NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Station, Unit No. 1	0 5 0 0 0 2 8 5	1 OF 0 7

TITLE (4): Potential Failure to Maintain Containment Integrity When Required

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 NAMES	DOCKET NUMBER(S)
0 3	2 7	8 7	8 8	0 1 1	0 2	0 6	0 2	8 8	N	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	XX 50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
WILLIAM C. PHILLIPS, Shift Technical Advisor	4 0 2	4 2 6 - 4 0 1 1

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)

YES (If yes, complete EXPECTED SUBMISSION DATE)	XX NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

A notification to the NRC was made at 1133 (CDT), April 19, 1988, per 10 CFR 50.72(b)(1)(ii)(A), concerning the loss of containment integrity, due to the discovery of a non-installed Swagelok cap on containment penetration M-38. Following the discovery, the cap was installed and a thorough inspection of non-wetted small diameter piping was conducted to verify containment integrity. Fort Calhoun Station, Unit No. 1 continued to operate at 100 percent power throughout this timeframe.

While performing a verification of the containment penetration M-38 piping configuration, to be used in the containment isolation valve leakage surveillance test, ST-CONT-3, an Engineer discovered a Swagelok cap missing from a test tee. The test tee was located upstream of PC-743, containment general area high pressure alarm transmitter. Following this discovery the cap was reinstalled and tested for leakage.

To prevent recurrence: 1) ST-CONT-3 will be upgraded to include: Detailed drawings including test tees and procedural signoffs for the removal and installation of the designated caps. 2) A separate documented and double verified checklist of Swagelok caps will be written, and this checklist will be performed prior to power operation following a refueling outage to ensure containment integrity. These actions will be completed prior to the end of the 1988 refueling outage.

On April 18, 1988, the Omaha Public Power District was in the process of upgrading the containment isolation valve leakage surveillance test, ST-CONT-3. During a verification of the containment penetration M-38 piping configuration, used in the containment isolation valve leakage surveillance test, an Engineer discovered a Swagelok cap missing from a test tee, and lying on top of pressure transmitter PC-743. The test tee is located upstream of PC-743, containment general area high pressure alarm transmitter. Upon this discovery, the Instrument and Control (I&C) department was notified to investigate the problem. The cap was reinstalled and tested for leakage to ensure containment integrity could be satisfied.

The incident was reported to an alternate to the Plant Manager. The Alternate called a Plant Review Committee meeting to discuss the incident. The PRC concluded that the event constituted a one hour notification, and notification to the NRC was made at 1133 (CDT), April 19, 1988, per 10 CFR 50.72(b)(1)(ii)(A). The notification was based on the potential loss of containment integrity resulting from the uninstalled Swagelok cap. Following the discovery, a thorough containment pipe penetration inspection was conducted to verify containment integrity. Fort Calhoun Station, Unit No. 1 continued to operate at 100 percent power throughout this timeframe.

Division Management activated a Management Investigative Safety Team (MIST) to provide additional support in the review and analysis of the incident. The MIST recommended accelerating the development of the upgrade to containment isolation valve leak surveillance test, ST-CONT-3, to verify alignment and layout of all penetrations. Additionally, special procedure SP-CONT-3 was written and performed to complete penetration drawings and to document double verification of containment integrity. SP-CONT-3 verified that all penetrations were intact and containment integrity was met. Additionally, an Engineering Evaluation and Assistance Report (EEAR) will be initiated to evaluate the desirability of adding isolation valves up stream of the test tees.

A review of past records indicated that the Swagelok cap located on the PC-743 test tee was removed on March 27, 1987, for leak rate testing. The test conducted was portion 1 of ST-CONT-3, which tests the leakage through the instrument isolation valves. In preparation to conduct the test a valve lineup was performed (see attached FIGURE 1). On the following day, portion 2 of ST-CONT-3 was performed, which tests the leakage through the header containment isolation valve. In preparation to conduct the test a valve line up was performed (see attached FIGURE 2). It appears that the I&C personnel reinstalled only three of the four test tee caps, inadvertently missing the PC-743 test tee cap, located in a different area from the other three.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (8) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort Calhoun Station, Unit No. 1	0 5 0 0 0 2 8 5	8 8	— 0 1 1 —	0 2	0 3 OF 0 7

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

Following the incident a Human Performance Evaluation System (HPES) root cause analysis was performed. The root cause was an inadequate procedure. Contributing factors include human error, and the location of the valves relative to each other. The test tees, required to be open for the tests, are located in two different areas, (areas 2 and 3 of attached FIGURE 3). The test procedure requires a system tag out. Containment penetration M-38 was tagged out to perform portion 2 of ST-CONT-3. The procedure further instructs personnel to remove the test tee caps to provide vent paths for instrument protection. To accomplish this valves VA-227, VA-260, and VA-275 (see attached FIGURE 2) are opened. The surveillance test was conducted and the pressure was returned to normal. Vent paths were capped, (test tees 1, 2, and 3 in Area 2 of attached FIGURE 3). It is assumed that test tee cap 4 (located in Area 3 of attached FIGURE 3) was left off. The tags were cleared and the valves realigned, completing the test of M-38. The surveillance test did not have a checklist with individual signoffs, nor did any of the work focus in the area of the PC-743 test tee.

On May 12, 1987, two maintenance orders, MO 872525 and MO 872526, were performed verifying that all Swagelok caps were in place or removed, as required to ensure containment integrity prior to startup as required by Technical Specifications. Two unofficial and documented checklists were used. During the verification, the M-38 penetration was checked to ensure the penetration sleeve had an installed cap and the inside containment cap removed to ensure actual containment pressure was being indicated. Neither list included a signoff for the PC-743 test tee cap.

The test tee cap being inadvertently omitted resulted in degradation of the high containment pressure general area alarm. The alarm may not have actuated on increasing pressure as designed. In addition, the function of A/PC-765, which provides input for the Reactor Protective System (RPS) channel High Containment Pressure trip, was potentially degraded. If it failed to function as designed, the Containment High Pressure RPS trip logic would be in a 2 out of 3 configuration. The RPS trip logic is normally 2 out of 4 on all trip functions. Additionally, A/PC-744-1 and A/PC-742-1, "A" channel Containment Pressure High Signal (CPHS) to the Prime and Backup trip relays to the Engineered Safeguard Features (ESF) system was potentially degraded. This would place the CPHS trip logic in a 2 out of 3 configuration. The CPHS trip logic is normally 2 out of 4.

The incident resulted in several potential Technical Specification violations. Technical Specification 1.3 states that the RPS trip settings and the permissible bypasses for the instrument channels shall be within the Limiting Safety System Settings as specified in Table 1.1. The omission of the test tee cap resulted in a potential degradation of the setpoints as defined in Technical Specification 1.3(7) and Table 1.1(7), for channel "A" RPS, which is designed to trip the reactor on containment high pressure. The setting as listed in Table 1.1 is less than or equal to 5 psig. Technical Specification

2.6 states that containment integrity shall not be violated unless the reactor is in the cold shutdown condition. Except for testing one Control Element Assembly Mechanism (CEDM) at a time, positive reactivity changes shall not be made by Control Element Assembly (CEA) motion unless containment integrity is intact. Because of the resultant incident the provisions of Technical Specification 2.6 listed above were compromised. Technical Specification 2.14 requires that the ESF system initiation instrumentation setting limits shall be as stated in Table 2.1. The omission of the test tee cap resulted in a violation of Technical Specification 2.14(1) and Table 2.1(1), for the prime and backup relays for the Containment High Pressure Signal (CPHS). The 5 psig setpoint which would be exceeded quickly in the event of a Design Basis Accident (DBA). CPHS setpoint is chosen to cover a spectrum of break sizes, yet be far enough above normal operation to prevent spurious actuation. Technical Specification 2.15 states that if an RPS channel becomes inoperable, it shall be placed in bypass within one hour. It can remain bypassed for up to 48 hours. After that, it must be placed in the tripped condition. The event resulted in the violation of this Specification. Technical Specification 3.5 (3)(c) states that the combined leakage rate of all penetrations and valves subject to type "B" and "C" leakage tests shall be less than or equal to 0.6La. If this cannot be maintained, repairs shall be initiated immediately. If repairs cannot be completed and leakage restored to below 0.6La the reactor shall be shutdown and depressurized until repairs can be completed and local leakage meets this acceptance criteria. The event resulted in the leakage in excess of this limit; therefore Technical Specification 3.5 was violated.

A reactor trip on containment high pressure is provided to assure that the reactor is shut down simultaneously with the initiation of the safety injection system. The setting of this trip is identical to that of the containment pressure high signal (CPHS) which initiates engineered safeguard features (ESF) system. The trip is initiated by a 2 out of 4 coincidence logic from containment pressure channels. The system is designed to maintain containment less than 60 psig during any design basis accident. In addition to starting safety injection, the following functions are achieved due to the CPHS: all containment isolation valves are shut, ventilation is isolated (VIAS), Diesel Generators are started and the sequencers are initiated.

During the worst case design basis accident (double ended guillotine rupture of the Reactor Coolant system (RCS) hot leg), containment pressure is calculated to be less than 55 psig. By design, the A/HCV-742 valve is not closed by a CIAS. Safety analysis calculations were conducted by the NSSS vendor assuming a design basis accident with the test tee uncapped. These calculations used the Updated Safety Analysis Report (USAR) assumptions with the exception that the Technical Specification leak rate of 0.1 percent was assumed rather than the 0.2 percent as stated in the USAR. Calculations combined the design

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort Calhoun Station, Unit No. 1	0 5 0 0 0 2 8 5	8 8	— 0 1 1	— 0 2	0 5 OF 0 7

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

leakage from the containment and the leakage from this open penetration to calculate the site boundary doses over the first two (2) hours of the event. The site boundary doses during the first two (2) hours of the event were 258.6 rem to the thyroid and 6.4 rem whole body. This compares to the 10 CFR 100 limits of 300 rem to the thyroid and 25 rem whole body. This release would activate several high radiation alarms. The Abnormal Operating Procedure for high radiation directs personnel to secure all auxiliary building supply and exhaust fans and to isolate all compartments. An elimination process is then started to pin point the affected area. Once the affected area is located, corrective measures are taken to eliminate the source of leakage.

In the past, the ST-CONT-3 did not specify the exact Swagelok caps which were to be removed and provide signoffs for removal and reinstallation of Swagelok caps. Fort Calhoun utilized maintenance orders with a non documented cap list to verify containment integrity prior to startup. The NRC had also expressed concern that ST-CONT-3 did not provide adequate instruction for the testing being performed (Reference Inspection Report 50-285/87-10 dated May 27, 1987). In response to the NRC concern, Fort Calhoun agreed to review and upgrade ST-CONT-3 prior to the end of the 1988 refueling outage. It was during the diagram verification walkdown of the containment penetrations for the procedure upgrade that the uncapped tee was discovered.

To prevent recurrence:

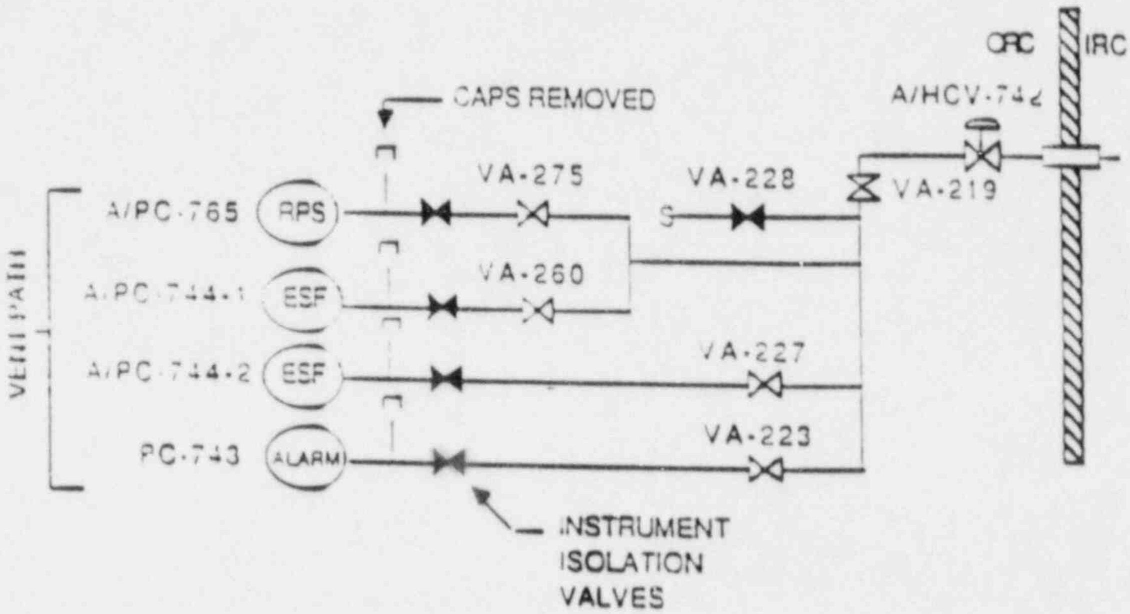
1. ST-CONT-3 will be upgraded to include:
   a. Detailed drawings including all test tees and
   b. Procedural signoffs for the removal and installation of the designated caps.

2. A separate documented and double verified checklist will be written, and this checklist will be performed prior to power operation following a refueling outage to ensure containment integrity.

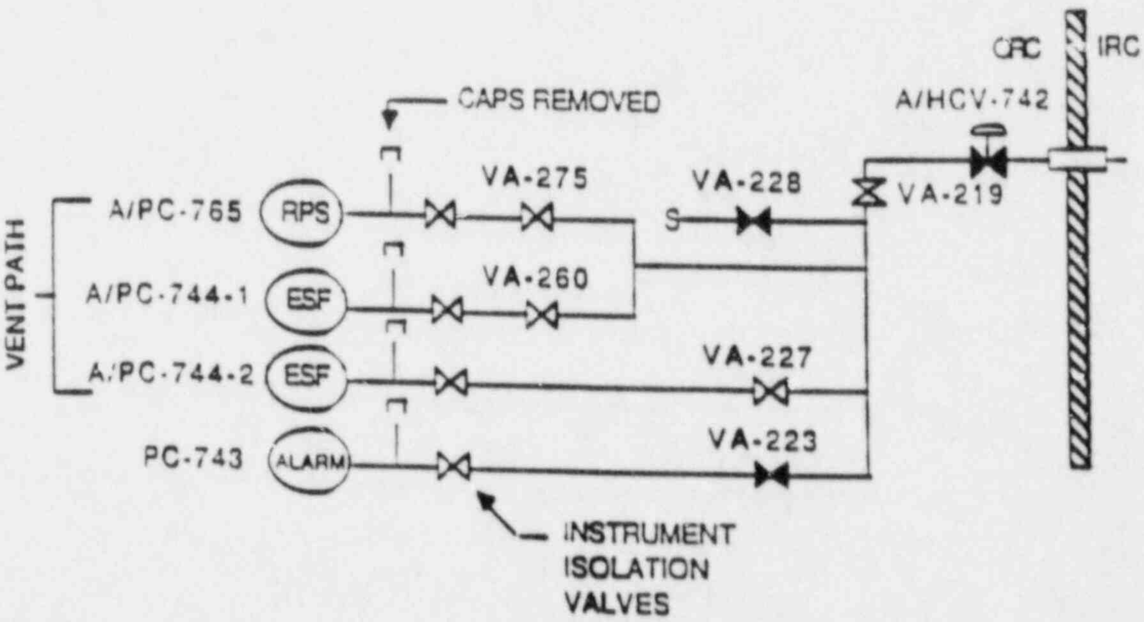
These actions will be complete prior to the end of the next refueling outage.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Fort Calhoun Station, Unit No. 1	0 5 0 0 0 2 8 5	8 8	— 0 1 1 —	0 2	0 6 OF 0 7

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



TEST 1
FIGURE 1



TEST 2
FIGURE 2

NRC Form 366A (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

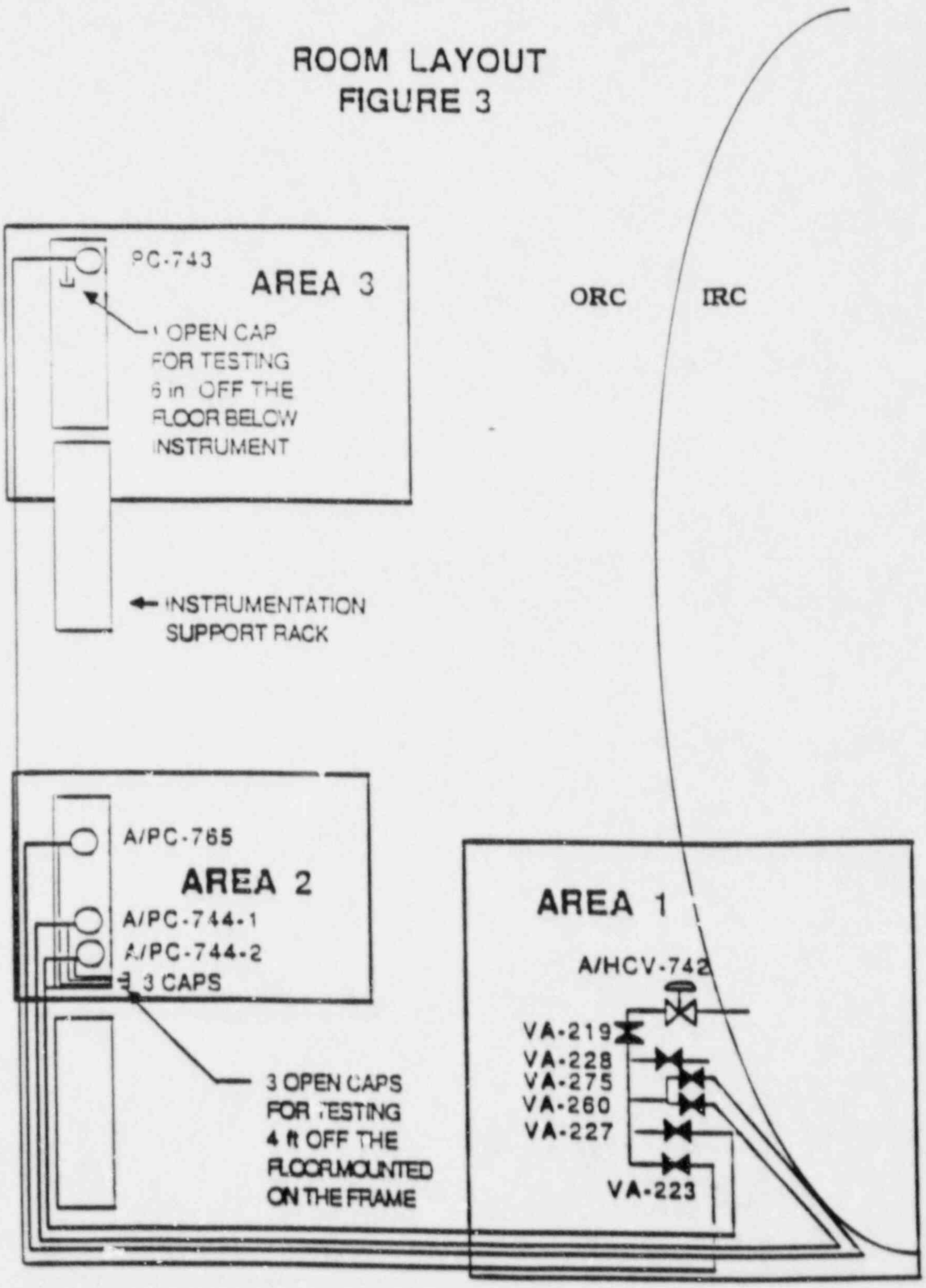
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Station, Unit No. 1	0 5 0 0 0 2 8 5	8 8	— 0 1 1 —	0 2	0 7 OF 0 7

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

ROOM LAYOUT
FIGURE 3



Omaha Public Power District
1623 Harney Omaha, Nebraska 68102-2247
402/536-4000

June 2, 1988
LIC-88-475

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: Docket No. 50-285

Gentlemen:

SUBJECT: Licensee Event Report for the Fort Calhoun Station

Please find attached Licensee Event Report 88-011 Revision 2 dated June 2, 1988. This LER supercedes the previous version in its entirety. This report is being submitted per requirements of 10 CFR 50.73. Changes are denoted by vertical lines in the right hand margin.

Sincerely,

R. L. Andrews

R. L. Andrews
Division Manager
Nuclear Production

RLA/me

Attachment

c: R. D. Martin, NRC Regional Administrator
P. D. Milano, NRC Project Manager
P. H. Harrell, NRC Senior Resident Inspector
INPO Records Center
American Nuclear Insurers

IE22
1/1