Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

May 12, 1992
LIC-92-134L

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

Reference: Docket No. 50-285

Gentlemen:

Subject: Licensee Event Report 92-015 for the Fort Calhoun Station

Please find attached Licensee Event Report 92-015 dated May 12, 1992. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(v)(B). If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/lah

Attachment

c: R. D. Martin, NRC Regional Administrator, Region IV
D. L. Wigginton, NRC Senior Project Manager
S. D. Bloom, NRC Project Engineer
R. P. Mullikin, NRC Senior Resident Inspector
INPO Records Center

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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 5

TITLE (4)
Loss of Shutdown Cooling Flow Control and Flow Indication

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)
									N	0 5 0 0 0
0 4	1 2	9 2	9 2	— 0 1 5	— 0 0	0 5	1 2	9 2		0 5 0 0 0

OPERATING MODE (9): 5
POWER LEVEL (10): 0 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)	X 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)	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
Keith A. Voss, Shift Technical Advisor	AREA CODE 4 0 2 — 5 3 3 - 6 9 3 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)

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

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

On April 12, 1992, while the Fort Calhoun Station was in a refueling outage and the plant was in an off-normal electrical line-up, a 480V bus feeder breaker tripped. This resulted in loss of power to the shutdown cooling flow control valve (FCV-326) controller and shutdown cooling flow indication. Valve FCV-326 fails open on loss of control power. The pump providing shutdown cooling flow was appropriately secured by the Licensed Senior Operator to protect the pump from runout. The operating crew determined the cause of the loss of power, restored power, placed the FCV-326 controller in manual, throttled the valve and restarted the pump within 7 minutes.

The root causes for this event were determined to be failure to have a policy/procedure for performing testing/work during shutdown periods when the plant is in an off-normal electrical line-up, and failure of a surveillance test to adequately list the initial plant conditions required to perform the test.

The effect of this event on nuclear safety was minimal. Shutdown cooling flow was secured for approximately 7 minutes, and the Reactor Coolant System temperature increased by a maximum of 6 degrees F.

Corrective actions will include developing a policy on controlling/evaluating evolutions during off-normal electrical system line-ups and significant electrical system work, and evaluating the need to include electrical system line-up requirements/precautions into appropriate plant procedures.

The Electrical Distribution System at Fort Calhoun Station (Figure 1) includes two safety related 4.16kV buses (1A3 and 1A4) which feed a total of nine 480V buses. Of the nine 480V buses, three (buses 1B3A-4A, 1B3B-4B and 1B3C-4C) can be tied to either of two adjacent 480V buses, allowing them to be fed from either 1A3 or 1A4. The normal elertrical line-up has 4.16kV Bus 1A3 feeding five 480V buses (1B3A, 1B3B, 1B3C, 1B3A-4A and 1B3C-4C) and 4.16kV Bus 1A4 feeding four 480V buses (1B4A, 1B4B, 1B4C and 1B3B-4B).

The power for plant control and instrumentation is provided by the 125V DC System. This system is designed to operate without interruption during design basis accident and adverse environmental conditions. The system includes two separate DC buses. Each 125V DC bus feeds three inverters. Two inverters on each 125V DC bus (Inverters A, B, C, and D) feed the Reactor Protection System and Engineered Safety Feature loads, and one additional inverter on each 125V DC bus (Inverters 1 and 2, not shown on Figure 1) feeds the auxiliary instrument bus loads.

Each 125V DC bus is powered by an AC-DC battery charger in parallel with a battery. The system has a third battery charger that can be aligned to either bus when needed. The normal power supply for the 120v AC instrument buses is through the inverters from the 125V DC buses. Each 120V AC bus is provided with a bypass power supply, a 480/120V transformer, that can power the bus if needed.

The Low Pressure Safety Injection (LPSI) pumps are used to provide shutdown cooling flow. Shutdown cooling flow is controlled by flow control valve FCV-326. This valve fails open on a loss of control power. The flow transmitter power supply for FCV-326 is 120V AC Instrument Panel Bus #2 (AI-42B). AI-42B is normally powered from Inverter 2. The bypass power supply for this inverter comes from a 480V bus (1B4A) which is normally powered from 4.16kV bus 1A4.

On April 12, 1992, while Fort Calhoun Station was in a refueling outage (in Mode 5, refueling shutdown), the plant Electrical Distribution System was in an off-normal line-up. 480V Bus 1B[illegible] had been recently returned to service, but the normal supply breaker for this bus was not available. Therefore, bus-tie breakers were closed (connecting 480V Buses [illegible] B3A-4A and 1B4A) in order to power Bus 1B4A from the opposite safety related [illegible]V bus (1A3).

The 125V DC batteries had been changed out as part of the refueling outage work scope, and the discharge test for the new Battery #2 had just been completed. Battery Charger #2 (which is powered from Motor Control Center MCC-4A1 which is in turn powered from 480V Bus 1B4A) was placed on 125V DC Bus #2 to recharge Battery #2 and Battery Charger #3 was removed from service, per the new battery installation modification instructions. An additional test was to be performed to simultaneously test Battery Charger #2.

A caution in the modification package warned of a possible low voltage alarm on 125V DC Bus #2 during swapping. The low voltage caused Inverter 2 and Inverter D to transfer to their bypass power sources (480/120V transformers) as designed. The operating crew decided to leave both inverters in bypass because they concluded that returning them to their normal power supply would further degrade 125V DC Bus #2 voltage.

A surveillance test (OP-ST-SI-3007) was in progress to test High Pressure Safety Injection (HPSI) System pumps and check valves. This test is performed every refueling outage. One of the check valves tested is the suction header check valve to HPSI pumps SI-2A and SI-2C. For this valve, the test requires that both of these HPSI pumps be run. The two HPSI pumps were started as required by the test.

The overcurrent trip device on the 1B3A feeder breaker initiates a breaker trip once the overcurrent setpoint is surpassed. Once initiated, the trip device proceeds to trip the breaker even if input/load current is subsequently reduced. The additional loading from the off-normal electrical alignment and the conduct of the HPSI System check valve test caused an overcurrent condition to be sensed by the breaker. Approximately 50 seconds after the pumps were secured the feeder breaker to 1B3A tripped free on the long-time delay overcurrent trip. Once the overcurrent condition was sensed, the trip devices proceeded to trip the breaker even though the load was reduced approximately 50 seconds before the breaker tripped. Major loads being supplied through the breaker included: Air Compressor CA-1C, Battery Charger #2, Instrument Bus #2, Component Cooling Water Pump AC-3B, Spent Fuel Pool Cooling Pump AC-5A, the Security Battery Charger, Radiation Monitor RM-060 and several ventilation fans. The operating crew immediately checked the status of shutdown cooling, started an air compressor, and verified that the standby component cooling water pump started.

The Licensed Senior Operator (LSO) noted the loss of control signal to FCV-326 and loss of shutdown cooling flow indication, and realized that such a failure would cause the valve to fail open. The LSO secured the LPSI pump, at 1954, to protect the pump from runout. The Operators recognized that normal shutdown cooling flow was available and could be restored anytime prior to boiling, with flow controlled by alternate means. The Operators also recognized that they had sufficient time to troubleshoot and restore normal shutdown cooling flow control. The LSO then entered the Abnormal Operating Procedure for loss of shutdown cooling (AOP-19). The Shift Technical Advisor provided an estimated time to boil of 40 minutes, and the Shift Supervisor reviewed the Emergency Plan to determine if any Emergency Classifications were required.

The operating crew determined the cause of the loss of power and restored the power in accordance with the Abnormal Operating Procedure. This re-energized the lost buses and the control power to FCV-326. The operating crew placed the FCV-326 controller in manual, throttled the valve to 20% open and restarted the pump at 2001. The maximum temperature increase recorded for the Reactor Coolant System (RCS) was from 105 degrees F to 111 degrees F, in the 7 minutes the shutdown cooling was secured.

The plant manager, operations supervisor and the NRC Resident Inspector were notified of the incident. On April 12, 1992 at 2107, the NRC was notified in accordance with the requirements of 10 CFR 50.72(b)(2)(iii)(B). This report is being submitted pursuant to 10 CFR 50.73(a)(2)(v)(B).

The effect of this event on nuclear safety was minimal. Equipment functioned as designed during the event. When the Licensed Operator noted the potential runout condition of the LPSI pump, it was secured to protect the pump. This Operator action was determined to be appropriate for the circumstances involved. The event resulted in a short termination of shutdown cooling flow through the core. After determining the cause of the problem, and determining that the bus was not faulted, the operators re-energized the bus and put the valve controller in a throttled condition. This allowed shutdown cooling to be restored, which terminated the event.

Shutdown cooling was secured for approximately 7 minutes, and the RCS temperature increased by a maximum of 6 degrees F. The Operators recognized that two LPSI pumps were available if needed, and shutdown cooling flow could be restored even if the cause of the power loss had not been determined and corrected before the estimated time to boil had elapsed.

Following the event, the 1B3A feeder breaker was tested to collect as-found data on the breaker. Results of this post actuation testing indicated that the breaker actuated as designed.

The root causes of this event were determined to be failure to have a policy/procedure for performing testing/work during shutdown periods when the plant is in an off-normal electrical line-up, and failure of Surveillance Test OP-ST-SI-3007 to adequately list the initial plant conditions required to perform the test.

The following corrective actions will be completed:

1. A management policy/procedure on controlling/evaluating evolutions during periods of off-normal electrical system line-ups and significant electrical system work will be developed. This will be completed prior to the 1993 refueling outage. In the interim, if significant off-normal bus alignments are required, testing/work to be conducted during such times will be subject to management review.

2. The need to include electrical system line-up requirements/precautions into appropriate plant procedures will be evaluated. This evaluation will be completed prior to the 1993 refueling outage.

LER 89-01 documents a previous event involving a loss of shutdown cooling.

FIGURE 1

FORT CALHOUN STATION ELECTRICAL DISTRIBUTION SYSTEM