NRC FORM 366 (4-95) 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)
Fort Calhoun Station Unit No. 1	05000285	1 OF 7

TITLE (4)
Fuel Movement without Control Room Filtration in Operation

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
04	25	96	96	-- 003 --	00	11	07	96	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)	X	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)
Erick 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 NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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)

In April 1996, a procedure change and associated evaluation was being reviewed by an engineer who was researching a potential facility license change related to Control Room Heating Ventilating and Air Conditioning. During the review, the engineer noted an inconsistency in the safety evaluation's discussion of control room habitability requirements for fuel handling and 10 CFR 100.11 requirements. The referenced documents did not provide an adequate basis for the change. The change allowed the control room filtration system to be in an "operable" status instead of an "operating" status as assumed by the analysis. Although the plant was not handling fuel at the time of the review, the plant had conducted fuel movements in 1995 with the control room filtration system in an "operable" status, but not an "operating" status.

The root causes of this event were determined to be inadequate implementation of a change to the station design basis and inadequate preparation and review of the attendant procedure change.

Corrective actions include revising appropriate station operating procedures to correctly reflect the current station design basis, and strengthening the station procedures for implementing engineering analyses and performing 10 CFR 50.59 safety evaluations.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Station Unit No. 1	05000285	96	003	00	2 OF 7

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

BACKGROUND

Control room dose limits are addressed in Title 10 Code of Federal Regulations, Section 50 (10 CFR 50), Appendix A, General Design Criterion (GDC) 19, and in the Standard Review Plan (SRP) Section 6.4. GDC 19 and SRP 6.4 specify that the limit for thyroid and beta skin dose in the control room following an accident as 30 rem. The Fort Calhoun Station (FCS) Updated Safety Analysis Report (USAR) Section 14 describes the plant Design Basis Accidents (DBAs). USAR Section 14.23 addresses "Control Room Habitability during Toxic Chemical Release Accidents," but this section does not address radioactive releases. USAR Section 14.18, "Fuel Handling Accident (In Spent Fuel Pool and Containment)," addresses offsite dose limits, but does not address control room dose limits. Several other USAR Chapter 14 analyses address offsite radiological consequences but not control room doses. The only USAR analysis that addresses both offsite and control room dose calculations is the Loss of Coolant Accident (LOCA) analysis (USAR Section 14.15). However, the assumptions for the LOCA calculations are not applicable to a Fuel Handling Accident. USAR Section 9.5.1.5 discusses protection of control room operators from radioactivity during fuel movement, but does not discuss the dose calculations.

EVENT

Control room dose calculations were issued under Engineering Analysis EA-FC-90-037, "Control Room Habitability Evaluation," in May of 1990. This analysis, in part, calculated control room doses resulting from Fuel Handling Accidents in either the Containment or the Spent Fuel Pool area. The analysis assumed that the control room emergency filtration system would be placed into operation before any fuel movement in the containment or before any irradiated fuel movement or new fuel movement over irradiated fuel in the spent fuel pool area. When EA-FC-90-037 was issued, work was still in progress on a separate Engineering Analysis EA-FC-90-094 "Fuel Handling Accident & Bounding Source Term" that would update the offsite (but not the control room) dose calculations for Fuel Handling Accidents.

While the control room dose calculation (i.e., EA-FC-90-037) was being completed, Procedure Change (PC) 32916 was initiated to add a requirement to Operating Instruction (OI) OI-FH-1 "Fuel Handling Equipment Operation" for a control room filtration unit (VA-64A or B) to be operating in the Filtered Makeup Mode (FMM) during movement of specified loads over irradiated fuel in the core or the Spent Fuel Pool (SFP). During a July 19, 1990 Plant Review Committee (PRC) meeting, PC 32916 was approved for OI-FH-1, however, it was noted that "Engineering is going to do an analysis to see if this requirement can be removed." A PRC action item (CID 900627) was assigned to track the additional evaluation that would be required. Related

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

procedure changes were approved for OI-HE-1 "Polar Crane Normal Operation" and Operating Procedure (OP) OP-11 "Reactor Core Refueling." In addition, USAR Section 9.5.1.5 was subsequently revised to identify situations in which the control room ventilation system would be placed in the FMM with one unit in operation.

In response to the previously mentioned PRC action item, a follow-up presentation was made to the PRC by Design Engineering on February 13, 1991. The presentation said that the existing requirements of OI-FH-1 could not be relaxed unless an additional analysis was authorized. This presentation addressed the Reactor Vessel Head (RVH) drop issue only; there was no discussion of the Fuel Handling Accident analysis. The PRC action item was closed after this presentation.

In October 1993, the Reactor Engineer contacted the Supervisor - System Engineering with a request for investigation of the need for prerequisite 4.7 in OI-FH-1. (Prerequisite 4.7 was the requirement added via PC 32916). The Reactor Engineer referred to memo PED-FC-90-1165 (dated June 21, 1990), which had suggested that an analysis was in progress that would provide site boundary and control room doses, ". . . hopefully without reliance upon control room charcoal filters for accident mitigation." The Reactor Engineer's request was forwarded to Design Engineering for action. In response to this request, a memorandum (PED-FC-93-2932) from Design Engineering was generated which stated that the required analysis had been completed. This memo concluded that ". . .the control room ventilation is no longer required to be in the filtered makeup mode before fuel handling or operations in the spent fuel pool area." This memo was presented to the PRC on May 27, 1994. Following this presentation, PC 43071 was initiated to revise OI-FH-1 to "Eliminate placing CR ventilation in filtered." PC 43071 was approved on June 30, 1994. A revision to USAR Section 9.5.1.5 was processed as a result of PC 43071. Two changes to OP-11 to eliminate placing control room ventilation in FMM during fuel handling were approved on January 21, 1995 and February 7, 1995.

In April 1996, the Nuclear Safety Evaluation (FC-154) for PC 43071 was reviewed by a Licensing Engineer who was researching a potential Facility License Change related to control room Heating Ventilating and Air Conditioning (HVAC). During the review, the engineer noted an apparent inconsistency in the safety evaluation's discussion of control room habitability requirements and 10 CFR 100.11 requirements. Review of reference documents suggested that the Engineering Analysis referenced in PC 43071 (i.e., EA-FC-90-094) did not provide an adequate basis for the change. Condition Report (CR) 199600466 was written to identify the apparent discrepancy between USAR Section 9.5.1.5, OI-FH-1 and EA-FC-90-094. OI-FH-1 was found to require that the control room ventilation system FMM need only be operable before moving irradiated fuel. Both the USAR and OI-FH-1 had previously required that the control room ventilation system FMM actually be operating before moving irradiated fuel.

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

CR 199600466 also questioned the justification for changing OI-FH-1. One of the corrective actions identified by CR 199600466 was to revise the appropriate section of the USAR to reflect the control room dose information from the appropriate analyses.

On April 25, 1996, at 1606 Central Daylight Time (CDT), the PRC determined that this change was not adequately supported by existing analyses (EA-FC-90-037 and EA-FC-90-094) and that fuel had been moved during the 1995 Refueling Outage without the control room ventilation system FMM being in operation as required by analysis EA-FC-90-037. A four-hour non-emergency report was made to the NRC Operations Center pursuant to 10 CFR 50.72(b)(2)(iii)(D) on April 25, 1996 at 1744 Eastern Daylight Time (EDT). Subsequent investigation also revealed that fuel movement had occurred in May 1995 for fuel sipping.

Design Engineering then initiated analysis (EA-FC-96-030) to reassess the control room doses following a fuel handling accident in the SFP area of the Auxiliary Building or in the Containment. The analysis, EA-FC-96-030, was completed on May 20, 1996. EA-FC-96-030 demonstrated that the dose to the control room operators due to either of the postulated accidents would be below the maximum acceptable control room dose assuming automatic initiation of control room ventilation FMM following the accident.

On May 28, 1996, Design Engineering briefed the PRC on the results of EA-FC-96-030. Design Engineering stated that the results of this analysis showed that the procedure change for OI-FH-1 that allowed automatic initiation of the control room ventilation in the FMM was supported. Using the information presented the PRC determined that the event reported on April 25, 1996, was no longer reportable. At 1631 EDT on May 28, 1996, the NRC Operations Center was informed that the event, previously reported on April 25, 1996, was being withdrawn.

In October 1996, during the process of revising the section of the Technical Specifications dealing with control room ventilation, one of the reviewers noted that analysis EA-FC-96-030 credited the stack radiation monitors (RM-052 and RM-062) as redundant. OI-FH-1 and the other related design basis information (USAR) did not appear to specifically identify the need for the correct power supply alignment for the radiation monitors which would be needed to meet the FCS single failure criteria. CR 199601167 was generated to investigate this apparent discrepancy.

The investigation into this issue revealed that although analysis EA-FC-96-030 did use the FCS single failure criteria, the assumptions in the analysis had not been verified as properly implemented in the appropriate plant operating instructions. This included the operating instructions that had been believed to implement the analysis and allowed the subsequent withdrawal of the original notification on May 28, 1996. The single failure assumption used in the analysis was not listed in the assumptions

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Station Unit No. 1	05000285	96	- 003 -	00	5 OF 7

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

section of analysis EA-FC-96-030, but was implied in the data section of the analysis under the title "Stack Monitor Smoothing Algorithm."

On October 8, 1996, this additional information on the failure to properly implement the single failure criteria in OI-FH-1 was presented to the PRC by Design Engineering. At 1021 CDT, the PRC determined that PC 43071 was not supported by analysis EA-FC-96-030. A four-hour non-emergency report was made to the NRC Operations Center pursuant to 10 CFR 50.72(b)(2)(iii)(D) on October 8, 1996, at 1211 EDT. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(v) and 10 CFR 50.73(a)(2)(ii).

SAFETY SIGNIFICANCE

The implementation of PC 43071 without appropriate supporting analysis led to a condition where the design basis of the control room ventilation system was not met during the 1995 refueling outage or fuel sipping in May 1995. An evaluation of the potential effects of the errors reported in this LER indicate that for the design basis fuel handling accident, using the NRC required 176 fuel pins failing on the impact of the fuel bundle, without the control room ventilation in the FMM (since a single failure could have prevented automatic operation of this system) and without operator action, calculated doses to the control room operators could have been as high as 0.53 rem whole body, 170 rem thyroid, and 51 rem beta skin. Note that whole body gamma dose remains well within the design limit of 5 rem. Using the same conditions, except with a best estimate for failure of fuel pins during the accident of 28 pins as discussed in USAR section 14.8.2, the thyroid dose would be about 27 rem and the beta skin dose 8 rem. A best estimate analysis indicates that control room habitability would be met during an accident.

Operator action during a fuel handling accident would further mitigate the consequences of an accident. Rapid operator response to a fuel handling accident is expected for the following reasons:

1) The operators have a heightened awareness to problems associated with fuel handling due to operator training and the required pre-job briefing for fuel handling.

2) The requirement for continuous communication by the operators handling fuel and a dedicated operator in the control room.

3) Several radiation monitors, area monitors as well as stack monitors, are in service to provide control room annunciation of high radiation levels.

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

The events noted in this LER do not affect the analysis of releases to the general public due to a fuel handling accident. There is no increase in dose at the Exclusion Area Boundary (EAB) or the Low Population Zone (LPZ) as a result of the events noted.

CONCLUSION

Investigations were conducted to determine the cause of the problems noted in this report. The following conclusions were drawn from these investigations:

1) The memorandum (PED-FC-93-2932) that was used as technical support for PC 43071 to OI-FH-1 (June 1994) was prepared without adequate evaluation or review to ensure its validity.

2) The safety evaluation for PC 43071 to OI-FH-1 (June 1994) was prepared without adequate evaluation and approved without adequate review to ensure its validity.

3) EA-FC-96-030 was used to accept a revision to an existing procedure that had not received a valid safety evaluation. This resulted in inappropriately withdrawing a 10 CFR 50.72 notification.

The decision to initiate and eventually approve the 1994 PC was based on the premise that appropriate evaluations had been completed to justify the change. This premise could have been assessed and/or questioned more easily if the USAR Fuel Handling Accident analysis had addressed control room dose requirements. This lack of control room dose calculation information in the USAR is considered a contributing cause to these events.

CORRECTIVE ACTIONS

An Operations Memorandum was issued to require the appropriate configuration of system equipment to ensure that the assumptions of the current analyses were implemented in a conservative fashion for plant operations prior to fuel movements during the current refueling outage. Plant procedures have been revised to comply with the operations memorandum.

Procedure NOD-QP-3, "10 CFR 50.59 Safety Evaluations" had been previously revised to provide additional guidance on the necessary level of expertise and the use of informal documentation in the preparation of these safety reviews. In addition, the importance of the 10 CFR 50.59 program has been emphasized, in the last two years, to all personnel involved in the program.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort Calhoun Station Unit No. 1	05000285	96	- 003 -	00	7 OF 7

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

In addition to these completed corrective actions the following additional actions will be taken to minimize the possibility of an event of this nature occurring in the future:

1. EA-FC-90-37, EA-FC-90-94 and EA-FC-96-030 will be reviewed to:

   a) Identify and verify both inputs and assumptions within the engineering analyses, including those pertaining to operating procedures and system configurations;

   b) Verify that the engineering analyses results and affected operating configurations, if any, were appropriately incorporated into the USAR, operating instructions using the 50.59 process, and Design Basis Documents.

   This review will be completed by January 15, 1997. A plan to implement any necessary corrective actions will be developed by January 31, 1997.

2. A review of OPPD and industry practices for administrative control of assumptions used in EAs will be conducted to ensure that complete, accurate and timely information is sent to the PRC. The conclusions of this review will be implemented through revisions of appropriate Design Engineering procedures by April 15, 1997.

PREVIOUS EVENTS

LER 95-002 reported a previous incident where an assumption in an engineering analysis was not adequately implemented in plant procedures.