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

December 19, 1990
LIC-90-0971

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 90-26 for the Fort Calhoun Station

Please find attached Licensee Event Report 90-26 dated December 19, 1990. This report is being submitted pursuant to requirements of 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/tcm

Attachment

c: R. D. Martin, NRC Regional Administrator
W. C. Walker, NRC Project Manager
R. P. Mullikin, NRC Senior Resident Inspector
INPO Records Center

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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)	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): Manual Reactor Trip Due to Loss of Instrument Air Pressure

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)
1 1	1 9	9 0	9 0	— 0 2 6 —	0 0	1 2	1 9	9 0	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)	[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 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
B	L D	P S F	X 9 9 9	N					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

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

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

On November 19, 1990 while the plant was operating at 100% power, a pipe joint at an isolation valve in the Turbine Building instrument air header failed. The resulting loss of instrument air pressure caused a feedwater transient which ultimately resulted in manual tripping of the reactor. The root cause for this event was improper installation of the isolation valve (inadequate joint insertion coupled with poor soldering technique) under a modification in 1984. The contributing cause was inadequate control of the installation process.

The completed corrective actions for this event included repairing the failed joint, installing braces around applicable header isolation valves, checking the turbine building header for leaks, and repairing one additional leak. The administrative controls over modifications have been improved since 1984. The long term corrective actions for this event include reinstalling the isolation valves using better techniques, providing better training on soldered joints, and discussing this event in licensed and non-licensed operating personnel requalification training.

The Instrument Air system at Fort Calhoun Station supplies pneumatically operated components within the plant. The system normally operates with one of three air compressors in service. Depending on system requirements, this compressor will cycle as necessary from 0% load at 108 psig to 50% load at 102 psig to 100% load at 98 psig. If system pressure drops to 94 psig, a second air compressor will automatically start. The third air compressor can be started from the control room as needed.

Instrument Air in the Turbine Building is supplied to the components through a 2 inch diameter ring header. Some of the components supplied in the Turbine Building are the feedwater regulating valves, the condenser hotwell level control, and condensate recirculation control valves. There are five isolation valves in this header which were installed using soldered joints through modification MR-FC-84-57 in April, 1984.

The feedwater regulating system maintains the steam generator downcomer level within acceptable limits by positioning the feedwater regulating valves (FWRV) supplying each steam generator. The controller monitors feedwater flow, steam flow, and downcomer level. The output signal for each controller provides a signal to the air operated FWRVs. To allow for decay heat removal, the FWRVs are designed to automatically rampdown to 8 (±2) percent open in the event of a reactor trip. The operators can take manual control of the feedwater regulating system at any time. If the operators have manual control over the FWRVs they will not rampdown. Instrument Air to the main feedwater valves is isolated at 75 psig and will maintain the valves in their "as-is" position. The main feedwater bypass valves fail closed during a loss of Instrument Air.

At approximately 1623 hours on November 19, 1990, the plant was operating at 100% power. A pipe joint separated in the Turbine Building Instrument Air header. Numerous alarms indicating low instrument air pressure were received in the Control Room. The standby air compressor started automatically and a control room operator started the third air compressor. At 1626, a licensed operator saw the steam generator level rapidly falling and took manual control over the feedwater regulating valves in an attempt to regain control. At 1626, air system pressure was observed to be about 73 psig with all three air compressors running. (A post trip review of the transient showed instrument air system pressure had decreased to approximately 70 psig.) At 1629, the Shift Supervisor directed a licensed Control Room operator to manually trip the reactor due to an inability to gain manual control of feedwater flow. The emergency diesel generators started and remained available as designed.

The operating crew immediately entered Emergency Operating Procedure EOP-00, Standard Post Trip Action. Following the trip, the steam generator level underwent the normal post-trip shrink and then immediately started increasing at a rapid rate. This was because the FWRVs had not ramped down and two main feedwater pumps were still in operation. The FWRVs were isolated using the motor operated valves, and the operators manually started feeding the steam generators using Auxiliary Feedwater Pump FW-6. Since the motor operated valves do not close instantaneously, the steam generator levels increased to approximately 100% narrow range, then started to decrease due to steaming through the atmospheric dump valve. At 1633, the break in the instrument air header in the turbine building was isolated and system air pressure started to rise.

At 1639, the operating crew entered Emergency Operating Procedure EOP-20, Functional Recovery Procedure, which contains guidance for the loss of instrument air. At 1723, plant conditions were stable enough for transition to Emergency Operating Procedure EOP-01, Reactor Trip Recovery. The NRC was notified of the plant trip at 1728 pursuant to the requirements of 10CFR50.72(b)(2)(ii). At 1755, the isolated section of the Instrument Air header was repaired and ready to be re-pressurized. At 1758, the header was returned to normal. At 1803, the operating crew exited Emergency Operating Procedure EOP-01, Reactor Trip Recovery. The plant was maintained at Hot Shutdown conditions during the post-trip review process.

The transient that occurred in the steam generators can be attributed to the loss of Instrument Air pressure. The operators transferred control of FWRVs to the auxiliary control station prior to the trip in an attempt to regain adequate feedwater flow to the steam generators. The system is designed so that if control is transferred to the auxiliary controller, the automatic rampdown of the FWRVs is overridden. Even if the control had not been transferred, the rampdown would not have occurred. The isolation of Instrument Air to the FWRVs occurred automatically at 75 psig (as Instrument Air pressure decreased prior to the reactor trip) and resulted in the FWRVs staying in the pre-isolation position. There are motor operated valves that can be used to isolate this feedpath to the steam generators. The steam generators were fed by the auxiliary feedwater system, which uses a different flowpath than main feedwater. The plant systems and components operated as designed, and there were no inappropriate personnel actions during this event.

The Update Safety Analysis Report (USAR) defines the loss of feedwater flow incident as a reduction in feedwater flow to the steam generator when operating at power without a corresponding reduction in the steam flow from the steam generators. This type of event could be interpreted as a partial loss of feedwater flow caused by the instrument air failure; the plant response was within the parameters defined by the USAR. The loss of Instrument Air pressure did not present a significant hazard to the plant since the system is not required for a safe plant shutdown. Nuclear safety was not affected by this event.

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 0	— 0 2 6 —	0 0	0 4 OF 0 5

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

The initiator for this event was the failure of a pipe joint at one of the isolation valves (IA-201) on the instrument air header in the Turbine Building. When properly installed, the connecting pipes should be inserted one and one half inches into the valve body and then soldered. A post-trip review of the failed piping showed the pipe was inserted less than one half inch into the valve body. The solder did not fully penetrate the joint and a majority of the solder was collected on the externals of the mating portions, resulting in a weak joint. Normal stresses and vibrations resulted in eventual failure of the joint. The root cause for this event was therefore improper installation of the isolation valve (inadequate joint insertion coupled with poor soldering technique). The contributing cause was inadequate control of the installation process.

The following corrective actions have been completed for this event:

1. The failed joint was repaired under Maintenance Work Order (MWO) 907667 using proper solder techniques and a full joint fit-up.

2. The Turbine Building instrument air header was checked for leaks under MWO 904732. One additional solder joint required repair under MWO 904743.

3. A temporary modification (90-027) was installed on the Turbine Building instrument air header. This modification consists of braces that hold the header piping to the isolation valves. This will prevent separation of the piping in the event of another joint failure. An Engineering Change Notice (ECN #90-509) was initiated to permanently incorporate these braces into the plant design.

4. One other instrument air isolation valve (IA-201) was removed to determine the original installation technique used. There was less than full pipe insertion, but the solder penetration appeared to be adequate. The valve was reinstalled. This additional cause verification was done under MWO 904744.

5. Since the valves were installed in 1984, improved controls over the modification process have been implemented, including enhanced procedures, better training, and additional plant planning and engineering personnel.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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 0	— 0 2 6	— 0 0	0 5 OF 0 5

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

The following corrective actions will be completed:

1. The pipe bracing installed under temporary modification 90-027 will be permanently incorporated into the plant design by February 15, 1991.

2. Hands-on training for installation and testing of solder joints will be incorporated into the maintenance training program for pressure equipment personnel prior to the 1991 refueling outage.

3. The five isolation valves installed under modification MR-FC-84-57 will be removed and reinstalled ensuring proper fit-up and solder technique. This will be completed prior to the end of the 1991 refueling outage.

4. This event will be discussed in licensed and non-licensed operator requalification training. This training will be completed by July 1, 1991.

There have been 2 previous LERs (87-25 and 87-33) dealing with the Instrument Air system; however, neither involved a unit trip.