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

April 6, 2000
LIC-00-0029

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

References:
1. Docket No. 50-285
2. Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk), dated October 26, 1998 (LIC-98-0133)
3. NRC Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," dated June 3, 1999
4. Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk), dated August 2, 1999 (LIC-99-0068)
5. Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk), dated October 8, 1999 (LIC-99-0091)

Subject: Licensee Event Report 2000-001 Revision 0 for the Fort Calhoun Station

Please find attached Licensee Event Report 2000-001, Revision 0, dated April 6, 2000. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B). If you should have any questions, please contact me.

Sincerely,

S. K. Gambhir
Division Manager
Nuclear Operations

EPM/epm

Attachment

c: E. W. Merschoff, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
INPO Records Center
Winston and Strawn

IE22

NRC FORM 366 (6-1998) | U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Recalculation of Dose to the Control Room Operators Places the Plant Outside Design Basis

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	09	2000	2000	-- 001 --	00	04	06	2000	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Brad Serfas Nuclear Engineer	402-533-7216

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

The current design basis control room airborne dose calculations for accidents that result in radiological consequences to the Fort Calhoun Station (FCS) control room personnel assume an unfiltered control room in-leakage value of 0 standard cubic feet per minute (scfm). As a result of discussions with the NRC, tracer gas testing was performed to confirm this assumption. The testing determined that the unfiltered in-leakage rate is 8 scfm. On March 9, 2000, Design Engineering Nuclear (DEN) completed verification of the contractor analysis of the control room in-leakage test and the effect on control room dose. This evaluation determined that with the measured control room unfiltered in-leakage rate of 8 scfm, the design basis loss of coolant accident thyroid dose to the control room operators is about 32 rem. This is outside the current design basis of the plant of 30 rem thyroid dose.

DEN had previously prepared a Safety Analysis for Operability (SAO) 99-01 to address the basis for continued operability in the event that tracer gas testing resulted in unfiltered in-leakage greater than 0 scfm. With the measured unfiltered in-leakage being less than 100 scfm, the control room remains operable based upon the analysis and bases documented in the SAO.

Corrective actions to address this issue were included in the Station's response to Generic Letter 99-02 and LER 1998-013.

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

BACKGROUND

On June 3, 1999, Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," was issued. GL 99-02 states that testing nuclear-grade activated charcoal to standards other than American Society for Testing and Materials (ASTM) D3803-1989 does not provide assurance for complying with the current licensing basis as it relates to the dose limits of General Design Criterion (GDC) 19 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations and Subpart A of 10 CFR Part 100.

As requested by GL 99-02, Omaha Public Power District (OPPD) issued the initial response to GL 99-02. In this response, OPPD declared that, in accordance with requested action 5, it was pursuing an alternate course of action at Fort Calhoun Station (FCS) which differs from requested actions 1-4. The alternate course of action that was taken is to: (1) remove credit for the majority of the activated charcoal filters from the FCS design basis accident radiological consequences analysis, and (2) deviate from the schedule requirements of actions 1-4. Activated charcoal filters which were credited in the radiological consequences analysis would be tested to the ASTM D3803 -1989 protocol. OPPD committed to revise the radiological consequences analysis to not credit the Containment Air Cooling and Filtering Charcoal Filters (VA-6A/B). To offset the loss of containment atmosphere iodine removal provided by VA-6A/B, the revised Loss Of Coolant Accident (LOCA) radiological consequences analysis would credit the safety-grade Containment Spray System (1 pump/1 header). Based upon the results of the revised radiological consequences analysis, OPPD would then develop a detailed GL 99-02 project plan to complete required actions and address GL issues, including revision of the applicable Technical Specifications to specify the test efficiency and use of the ASTM D3803 -1989 protocol for any credited charcoal filters. OPPD committed to submit the project plan by November 29, 1999. OPPD stated that the bases for continued operability of the FCS charcoal filters include continuing compliance with current applicable FCS Technical Specifications regarding charcoal filter testing and the preliminary results of the revised radiological consequences analysis.

USAR Section 14.15.8.2 documents the control room radiological consequences following a design basis LOCA. One of the primary design parameters that determine thyroid dose is the amount of unfiltered in-leakage into the control room. Currently, no unfiltered in-leakage is assumed since the control room is maintained at a positive pressure with respect to adjacent areas, and air lock type vestibules are installed for each entrance to the control room. This position is documented in USAR section 14.15.8.2 and in the associated radiological consequences calculations.

The current FCS control room emergency ventilation system was installed under modification MR-FC-87-20. This modification was designed to maintain positive pressure in the control room relative to all adjacent spaces and to prevent in-leakage into ducting located outside the control room. Based upon these design criteria, the assumption of 0 scfm unfiltered in-leakage was subsequently incorporated into the FCS radiological consequences analysis as a design basis assumption. The design basis control room habitability analysis of record for FCS assume 0 scfm unfiltered in-leakage for the 30 days following a design basis LOCA.

Tracer gas tests conducted at several nuclear power facilities have shown that control room envelope integrity to be inconsistent with the licensing and design bases. The concern is that the actual control room boundary may not be as leak-tight as assumed in design calculations, thereby allowing a larger radiological dose to the operations staff than previously analyzed. The NRC has established a position of requiring periodic verification of control room envelope integrity in recent correspondence with the Nuclear Energy Institute (NEI). More specifically, the NRC challenged OPPD during an August 17, 1999, meeting on the basis for assuming control room unfiltered in-leakage of 0 scfm in the radiological consequences analysis.

EVENT DESCRIPTION

In order to establish a documented empirical value for unfiltered control room in-leakage, OPPD decided to conduct tracer gas leakage testing for the FCS control room.

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

Design Engineering Nuclear (DEN) prepared a contingency Safety Analysis for Operability (SAO) 99-01 prior to tracer gas testing to address the basis for continued operability in the event that tracer gas testing resulted in unfiltered in-leakage greater than 0 scfm. SAO 99-01 was approved on November 9, 1999.

Tracer gas testing was conducted on the FCS control room envelope in December of 1999. Preliminary measured in-leakage results were between 30 and 40 scfm. Measured in-leakage is defined as the total of the filtered and unfiltered in-leakage contributions. The final report documented a maximum unfiltered in-leakage rate of 8 scfm. Based upon the measured unfiltered in-leakage being less than 100 scfm, the control room remains operable as described in the analysis and bases documented in SAO 99-01. SAO 99-01 compensatory actions are described in the SAFETY SIGNIFICANCE section of this LER.

On March 9, 2000, DEN completed verification of the contractor analysis of the control room in-leakage test and the effect on control room dose. The evaluation determined that with the measured control room unfiltered in-leakage rate of 8 scfm, the post-LOCA thyroid dose to the control room operators would be about 32 rem, and that the condition was reportable on March 9, 2000, at 1054 Central Standard Time (CST). A one (1) hour non-emergency report was made to the NRC Operations Center at 1142 CST on March 9, 2000, pursuant to 10 CFR 50.72(b)(1)(ii)(B). This report is being made pursuant to 10 CFR 50.73(a)(2)(ii)(B).

SAFETY SIGNIFICANCE

The design basis control room habitability analysis of record for FCS assumes 0 scfm unfiltered in-leakage for the 30 days following a design basis LOCA. FCS was uncertain as to whether the control room tracer gas test scheduled for December 1999 would validate unfiltered in-leakage of 0 scfm. To address the possibility of unfiltered in-leakage to the control room, the existing LOCA radiological analysis was re-run with 100 scfm unfiltered in-leakage.

Calculated whole body, thyroid, and skin doses are shown below. These doses represent the following post-LOCA release pathways: (1) Containment leakage, (2) safety injection and refueling water tank leakage, and (3) Containment Pressure Relief Line releases. These cases have been run with no credit (0 percent efficiency) for VA-6A/B and full credit for containment spray iodine removal in accordance with Standard Review Plan (SRP) 6.5.2, "Containment Spray as a Fission Product Cleanup System. "

Control Room Unfiltered In-Leakage Assumed for 30 Days (LOCA)

	Control Room Dose (rem)		
	Whole Body	Thyroid	Skin
0 scfm	1.4	24	26
100 scfm	1.5	114	26

Note: Whole Body dose does not include external shine contributions of 1.4 rem.

SRP 6.4 dose limits are 5 rem whole body, 30 rem thyroid, and 30 rem skin. The post-LOCA control room dose for the 100 scfm unfiltered in-leakage case exceeds the 30 rem thyroid dose limit.

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The compensatory measure to reduce the calculated control room thyroid dose below the 30 rem limit is the administration of Potassium Iodide (KI). Potassium Iodide will be issued in accordance with the procedural requirements contained in Emergency Plan Implementing Procedure (EPIP) Emergency Operating Facility (EOF) procedure 21 (EPIP-EOF-21), "Potassium Iodide Issuance." Administration of KI within 1 hour of the initial exposure provides a thyroid dose reduction factor of 10. Therefore, the revised control room doses with the compensatory measure (KI administration) are presented below.

Control Room Unfiltered In-Leakage Assumed for 30 Days (LOCA) with KI Administration

	Control Room Dose (rem)		
	Whole Body	Thyroid	Skin
100 scfm	1.5	11.4	26

Note: Whole body dose does not include external shine contributions of 1.4 rem.

The post-LOCA control room doses are within the dose limits specified in SRP 6.4 (5 rem whole body, 30 rem thyroid, and 30 rem skin), as shown above for the case that analyzes 100 scfm of unfiltered in-leakage and credits the administration of KI. The measured unfiltered in-leakage value of 8 scfm would produce considerably less dose. Therefore, this condition would have no impact on the public and no impact on plant personnel.

CONCLUSION

The cause of this condition is the incorporation of the assumption that if the control room is maintained at positive pressure then there is no unfiltered in-leakage. The FCS control room emergency ventilation system was installed under modification MR-FC-87-20. This modification was designed to maintain positive pressure in the control room relative to all adjacent spaces and to prevent in-leakage into ducting located outside the control room. Based upon these design criteria, the assumption of 0 scfm unfiltered in-leakage was subsequently incorporated into the FCS radiological consequences analysis as a design basis assumption without basis.

CORRECTIVE ACTIONS

The completed corrective action for this event included processing the SAO previously discussed. In addition, completed corrective actions have been previously discussed in LER 1998-013, revision 0 and OPPD responses to GL 99-02. Any additional corrective actions are covered by the FCS's corrective action program.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event did not result in a safety system functional failure in accordance with NEI 99-02, revision 0 .

PREVIOUS SIMILAR EVENTS

LER 1998-012 and LER 1998-013 reported similar issues where revisions to the Station's radiological consequences analysis placed the plant in a condition outside of the plant's design basis.