Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

LIC-07-0021
March 29, 2007

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Subject: Licensee Event Report 2007-001 Revision 0 for the Fort Calhoun Station

Please find attached Licensee Event Report 2007-001, Revision 0, dated March 29, 2007. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B). If you should have any questions, please contact me. The attached LER includes the following regulatory commitment:

Appropriate procedures will be revised to ensure that MS-100, MS-101, MS-102 and MS-103 remain open to so that RM-064 is not isolated during plant startup activities. The procedure revisions will be completed by May 31, 2007.

Sincerely,

Jeffrey A. Reinhart
Site Director
Fort Calhoun Station

JAR/epm

Attachment

c: B. S. Mallett, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. D. Hanna, NRC Senior Resident Inspector
INPO Records Center

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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.

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
Inoperability of a Post Accident Main Steam Line Radiation Monitor Due to Procedural Error

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
02	01	2007	2007	- 001 -	00	03	29	2007	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)
Erick Matzke, Compliance Engineer	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)*

On February 1, 2007, a station employee, who was cleaning, identified that all four manual steam isolation valves for RM-064 were closed. RM-064 is the post-accident main steam line radiation monitor. Investigation determined that RM-064 had most probably been isolated since November 29, 2006. This condition is in violation of the Fort Calhoun Station technical specifications.

The root cause of this event was that a procedure was written assuming its performance prior to completing the main steam system valve lineup. The procedural guidance closed the isolation valves which is contrary to the previously completed main steam system valve lineup.

The isolation valves were opened and the appropriate procedures will be corrected.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

BACKGROUND

Post-accident main steam line monitor, RM-064, (see Figure 1 (RE-064 is the element that contains RM-064)) is an off-line sampler designed to monitor activity in the main steam system by sampling the main steam headers. Since RM-064 was designed for post-accident indication only, it does not initiate any system actuations or any alert/high alarms. The sample isolation valves for the monitor are opened manually from the control room.

During a steam generator tube rupture event, RM-064 is placed in service to assist in quantifying the effluent release rate through atmospheric dump valve HCV-1040, the main steam safety valves, or the turbine-driven auxiliary feedwater pump (FW-10). The respective abnormal/emergency operating procedure or operating instruction OI-RM-1, "Radiation Monitoring," provides the guidance for placing RM-064 in service.

EVENT DESCRIPTION

On February 1, Fort Calhoun Station (FCS) was operating at 100 percent power. A station employee, who was cleaning in room 81, identified that all four manual steam isolation valves for radiation monitor RM-064, (MS-100, MS-101, MS-102, and MS-103) were closed. These valves are normally open and function to supply steam from steam generators RC-2A and RC-2B, via HCV-921 and HCV-922, to RM-064 for post accident monitoring. When directed by the applicable Abnormal Operating Procedure (AOP) or Emergency Operating Procedure (EOP) to sample the steam for radioactivity, isolation valves HCV-921 and HCV-922 would be opened by the control room operators to sample the main steam lines for radioactivity.

At the time of discovery, the control room was immediately contacted and the control room supervisor directed the valves to be opened. Upon opening them, the turbine building operator identified a steam leak from a fitting above steam line isolation valve MS-103; all four steam supply isolation valves were then closed. Compensatory actions for post accident monitoring per procedure Emergency Plan Implementing Procedure (EPIP)-EOF-6, "Dose Assessment," were instituted, as directed by procedure OI-RM-1 until the steam leak could be repaired. The repairs and post maintenance testing were completed for the leaking fitting on February 6, 2007. RM-064 was returned to service and declared operable.

Technical Specification (TS) 2.21, table 2-10, instrument 3 provides post-accident monitoring instrumentation operating limits for RM-064 in modes 1, 2, and 3. The applicable portions of the TS state:

> With the number of OPERABLE channels less than the required minimum channels operable requirements, initiate the pre-planned alternate method of monitoring the appropriate parameter(s) within 72 hours, and
>
> 1. either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or
>
> 2. prepare and submit a special report to the Commission pursuant to specification 5.9.3 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedules for restoring the system to OPERABLE status.

RM-064 was discovered isolated on February 1, 2007, and through investigation it appears that RM-064 has been isolated since November 29, 2006. This event is reportable per 10CFR50.73(a)(2)(i)(B).

CONCLUSION

On November 23, 2006, the main steam system valve lineup was performed and independently re-performed to ensure the main steam system was in a configuration for plant startup. On November 27, 2006, MS-100, 101, 102 and 103 were opened. As the plant was heating up, a steam leak was identified at a fitting near isolation valve MS-103. A clearance was established on November 28, 2006, which closed MS-100, 101, 102 and 103 to complete repairs. Approximately one day later, maintenance was completed and the clearance was removed. This clearance restoration opened MS-100, 101, 102 and 103.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Nuclear Station	05000285	2007	- 001 -	00	3 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Later on November 28, 2006, the steam generators were filled to between 85 – 89 percent narrow range as required per Operating Procedure (OP)-2A, attachment 1, "Plant Startup from Mode 4/5 to Mode 3." Later both steam generators were drained (level adjusted) to the specified range of 55 – 75 percent narrow range level. Procedure OI-FW-6, attachment 2, "Steam Generator Draining Bypassing the Blowdown Tank," was used to adjust the steam generator levels. This procedure closed valves MS-100, 101, 102 and 103, which is contrary to the startup lineup. Procedure OI-FW-6, Attachment 2 guidance was written assuming its performance prior to completing the main steam system valve lineup. This dependency was not understood by the plant staff.

On November 29, 2006, the plant heat up commenced and the station exited mode 4 (cold shutdown condition) at greater than 210 degrees F and was transitioning to mode 3 (hot shutdown condition). On December 3, 2006, a steam fitter mechanic completed the post maintenance test for the repair of the steam leak by documenting that there was no steam leaking from the fitting at MS-103. It appears that MS-102 and MS-103 were both closed at that time (in addition to MS-100 and 101), therefore no steam leak could be observed. On December 4, 2006, insulation was re-installed on the piping due to the successful post maintenance testing.

The root cause of this event was that procedure OI-FW-6, Attachment 2 guidance was written assuming its performance prior to completing the main steam system valve lineup. The procedural guidance in OI-FW-6 closed MS-100, 101, 102 and 103, which is contrary to the previously completed main steam system valve lineup. No other procedures were identified that had similar dependencies.

CORRECTIVE ACTIONS

As previously indicated the steam leak was repaired and valves MS-100, 101, 102 and 103 were opened.

Appropriate procedures will be revised to ensure that MS-100, MS-101, MS-102 and MS-103 remain open to so that RM-064 is not isolated during plant startup activities. The procedure revisions will be completed by May 31, 2007.

SAFETY SIGNIFICANCE

Main steam line radiation monitor, RM-064, when placed in service, is used for post-accident indication (steam generator tube rupture); it does not initiate any safety system actuations or any alert, warning, or high alarms. RM-064, when placed in service, assists in quantifying the effluent radioactivity release rate through HCV-1040 (atmospheric dump valve) main steam safety valves, and/or the turbine driven auxiliary feedwater pump.

A steam generator tube rupture can first be detected by steam generator blowdown sample radiation monitors (RM-054A and B), the condenser off gas monitor (RM-057), or chemistry samples. The radiation monitors initiate alarms in the control room and inform the operators of abnormal radioactivity levels and that corrective action is required.

Large leaks that exceed the capacity of one charging pump (greater than 40 gallons per minute) will be detected by lowering pressurizer level and/or low pressurizer pressure, which requires a reactor trip. The trip will initiate steam dump and bypass flow to the condenser. The steam generator tube leak can be detected by steam generator water level rising at a rate more rapidly than the intact steam generator. Also, increasing radioactivity as determined by radiation monitors or chemistry grab samples are indications that can be used to determine the affected steam generator, which will then be isolated. Once the steam generator is isolated, the plant would continue to be cooled down to shutdown cooling entry conditions. Therefore, this event had little impact on the health and safety of the public.

SAFETY SYSTEM FUNCTIONAL FAILURE

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

PREVIOUS SIMILAR EVENTS

There have not been any other instances of a similar nature resulting in the isolation of a radiation monitor at FCS in the last three years.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Nuclear Station	05000285	2007	- 00	- 00	4 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

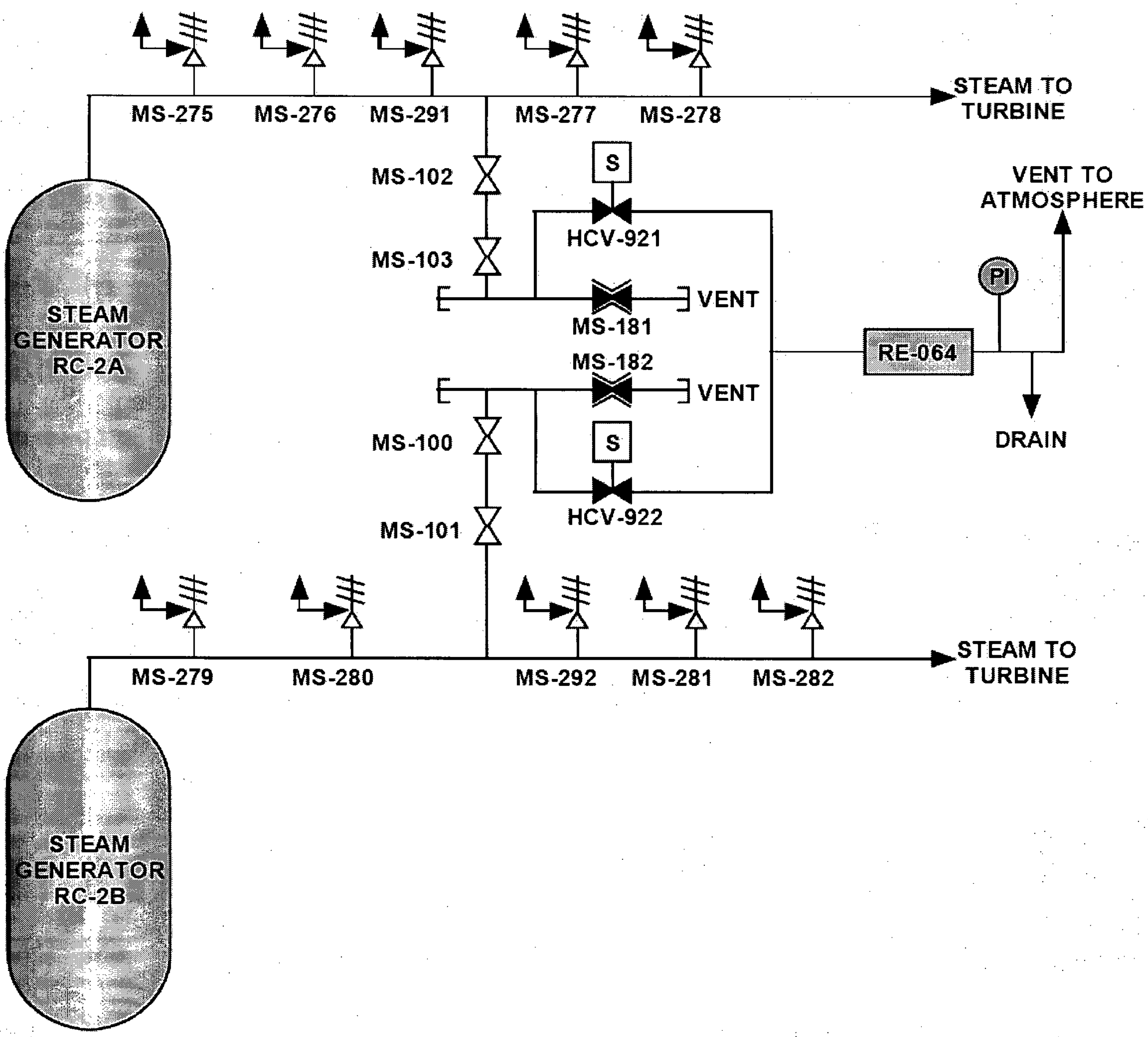

Figure 1 (Main Steam Line Radiation Monitor)