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

July 27, 2011
LIC-11-0080

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

Reference: Docket No. 50-285

Subject: Licensee Event Report 2011-002, Revision 1, for the Fort Calhoun Station

Please find attached Licensee Event Report 2011-002, Revision 1, dated, July 27, 2011. This report is being submitted pursuant to 10CFR50.73(a)(2)(i)(B). If you should have any questions, please contact me.

Sincerely,

Jeffrey A. Reinhart
Site Vice President

JAR/rda

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)		Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are inco rporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Priv acy Section (T-5 F53), U.S. Nuclear Regulator y Commission, Washington, DC 205 55-0001, or b y internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sp onsor, and a person is not required to respond to, th e information collection.

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4. TITLE

Failure of an RPS Trip Unit

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
2	4	2011	2011	- 002 -	1	7	27	2011	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

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)*
Richard Acker	402-533-6561

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)*

On November 29, 2010, during the performance of a work order, voltage at reactor protective system (RPS) connection T-74 was found 39 millivolt (mV) higher than connection T-17 (RPS ground). The allowed limit is 4 mV. T-74 is the signal common lead for steam generator (SG) pressure channels 902 and 905 inputs to Trip Unit 6 (Low SG Pressure) and Trip Unit 7 (Asymmetric SG Transient). Further investigation determined that the affected channels should have been declared inoperable. With a channel of RPS inoperable the appropriate section of Technical Specifications should have been entered. The TS LCO action times were not met.

The root causes are determined to be the following:

1. Management has not effectively enforced expectations of rigorous troubleshooting standards for equipment important to safety and/or operation.

2. The standards for implementation of the Corrective Action Program have been ineffective in identifying and driving resolution of repeat and less-significant failures of equipment important to safety and/or operation.

The wire between terminals T-74 and relay contact terminal 12 was replaced.

NARRATIVE

BACKGROUND

The Fort Calhoun Station (FCS) reactor protective system (RPS) is designed to rapidly shut down the nuclear chain reaction prior to reaching a condition that could damage the reactor core. The RPS generates a reactor trip signal, which releases the control element assemblies and allows the control rods to fall into the core.

The RPS protects the plant from abnormal events that have the potential to threaten safe operation and is designed in accordance with the proposed Institute of Electrical and Electronics Engineers (IEEE) 279-1968, Criteria for Protection Systems for Nuclear Power Generating Stations. The RPS comprises four independent instrumentation channels. Each channel has a separate vital power supply with parallel cable raceways. Two out of four logic is used to initiate a trip to ensure a single failure will neither cause nor prevent system operation. Each channel monitors 12 safety parameters; each parameter input is derived from an isolated instrument channel. Each parameter operates a two out of four coincidence logic matrix to maintain or remove power from the control element drive mechanism (CEDM) clutches. Individual channel trips occur when the measurement reaches a preselected value. The channel trips are combined in six two out of four matrices. Each individual measurement channel has inputs to three of the six logic matrices. The logic matrix trip relays are deenergized when two channels of the same measurement channel trip. Each of the two out of four logic matrices provides a trip signal to the interposing relays which in turn causes a direct trip of the contactors in the AC supply to the CEDM clutch power supplies. Any one of the six logic matrices will deenergize the four clutch power supplies. The logic matrices are arranged in a one out of six logic configuration. The clutch power supply DC outputs are ungrounded.

FCS Technical Specifications (TS) 2.15 requires:
The operability, permissible bypass, and Test Maintenance and Inoperable bypass specifications of the plant instrument and control systems shall be in accordance with Tables 2-2 through 2-5.

(1) In the event the number of channels of a particular system in service falls one below the total number of installed channels, the inoperable channel shall be placed in either the bypassed or tripped condition within one hour if the channel is equipped with a bypass switch, and eight hours if jumpers or blocks must be installed in the control circuitry. The inoperable channel may be bypassed for up to 48 hours from time of discovering loss of operability; however, if the inoperability is determined to be the result of malfunctioning RTDs or nuclear detectors supplying signals to the high power level, thermal margin/low pressurizer pressure, and axial power distribution channels, these channels may be bypassed for up to 7 days from time of discovering loss of operability. If the inoperable channel is not restored to OPERABLE status after the allowable time for bypass, it shall be placed in the tripped position or, in the case of malfunctioning RTDs or linear power nuclear detectors, the reactor shall be placed in hot shutdown within 12 hours. If active maintenance and/or surveillance testing is being performed to return a channel to active service or to establish operability, the channel may be bypassed during the period of active maintenance and/or surveillance testing. This specification applies to the high rate trip-wide range log channel when the plant is at or above 10-4% power and is operating below 15% of rated power.

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(2) In the event the number of channels of a particular system in service falls to the limits given in the column entitled "Minimum Operable Channels," one of the inoperable channels must be placed in the tripped position or low level actuation permissive position for the auxiliary feedwater system within one hour, if the channel is equipped with a bypass switch, and within eight hours if jumpers or blocks are required; however, if minimum operable channel conditions for SIRW [Safety Injection Refueling Water] tank low signal are reached, both inoperable channels must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If at least one inoperable channel has not been restored to OPERABLE status after 48 hours from time of discovering loss of operability, the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the containment ventilation isolation valves are closed.

EVENT DESCRIPTION

On November 29, 2010, during the performance of a work order, voltage at RPS connection T-74 was found 39 millivolt (mV) higher than connection T-17 (RPS ground). The allowed voltage difference is 4 millivolt (mV). T-74 is the signal common lead for steam generator (SG) pressure channels 902 and 905 which are inputs to Trip Unit 6 (Low SG Pressure) and Trip Unit 7 (ASGT). T-74 is electrically connected to the Thermal Margin/Low Pressure (TM/LP) calculator common T-17 (RPS ground) through a relay. Since T-74 and T-17 are normally connected through closed relay contacts, the electric potential between the terminal points remains zero so long as there is electrical continuity.

Electrical continuity between T-74 and T-17 was assumed following the installation of the Asymmetric SG Transient Trip (ASGT) modification MR-FC-83-44 in 1984. T-74 and T-17 need to be at nearly the same potential to maintain SG Low Pressure and ASGT trip setpoint calibration accuracy.

Multiple occurrences of poor electrical continuity (e.g. potential difference) between TM/LP calculator AI-31A-AW12 points T-74 and T-17 have been documented since 1993. Over several years it was assumed that the relay contacts or terminal points had degraded. In response to condition report (CR) 1998-0443, a quarterly surveillance test (IC-ST-RPS-0018, Quarterly Functional Test of Steam Generator Low Pressure and Asymmetric Steam Generator Transient RPS Bistable Trip Units) was revised to verify appropriate value of signal common. A refueling cycle surveillance test (IC-ST-RPS-0044, Calibration of Steam Generator Low Pressure Trip Unit A/TU-6 and Asymmetrical Steam Generator Transient Trip Unit A/TU-7) checked for proper functioning of the circuit. CR 2010-5667 describes an out-of-tolerance (OOT) condition found on November 8, 2010, which resulted in daily measurement of the voltage difference between T-74 and T-17. On November 29, 2010, during the performance of the daily measurement, a 39 millivolt difference was measured between T-74 and T-17. During subsequent troubleshooting personnel discovered intermittent continuity in the wire between T-74 and relay contact terminal 12.

The wire between terminals T-74 and relay contact terminal 12 was replaced. The suspected section of wire was removed for examination. Initial failure analysis has proved inconclusive. No failure or degradation of the wire has been detected. Further investigation determined that the affected channels should have been declared inoperable. With a channel of RPS inoperable TS 2.15 should have been entered. The TS LCO action times were not met. This report is being made in accordance with 10 CFR 50. 73(a)(2)(i)(B).

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NARRATIVE

CONCLUSION

The root causes are determined to be:

1. Management has not effectively enforced expectations of rigorous troubleshooting standards for equipment important to safety and/or operation.
2. The standards for implementation of the Corrective Action Program have been ineffective in identifying and driving resolution of repeat and less-significant failures of equipment important to safety and/or operation.

There are an additional six contributing causes identified as well.

CORRECTIVE ACTIONS

The wire between terminals T-74 and relay contact terminal 12 was replaced. Additional corrective actions are identified in the condition reporting system. Corrective actions include: training on the correct use of Operating Experience (OE) in work orders; observations for each maintenance discipline of routine maintenance activities to ensure/assess that personnel are rigorously following procedures and work order instructions will be performed and; System Health Reports to document equipment failures for the period and repeat equipment failures within the past five years will be revised. Additionally, trending of these indicators will be performed.

SAFETY SIGNIFICANCE

The overall effect on the one SG Low Pressure trip setpoint was less than 10 pounds per square inch (psi) at 39 millivolts and not more than 30 psi at the highest ever recorded difference of 120 millivolts recorded on May 14, 2009.

Even with a failed instrument channel, the margin of safety as defined in the basis of the FCS TS was maintained. The plant was able to trip on Low SG pressure in, at least, a two out of two reactor protection logic. Even with these RPS channels inoperable, the single failure criteria for the RPS system was met. Therefore the health and safety of the public was not impacted by this RPS failure.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event does not result in a safety system functional failure in accordance with NEI-99-02.

PREVIOUS EVENTS:

No previous LER has documented problems with this RPS circuitry.