiated which include the following:

NARRATIVE

1. For SOR switches used in safety related applications for which Uncertainty Calculations exist,
   a. Revise the associated calculations to include the temperature correction factor and drift as specified in the vendor manual,
   b. Revise the associated surveillance tests based on the updated Uncertainty Calculations, and
   c. Update the Equipment Database information as needed.
2. For SOR switches used in safety related applications for which Uncertainty Calculations do not exist and the evaluation was indeterminate, perform an analysis to evaluate the impact of temperature and drift on the setpoints and implement recommendations from the analysis.
3. For the non-safety related switches, revise the affected calculation or modify the switches to install breather drains, revise and perform the affected surveillance tests, and update the Equipment Database information as necessary.

SAFETY SIGNIFICANCE

Both the actual nuclear safety impact and the potential nuclear safety impact for this event are low based on the following considerations.

The affected pressure switches provide safety related signals for high containment pressure to the reactor protection system (RPS) and engineered safeguards actuation system (ESFAS) circuitry. Specifically, the switches perform the following functions as specified in the FCS Technical Specifications:

- Technical Specification Table 2-11, RPS Limiting Safety System Settings, item 7, High Containment Pressure s 5 psig;

  Reactor Trip

- Technical Specification Table 2-1, Engineered Safety Features System Initiation Instrument Setting Limits Item 1, High Containment Pressure s 5 psig;

  Safety Injection, Containment Spray, Containment Isolation, Containment Air Cooler- DBA Mode, and Steam Generator Isolation

While the Technical Specification limit is 5 psig, a review of the FCS USAR shows that the setpoint value used in the accident analysis is 5.4 psig. This is shown in Table 14.1-1- Reactor Protective System Trips and Safety Injection for Safety Analyses Setpoints which identifies a Containment Pressure High Setpoint of 5.0 psig and a Safety Analysis Setpoint 5.4 psig. This specific actuation feature is credited only in USAR 14.6, "Containment Pressure Analysis", which credits the switches for initiation of Engineered Safety Features at a setpoint of 5.4 psig.

Further analysis was performed to determine the worst case installed setpoint value when the temperature effect is included. This was done by adding the temperature effect associated with a 30°F temperature shift to the worst case "as found" value from the historical calibration data and verifying that value is less than the value used in the accident analysis. From this it was determined that the worst case installed setpoint was less than the value of 5.4 psig used in the safety analysis. This demonstrates that while the trip setpoint used in the calibration procedure may not be adequate to protect the Technical Specification limits, the safety limit used in the accident analysis was not exceeded and therefore this condition did not impact Nuclear Safety.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Fort Calhoun Station	05000285	2012	- 004 -	2	4 OF 4

NARRATIVE

SAFETY SYSTEM FUNCTIONAL FAILURE

This event does result in a safety system functional failure in accordance with NEI-99-02.

PREVIOUS EVENTS

No events of a similar nature have been identified.