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

10 CFR 50.73

LIC-15-0090
August 3, 2015

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

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

Subject: Licensee Event Report 2015-004, Revision 0, for the Fort Calhoun Station

Please find attached Licensee Event Report 2015-004, Revision 0. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) and 50.73(a)(2)(v)(B). There are no new commitments being made in this letter.

If you should have any questions, please contact Terrence W. Simpkin, Manager, Site Regulatory Assurance, at (402) 533-6263.

Sincerely,

Louis P. Cortopassi
Site Vice President and CNO

LPC/epm

Attachment
c: M. L. Dapas, NRC Regional Administrator, Region IV
C. F. Lyon, NRC Senior Project Manager
S.M. Schneider, NRC Senior Resident Inspector

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by 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 sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE: Inoperability of Auxiliary Feedwater Trains due to Failure of Steam Generator Isolation Valve

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
06	05	2015	2015	004	- 00	08	03	2015		05000
									FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 4

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

LICENSEE CONTACT	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
B	BA	ISV	FISHCOCO	Y					

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 June 5, 2015, at approximately 1330 Central Daylight Time (CDT), during startup HCV-1107A (Steam Generator (SG) RC-2A Auxiliary Feedwater (AFW) Inlet Valve) was declared inoperable when it failed to move during testing. During the investigation HCV-1108A (SG RC-2B AFW Inlet Valve) was found to hesitate during operation. The valve sealing materials had been replaced during the refueling outage with materials that were not appropriate for the service conditions.

A cause determination determined that the original valve specification for HCV-1107A was not appropriate for the plant application.

HCV-1107A and HCV-1108A were repaired using the original materials. An operability evaluation was completed for use of the original materials for one operating cycle. An engineering solution to the issue will be completed prior to startup from the next refueling outage.

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

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NARRATIVE

BACKGROUND

Fort Calhoun Station (FCS) is a two-loop reactor coolant system (RCS) of Combustion Engineering design.

The Auxiliary Feedwater (AFW) system uses a steam driven feedwater (FW) pump (FW-10) and an electrically powered feedwater pump (FW-6). The system injects through a common header to four isolation valves two (2) in series for each SG. HCV-1107A (Steam Generator (SG) RC-2A AFW Inlet Valve) and HCV-1108A (SG RC-2B AFW Inlet Valve) inside containment and HCV-1107B (SG RC-2A AFW Inlet Valve) and HCV-1108B (SG RC-2B AFW Inlet Valve) outside containment. The HCV-1107 valves direct flow to the ‘A’ SG. The HCV-1108 valves direct flow to the ‘B’ SG.

EVENT DESCRIPTION

On June 5, 2015, during startup from the refueling outage (FCR27), preparations were made for a full flow test of AFW pump FW-10 per OP-ST-AFW-3011 (AFW Pump FW-10, Steam Isolation Valve, and Check Valve Tests) and OI-AFW-4, (AFW Startup and System Operation), Attachment 1 (FW-10 Turbine-Driven AFW Pump Operations). As part of the valve lineup to establish the flow path from FW-10 to the SG, HCV-1107A was to be opened from the control room. After taking the control switch to open, the valve’s open position indication light did not energize to confirm the valve had opened. An operator was dispatched to containment to investigate. The operator found the valve was still closed. After a second attempt to open the valve, the operator reported hearing air vent from the valve actuator, but no movement of the valve was observed. HCV-1107A was declared inoperable on June 5, 2015, at 1330 Central Daylight Time (CDT). This event was documented in Condition Report (CR) 2015-07564.

Operators were dispatched to HCV-1108A to observe the valve locally while the control room cycled the valve. The valve was observed to fully open and close, but, with noticeable hesitation.

At 1822 CDT on June 5, 2015, the station notified the NRC Headquarters Operations Office (HOO) of the inoperability of both trains of the AFW system per 10 CFR 50.72(b)(3)(v)(B). This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) and 50.73(a)(2)(v)(B).

CAUSE OF THE EVENT

In 2013, CR 2013-19342 identified issues regarding Polytetrafluoroethylene or PTFE, commonly known as Teflon®, use in a radiation environment. An Operability Evaluation (OpEval) was performed in support of CR 2013-19342 regarding the operability of HCV-1107A and HCV-1108A. The OpEval found the valves to be operable but specified the replacement of the PTFE seal with more radiation resistant material. Engineering Analysis (EA) 13-040, “Evaluation of Valves with Teflon Subcomponents located in Radiation Areas,” evaluated the radiation resistance of HCV-1107A and HCV-1108A with respect to the PTFE low-friction seal but failed to evaluate the seal rings temperature capabilities.

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An Item Equivalency Evaluation (IEE) authorized installation of an alternate material (Chlorotrifluoroethylene or CTFE, commonly known as KEL-F®) for the internal valve seal that had a lower rated temperature (390 degrees F) than the previously used material (PTFE) (500 degrees F). The seal change-out was performed for both HCV-1107A and HCV-1108A inboard isolation valves. The cold stroke of these valves was successful prior to RCS temperature exceeding 300 degrees F.

The station proceeded with reactor coolant heat up. Subsequently, when performing testing approximately four days after heating up, HCV-1107A did not open, and HCV-1108A opened within the required time but did not actuate smoothly. Thermography and inspection of HCV-1107A and HCV-1108A determined the following: The qualified temperature for the sealing material, CTFE, had been exceeded for HCV-1107A and the sealing material was significantly degraded resulting in failure of the valve to operate. Although HCV-1108 was at a lower temperature, slight degradation of the sealing material was noted which likely resulted in the observed behavior of the valve noted by the operator.

During initial plant design, circa 1967 to 1969, a design temperature requirement of 550 and 600 degrees F was developed for AFW piping, yet the Architect/Engineer (A/E) specified a service requirement of 70 degrees F to 120 degrees F for the valve manufacturer. FCS failed to recognize the improper valve specification. No Engineering Analysis (EA) was identified that reconciled the differences between the valve's temperature specification and the piping temperature specification. Based on the A/E specifying service requirement of 70 degrees F to 120 degrees F, the valve manufacturer provided a valve design with a maximum service temperature of 180 degrees F, limited by the Nitrile valve seals. This exceeded the requirement specified; therefore, the A/E and FCS accepted the manufacturer-recommended valve. The consequence of this condition caused the valves to have less margin for soft material failures due to high temperatures and caused the valve seals to become degraded.

Therefore the station determined that the root cause of this event was:

The original valve specification for HCV-1107A or HCV-1108A was not appropriate for the plant application.

CORRECTIVE ACTIONS

Short Term Corrective Actions

The station was cooled down to allow repair of HCV-1107A and HCV-1108A. The station concluded O-rings and seals needed to be replaced. HCV-1107A and HCV-1108A valves were rebuilt to restore the valve configuration to pre-FCR27 configuration (PTFE seals). An Operability Evaluation (15-007) was completed under CR 2015-07613 which determined that HCV-1107A/1108A were operable for one operating cycle with PTFE valve seals.

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Long Term Corrective Actions

FCS will implement an engineering solution, analysis, and/or modification as required, including a revision to the valve specification such that HCV-1107A and HCV-1108A meet all applicable design requirements and operating conditions. This action is to be completed prior to startup from the next Refueling Outage (FCR28). HCV-1107B/1108B are outside containment and are unaffected by the same temperature or radiation issues.

SAFETY SIGNIFICANCE

The AFW system operates in conjunction with the Main Feedwater (MFW) system to remove decay heat from the RCS. At the time of the event, decay heat was being removed via the MFW flow path. The issues with the HCV-1107A and HCV-1108A (AFW flow path) were discovered during testing and did not affect the MFW flow path. The decay heat rate was very low since the plant was returning from a refueling outage. Had there been a loss of decay heat removal via the MFW flow path, there would have been ample time for compensatory actions.

SAFETY SYSTEM FUNCTIONAL FAILURE

This does represent a safety system functional failure in accordance with NEI 99-02, Revision 7.

PREVIOUS EVENTS

No previous LERs.