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

October 26, 1998
LIC-98-0133

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 (W. G. Gates) to NRC (Document Control Desk), dated July 20, 1990 (LIC-90-0449)

Subject: Licensee Event Report 1998-013 Revision 0 for the Fort Calhoun Station

Please find attached Licensee Event Report 1998-013 Revision 0 dated October 26, 1998. This report is being submitted pursuant to 10CFR50.73(a)(2)(ii)(B). If you should have any questions, please contact me.

Sincerely,

S. R. Gambhir
Division Manager
Nuclear Operations Division

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 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 4/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO THE INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Station Unit No. 1	05000285	1 OF 5

TITLE (4)
Control Room Radiological Dose Consequences Outside of 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
09	24	1998	1998	-- 013 --	00	10	26	1998	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)
ick P. Matzke, Station Licensing Engineer	402-533-6855

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)

Fort Calhoun Station is currently conducting a comprehensive review of the Radiological Consequences Program. During that review, it was discovered that the Updated Safety Analysis Report (USAR) documents a condition that appears to be outside the design basis of the plant. The design basis for the control room post Loss of Coolant Accident (LOCA) whole body dose is General Design Criteria (GDC) 19, which imposes a limit of 5 rem whole body exposure to the control room operators over a 30 day period. The USAR documents whole body doses inside a small noncritical portion of the control room envelope (within 8 feet of the east doors to the control room) which is calculated to exceed the GDC 19 limit.

The affected area of the control room does not contain any control panels or other equipment needed for safe shutdown of the plant. The area is only occupied as needed to transit to and from the control room. The control room is able to meet its safety functions without additional restriction. This issue has been documented in correspondence to the NRC on July 20, 1990.

Action is being taken to upgrade the Radiological Consequences Program and update the calculations. Appropriate actions will be pursued upon completion of the revised calculations.

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

BACKGROUND

The Radiological Consequences Program develops and maintains evaluations that ensure plant initial design, modifications, events, and postulated accidents do not result in Fort Calhoun Station (FCS) personnel or the general public receiving radiation dose in excess of established regulations.

The Radiological Consequences Program has direct applicability to Section 11.2, "Radiation Protection and Monitoring," and several parts of Section 14, "Safety Analysis," of the Updated Safety Analysis Report (USAR).

Specific examples of radiological consequence applications are:

- ensuring that certain vital areas of the station (i.e., control room, technical support center) remain habitable during postulated plant accidents,
- evaluation of new reactor core assembly enrichments to ensure that design and licensing requirements are still met (i.e., projected doses for accident scenarios),
- review/evaluation of plant modifications/design changes having the potential to affect doses to station personnel/equipment and the general public during postulated accidents (i.e., control room ventilation modifications).

EVENT DESCRIPTION

FCS is currently conducting a comprehensive review of the Station Radiological Consequences Program. During that review, it was discovered that the USAR documents a condition that appears to be outside the design basis of the plant. The design basis for the control room post Loss of Coolant Accident (LOCA) whole body dose is General Design Criteria (GDC) 19, which imposes a limit of 5 rem whole body exposure to the control room operators over a 30 day period. USAR Section 14.15.8.2, "Overhead Cloud Shine," states:

> "The radiation shine from the ventilation duct penetrations in the south wall has not been quantified since this impacts the mechanical equipment room, which is not habitable for continuous occupancy as discussed in Reference 14.15-29. Similarly, radiation shine through the elevator machine exhaust penetration has not been quantified since this area does not require post-accident occupancy.
>
> Radiation shine from the control room doors was calculated as a function of distance into the control room (Ref. 14.15-31). Based on these results, at a distance of 8 feet a 3 rem integrated dose is calculated which when added to other doses is less than the 5 rem GDC 19 limit.
>
> Locations within 8 feet of the doors will not be occupied to any significant extent, and, therefore, will not present a radiation dose concern."

And later in Section 14.15.8.2, "Summary of LOCA Doses:"

> "The 5 rem gamma dose limit is met throughout the control room envelope except for the mechanical equipment room, which is not continuously occupied at any time. The area directly inside the two control room doors is also calculated to exceed 5 rem; however, these areas will not be occupied to a large extent during post-LOCA conditions."

These USAR sections document whole body doses inside a small noncritical portion of the control room envelope (within 8 feet of the east doors to the control room) which exceed the GDC 19 limit. The change to the USAR that documented this information was made in the 1990/1991 time frame.

On September 24, 1998, at 1943 CDT, it was conservatively determined that the plant was outside its design basis. A one (1) hour non-emergency report was made to the NRC Operations Center at 2009 CDT pursuant to 10CFR50.72(b)(1)(ii)(B). This report is being made pursuant to 10CFR50.73(a)(2)(ii)(B).

SAFETY SIGNIFICANCE

As previously noted, the USAR states, "Radiation shine from the control room doors was calculated as a function of distance into the control room. Based on these results, at a distance of 8 feet a 3 rem integrated dose is calculated which when added to other doses is less than the 5 rem GDC 19 limit. Locations within 8 feet of the doors will not be occupied to any significant extent, and therefore, will not present a radiation dose concern." The affected area of the control room does not contain any control panels or other equipment needed for safe shutdown of the plant. The area is only occupied as needed to transit to and from the control room. The control room is still able to meet its safety functions without additional restriction. Therefore, this event has minimal impact on the health and safety of the public.

CAUSAL ANALYSIS

On July 20, 1990, in correspondence with the NRC (Reference 2), the Omaha Public Power District (OPPD) documented this issue. At that time OPPD did not ask for a deviation from GDC 19 or Standard Review Plan (SRP) 6.4.

The Nuclear Safety Review Group (NSRG) performed a review of the Radiological Consequences Program in December 1997. The primary emphasis of the review was to determine the adequacy of providing management direction and oversight of the program. The intent of the review was not to conduct a detailed review of the technical aspects of the program. Technical aspects of the program were evaluated, as necessary, to support the review objectives.

The specific objectives of the review were:

- verify that program owners display a high level of awareness regarding responsibility and accountability for the program for which they are responsible,

- determine if program procedures and basis documents are written in a user-friendly format, are technically accurate, and support program implementation,

- verify program owners routinely monitor, observe, and assess activities and plant conditions affecting their program to maintain and reinforce high standards of performance,

- determine if program personnel possess the knowledge and skills necessary to effectively implement program requirements.

The review concluded that the Radiological Consequences Program has not been effectively maintained. Prior to March 1997, ownership of the Radiological Consequences Program was not clearly established. No individual or group, until March 1997, had been established as having responsibility for implementation of the program. Rather, a number of groups/departments performed specific tasks related to the program. However, no one organization consistently provided oversight of program implementation.

Currently, communications between management and personnel responsible for implementation of the Radiological Consequences Program were evidenced to be good. Program owners and management personnel indicated an awareness of program weaknesses and areas for potential improvement.

CORRECTIVE ACTIONS

In March 1997, management clearly established program ownership in the Nuclear Engineering group of Design Engineering Nuclear (DEN).

An awareness to more formally define the program requirements and address other potential program weaknesses was recognized by the recently assigned program owner. A Senior Nuclear Design Engineer with experience in health physics was added to the staff of DEN. This individual was assigned as the Program Owner of the Radiological Consequences Program. This individual possesses a Nuclear Engineering degree and is a Certified Health Physicist by the American Board of Health Physics. Thus, the individual possesses a solid foundation of knowledge. A project plan has been developed to upgrade the Radiological Consequences Program.

The scope of the plan has five distinct phases:

- assessment of the current radiological consequences analyses
- assessment and collection of technical resources to perform radiological consequences analyses (personnel, computer software/codes, technical data)
- evaluation and documentation of assumptions/methods/input data selection
- development of revised radiological consequences analyses when necessary
- implementation of upgraded radiological consequences analyses when needed

The operators have been made aware of the issue until final resolution can be completed as described below.

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

The following additional corrective actions will be completed:

1. A self assessment of the program is currently being undertaken. This LER is the result of this assessment. The assessment will be completed by November 30, 1998.

2. The radiological consequences calculations will be recalculated and reverified to current industry standards. This effort will be completed by October 30, 1999. Needed corrections to plant design information or modifications to the plant will be evaluated as they are identified and appropriately scheduled.

PREVIOUS SIMILAR EVENTS

LER 1998-012 reported a similar incident where program ownership issues placed the plant in a condition outside of the plant's design basis.