Omaha Public Power District
P.O. Box 399 Hwy. 75 - North of Ft. Calhoun Fort Calhoun, NE 68023-0399
402/636-2000

June 15, 1992
LIC-92-138L

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 92-014 for the Fort Calhoun Station

Please find attached Licensee Event Report 92-014 dated June 15, 1992. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(iv). If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/jrg

Attachment

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

IE22 1

45.5120

NRC FORM 366 (6-89) 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): Fort Calhoun Station Unit No. 1

DOCKET NUMBER (2): 0 5 0 0 0 2 8 5

PAGE (3): 1 OF 0 4

TITLE (4): Reactor Trip Following Maintenance on Moisture Separator Level Instrument

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 5	1 4	9 2	9 2	— 0 1 4 —	0 0	0 6	1 5	9 2	N	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 8

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)	X 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)	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	
Keith A. Voss, Shift Technical Advisor	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
A	S N	L S	M 0 4 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE) ☒ NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On May 14, 1992, a steam leak was identified coming from a capped connection on the turbine trip switch (LA-1303B) on Moisture Separator ST-3C. The level float chamber associated with this switch was valved out and the connection repaired. The upper isolation valve for the float chamber was then cracked open as the first step to returning the float chamber to service. At 1557 hours, the turbine tripped due to a 'false' high moisture separator level indication on LA-1303B. The turbine trip subsequently caused the Reactor Protective System to trip the reactor on Loss of Load.

The overall effect of the event on nuclear safety was minimal. The Loss of Load trip is an anticipatory trip designed to protect plant equipment and is not credited in the plant safety analysis. The plant responded as designed.

The root cause of this event was found to be the Shift Supervisor's decision not to disable the turbine trip circuit prior to opening the float chamber's upper isolation valve.

Corrective actions will include discussion of this event with appropriate personnel, review of the definition of emergency maintenance and the applicability of a Standing Order which addresses non-routine activities requiring formalized plans.

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

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	9 2	0 1 4	0 0	0 2 OF 0 4

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

The Fort Calhoun Station (FCS) Main Steam Turbine consists of a high pressure turbine and two low pressure turbines. After leaving the high pressure turbine, the steam passes through one of four moisture separators to remove water from the steam before it goes to the low pressure turbines.

The High Moisture Separator Level Trip protects the turbine from water induction damage which could result from a postulated failure of the moisture separator level control system. This automatic turbine trip is initiated when the water level in the moisture separator reaches 48 inches on any one of the four moisture separators. This circuit is normally de-energized until the setpoint is reached. The only way to bypass this trip is to lift an electrical lead.

The Loss of Load Trip on the Reactor Protective System (RPS) is designed to protect the Reactor Coolant System (RCS) from an unacceptable heatup and pressurization due to a loss of normal steam removal from the steam generators. This trip occurs when any two of the four turbine stop valves are off their open seat. This trip is bypassed when reactor power is below 15% and is considered an anticipatory equipment protective trip that is not required for reactor safety.

When a trip signal is generated by the RPS, it causes the following to occur: the Control Element Assembly (CEA) clutch power supplies are de-energized allowing the CEAs to drop into the reactor core, the Emergency Diesel Generators are started, the turbine is tripped, and the sequence of event recorders are started.

At approximately 1300 hours, on May 14, 1992, with Fort Calhoun Station in Mode 1 (Power Operation) at 98% power, the Control Room was notified of a steam leak coming from a capped connection on the turbine trip switch (LA-1303B) on Moisture Separator ST-3C. This switch has a level float chamber that will initiate a turbine trip on high level. A Priority 1 (Emergency Maintenance) Maintenance Work Order (MWO) was initiated to tighten the fitting and the level float chamber was valved out, which stopped the steam leak. The MWO was classified as Priority 1 because the Shift Supervisor did not want to leave an instrument capable of initiating a reactor trip valved out overnight and there was concern over the amount of condensation that might occur above the upper isolation valve. Significant condensation in the line could create the potential for a "false" high level indication when returning the equipment to service.

After completing repairs on the connection, a discussion took place as detailed below to determine the method to return the equipment to service. An Instrument and Control (I&C) technician was sent to the Control Room by the I&C Coordinator and was ready to lift the lead to the trip unit. Based on his perception of the relative probabilities of a plant trip, the Shift Supervisor decided not to lift the lead. Further discussions between the Shift Supervisor and the Operations Engineer took place regarding the method of returning the equipment to service. These discussions focused on the isolation valves and the order in which they should be opened. If the lower isolation valve were to be opened first, it was felt the steam entering the float chamber would raise the level float and trip the turbine. Therefore, it was decided to open the upper isolation valve to re-pressurize the float chamber and then open the lower isolation valve. It was believed that the amount of condensation in the line above the upper isolation valve was probably very small and the resulting level in the float chamber would be below the trip setpoint.

The Turbine Building Operator went to valve-in the float chamber. Prior to opening the upper isolation valve, the operator called the Control Room to inform them that he was ready to open the valve. The operator then cracked open the upper isolation valve. At 1557 hours, the turbine tripped due to a "false" high moisture separator level indication on LA-1303B. This "false" signal resulted from condensed steam draining into the float chamber (with the lower isolation valve still closed) and lifting the level float. Following the turbine trip, the operator immediately re-closed the upper isolation valve.

The turbine trip subsequently caused the RPS to trip the reactor on Loss of Load and started the Emergency Diesel Generators. The Control Room operators immediately entered Emergency Operating Procedure (EOP) 00 (Standard Post Trip Actions). At this time, the operators verified the plant was responding as expected for this trip. This procedure was completed at 1559 hours and the determination was made that this was an uncomplicated reactor trip and the crew entered EOP-01 (Reactor Trip Recovery). The Emergency Diesel Generators were shut down per procedure by 1632 hours.

Within one hour of the trip, Maintenance verified that the high moisture separator trip signal was still present. Both the upper and lower isolation valves were still closed. Both isolation valves were then opened and the trip signal cleared.

At 1640 hours, the Resident Inspector was notified of the trip. The NRC was notified of the event on May 14, 1992 at 1700 hours, pursuant to 10 CFR 50.72(b)(2)(ii). This report is submitted pursuant to 10 CFR 50.73(a)(2)(iv).

An investigation of preceding events found that LA-1303B had been calibrated on March 4, 1992. This required the removal and reinstallation of the test fitting cap on LA-1303B. This is the last documented removal of the cap prior to the May 14, 1992 event. The cap was apparently cross-threaded when it was re-installed. During the investigation, the cross-threading issue was reviewed and it was concluded that the cross-threading was an isolated error on the part of the craftsperson performing the calibration and not a generic issue.

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

There were several communication problems during this incident between Operations and the supporting crafts. The event investigation found that there was a breakdown in the oral communications between different groups related to the perceived urgency of repairing the leak and returning the level switch to service. The investigation concluded that the event could have been prevented by lifting the lead to the trip circuit.

The overall effect of the event on nuclear safety was minimal. This event did not involve personnel injuries, releases of radioactive or hazardous materials, or radiation exposure. This event was an uncomplicated reactor trip from full power. The Loss of Load trip is an anticipatory trip designed to protect plant equipment and is not credited in the plant safety analysis. The plant responded as designed for this event and its response was bounded by the Updated Safety Analysis Report (USAR) analysis.

The root cause of this event was found to be the Shift Supervisor's decision not to disable the turbine trip circuit prior to opening the float chamber's upper isolation valve. The decision was based on the assumption that very little condensation had occurred above the upper isolation valve and on a perceived high risk in lifting the lead to disable the switch's turbine trip circuit. Communication problems contributed to the decision not to disable the trip circuit. The applicability of Standing Order G-87, "Non-Routine Activities Requiring Formalized Plans", which addresses the need for Plant Manager approval of "Very High Risk" maintenance activities, was not considered during the event. Utilization of Standing Order G-87 would have provided a mechanism for clarification of communication and resolution of the perceived high risk of lifting the lead for restoring the transmitter to an operable status.

The following corrective actions will be completed:

1. The definition of a Priority 1 MWO and the applicability of Standing Order G-87 to MWOs will be reviewed and clarified as needed by September 1, 1992.

2. The MWO planning and review process will be reviewed and revised, if necessary, by September 1, 1992, to be responsive to Operations Department needs with respect to timely completion of Priority 1 and Priority 2 MWOs.

3. Standing Order G-87 will be re-emphasized by September 30, 1992, in departmental meetings.

4. Training will be provided to Maintenance and Operations supervisory and support staff, including a discussion of Standing Order G-87, by October 1, 1992, as a lessons learned topic.

LER 86-001 describes the most recent previous occurrence at FCS of an automatic RPS trip from power.