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

September 21, 1992
LIC-92-261L

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 92-028 for the Fort Calhoun Station

Please find attached Licensee Event Report 92-028 dated September 21, 1992. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(iv), 10 CFR 50.73(a)(2)(i)(B), 10 CFR 50.73(a)(2)(ii), 10 CFR 50.73(a)(2)(v)(D) and 10 CFR 50.73(a)(2)(vii)(D). If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/lah

Attachment

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

IE22 1/1

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

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

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 9

TITLE (4): Partial Loss of Load Resulting in Pressurizer Safety Valve Lift and Subsequent Reactor Trip

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 8	2 2	9 2	9 2	0 2 8	0 0	0 9	2 1	9 2	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)	[X] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[X] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[X] 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
Daniel J. Rosloniec, Shift Technical Advisor	402 533-6899

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	IT	CNV	A038	Y					
D	AB	RV	C711	Y					

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 August 22, 1992 at 0152 (CDT) while the plant was operating at 100% power, the Reactor Protection System automatically tripped the reactor on Thermal Margin/Low Pressure (TM/LP). The event was initiated by the failure of an AC to DC power converter which affected the first stage turbine pressure signal and resulted in the repositioning of the Turbine Control Valves. The decrease in secondary steam demand caused an increase in Reactor Coolant System (RCS) pressure which was terminated by a premature opening of one of two Pressurizer Safety Relief Valves, followed by the Reactor trip.

The root cause of this event was the failure of an AC to DC power converter in the Turbine Electrohydraulic Control (EHC) panel. The root cause of the premature lift of RC-142 was that the (laboratory) test environment in which valve set pressure qualification was performed, did not provide an adequate representation of the actual field environment.

Corrective actions include a modification to change the [illegible] source for EHC pressure transmitters, reducing the pressure setpoint for initiat[illegible] High Pressurizer Pressure trip and Power Operated Relief Valve operation and adjus[illegible], Pressurizer Safety Valve set pressures using revised test procedures.

BACKGROUND

The Reactor Protective System (RPS) monitors certain critical plant parameters and compares them to predetermined setpoints. If one or more of the monitored parameters reaches its setpoint on two of four channels, the RPS will initiate a reactor trip. There are 12 different reactor trips that can be initiated by the RPS. The trip units of interest for this event are High Pressurizer Pressure and Thermal Margin/Low Pressure.

A Reactor trip signal is initiated when two of four High Pressurizer Pressure channels approach 2400 psia. This trip is provided, in conjunction with the Pressurizer and steam system safety valves, to prevent the Reactor Coolant System (RCS) from exceeding 110% of its design pressure of 2500 psia.

The Thermal Margin/Low Pressure (TM/LP) trip signal is provided to prevent operation when the Departure from Nucleate Boiling Ratio (DNBR) is less than 1.18. A TM/LP trip signal is initiated when two of four channels indicate that RCS pressure has reached a low pressure trip limit. The trip limit used is the higher of a fixed (1750 psia) and a variable low pressure trip limit. The variable low pressure trip limit is calculated using a combination of RCS temperature, pressurizer pressure, core power and axial shape index.

Two Power Operated Relief Valves (PORV) (PCV-102-1 and PCV-102-2) are designed to provide sufficient relief capacity during RCS high pressure transients to prevent the opening of the Pressurizer Safety Valves. The PORVs operate when two of four of the High Pressurizer Pressure channels approach 2400 psia.

Two Pressurizer Code Safety Valves provide over-pressure protection for the RCS. They are totally enclosed, spring loaded safety valves meeting ASME code requirements. A loop seal is provided to minimize valve leakage. The Technical Specifications require the lift settings of one valve (RC-142) to be adjusted to ensure valve opening at 2500 psia +/- 1% and the second (RC-141) at 2545 psia +/- 1%.

The pressurizer quench tank is designed to collect and condense the normal discharges from the pressurizer during operation and to collect non-condensable gas discharges from the reactor vessel head or the pressurizer during post-accident situations. In either case, the pressurizer quench tank prevents normal relief or safety valve discharges from being released directly to the containment atmosphere and/or sump. The steam discharged from the pressurizer is injected underwater by a sparger to enhance condensation by uniform distribution. The pressurizer quench tank can condense the steam discharged during a loss-of-load incident without exceeding the rupture disc setpoint, assuming normal blowdown of the relief valves.

The Turbine Electrohydraulic Control (EHC) System supplies control signals to the turbine steam admission valves during startup, normal operation, shutdown, testing and transient conditions. The Turbine Control Valves regulate steam flow to the Turbine. Two pressure transmitters combine to regulate positioning of the turbine control valves. Pressure transmitter PT-945 provides a pressure feedback signal and PT-943 provides throttle pressure compensation. The power supply for these two pressure transmitters is located in Turbine EHC Panel #2 (AI-50).

EVENT DESCRIPTION

On August 22, 1992 the Fort Calhoun Station was in Mode 1 (Power Operation) operating at 100% power. At approximately 0152 (CDT), a 115V AC to 28V DC power converter failed inside AI-50 (EHC Cabinet). This converter powered turbine control transmitters PT-945 (first stage pressure feedback to the control valve amplifier) and PT-943 (throttle pressure sensor for throttle pressure compensation). Inaccurate feedback signals from the de-energized pressure transmitters caused the Turbine Control Valves to move from a 40% open position to an approximately 22% open position resulting in a partial loss of load. The change in control valve position resulted in a generator load drop of 120 MW(E).

The first control room annunciator indication of a malfunction occurred several seconds later when a low steam generator level alarm was received for both steam generators. A rapid reduction in steam flow will result in a low level indication due to the design of the level instrumentation (commonly referred to as a "shrink" condition). The Reactor Operator for the secondary system immediately started actions to verify that a loss of feedwater had not occurred. However, after the steam generator levels appeared to be recovering, the primary board operator noticed RCS temperature and pressure increasing.

The mismatch between steam demand and reactor power due to the partial closure of the turbine control valves caused an increase in Reactor Coolant System (RCS) pressure and temperature. This resulted in one of the two Pressurizer Code Safety Valves (RC-142) opening at approximately 2398 psia. This valve has a required setpoint of 2500 psia +/- 1%. Secondary system pressure increased to 1003.8 psia on the "A" Steam Generator which resulted in one or more Main Steam Safety Valves opening.

Just before RC-142 opened, one of the four High Pressurizer Pressure Trip Units tripped providing one of two signals required to initiate a Reactor trip and opening of the PORVs on high pressure. The subsequent rapid depressurization of the RCS due to RC-142 opening, cleared the High Pressurizer Pressure indication and, shortly thereafter, resulted in the RPS automatically tripping the reactor on Thermal Margin/Low Pressure (TM/LP). The Reactor trip occurred 37 seconds after the power converter failed. Upon receiving the Reactor trip, the Main Turbine tripped and the Emergency Diesel Generators started.

Except for the premature lift, RC-142 functioned as designed. RCS pressure dropped to 1721 psia before it started to recover, approximately 1 minute into the event. Pressurizer level responded normally.

The steam from RC-142 discharged to the pressurizer quench tank. The pressurizer quench tank peak pressure was 12.51 psig with a level increase of approximately 5% (i.e., from 77% to 82% level).

Following the turbine trip, Fuse F-5 in AI-50 blew as a result of the power converter failure. This caused some turbine valves to indicate a mid-position.

The operations crew implemented Emergency Operating Procedure EOP-00, "Standard Post-Trip Actions," and then proceeded to EOP-01, "Reactor Trip Recovery." All safety functions were satisfied and all major plant equipment performed as expected with the exception of RC-142 having lifted prematurely.

The plant was stabilized in hot shutdown (Mode 3) and maintained at normal RCS temperature and pressure limits until in-situ testing of both Pressurizer Safety Valves was completed. The in-situ testing was completed on August 25, 1992 and a plant cooldown was then initiated to allow removal of the valves. The valves were then shipped to Wyle Laboratories for inspection and testing.

The NRC was notified of this event on August 22, 1992, at 0358, pursuant to 10 CFR 50.72(b)(2)(ii). This Licensee Event Report (LER) is being submitted pursuant to the following federal regulations:

1) 10 CFR 50.73(a)(2)(iv), due to the actuation of the Reactor Protective System and automatic start of the Emergency Diesel Generators;

2) 10 CFR 50.73(a)(2)(i)(B), due to a Pressurizer Safety Valve opening below the pressure range specified in Technical Specification 2.1.6(1);

3) 10 CFR 50.73(a)(2)(ii), due to the premature opening of a Pressurizer Safety Valve constituting a degradation of the reactor coolant pressure boundary;

4) 10 CFR 50.73(a)(2)(v)(D) and 10 CFR 50.73(a)(2)(vii)(D), due to the conclusion that the cause of the premature opening of RC-142 also resulted in a similarly low effective set pressure for RC-141 (indicating that both valves had potentially been outside their Technical Specification required settings).

SAFE[illegible]SSESSMENT

The Nu[illegible]ear Steam Supply System (NSSS) response to this event was normal for a loss-of-load event and was bounded by the Updated Safety Analysis Report (USAR) accident analysis. Peak RCS pressure was approximately 2398 psia, peak RCS hot leg temperature was 603 degrees F, and peak Steam Generator Pressure was 1003.8 psia.

USAR Section 14.9 "Loss of Load" indicates that the acceptance criteria for this transient are:

a. The peak RCS pressure remain below 110% of the design pressure (i.e., below 2750 psia),

b. A sufficient thermal margin must be maintained in the hot fuel assembly to assure that departure from nucleate boiling does not occur.

The significance of the EHC transmitter power converter failure is that a non-safety related equipment failure challenged safety-related equipment.

The consequence of the safety valve opening at approximately 2398 psia is minimal since the Pressurizer is equipped with two Power Operated Relief Valves (PORV's) which are designed to operate at approximately 2400 psia. The significance of challenging a Pressurizer Safety Valve is that it increases the probability of experiencing a valve malfunction and creating an unisolable Loss of Coolant Accident (LOCA). A function of the PORV's is to limit the number of challenges to the Pressurizer Safety Valves.

CONCLUSIONS

A root cause analysis was initiated to establish the cause of the partial loss of load. Prior to 1978, Pressure Transmitters PT-939, PT-943, PT-944 and PT-945 were "Schaevitz" brand pressure transmitters. These pressure transmitters had a high failure rate which had caused a Reactor trip and several near-misses. The Schaevitz transmitters were powered from an oscillating power supply inside AI-50. Power was provided by 115V AC house power and backed up by the Permanent Magnet Generator which is driven directly off the Main Turbine shaft.

In 1978, Design Change Request (DCR) 76-19 replaced the Schaevitz transmitters with "Rosemount" brand transmitters. However, the Rosemount transmitters could not use Schaevitz' oscillating power supply and backup power supply. Per General Electric's design, the 115V AC power source was then used to supply the new pressure transmitters via 115V AC to 28V DC "Acopian" brand converters "A" and "B." Though the new transmitters were more reliable, their power supply did not have a redundant backup supply via the Permanent Magnet Generator.

During the 1990 Refueling Outage, the periodic (every refueling outage) calibration of these pressure transmitters uncovered a high ripple output from power converter "A" on card "A86" inside AI-50. The power converter was replaced under Maintenance Work Order (MWO) 902334. The power converter was an "Acopian" brand, Model 28E15, "Miniature AC to DC Power Module." This power converter was replaced after approximately 12 years of service. Approximately two years after the "A" power converter was replaced, the "B" power converter failed, causing the partial loss-of-load event and subsequent reactor trip discussed in this LER. This pressure transmitter loop and power converter output had been successfully calibrated during the 1992 Refueling Outage approximately four months earlier.

The root cause of this event was determined to be the failure of AC to DC power converter "B" on card "A86" in AI-50 (EHC Cabinet).

The modification in 1978 which removed the backup power supply to the transmitters was determined to be a contributing cause. DCR 76-19 did not address the consequences of removing the backup power supply to these pressure transmitters. This design was supplied by General Electric. The modification process has changed significantly at Fort Calhoun Station since this DCR was written. Procedures are currently in place to thoroughly review and approve vendor designs.

In addition to the analysis of the cause of the partial loss of load, an investigation was initiated to establish the root cause of the premature opening of Pressurizer Safety Valve RC-142. RC-141 and RC-142 are Crosby Style HB-BP-86, size 3K6, Self-Actuated Safety-Relief Valves. The body and bonnet of the valves are carbon steel and the disc, nozzle and spindle are stainless steel. Set pressure is controlled by varying the compression of the spring by means of the adjusting bolt. Turning the adjusting bolt one flat (1/6 turn) changes the set pressure approximately 75 psi. Differential thermal expansion of the valve components can also affect the set pressure, as changes in the dimensions of the components can vary the compression of the spring.

With the plant in hot shutdown, Furmanite Corporation was brought onsite to perform in-situ setpoint testing on the pressurizer safety valves with their Trevitest test equipment. The Trevitest was used to determine the lift pressures with the valve in the as-installed configuration, and then trend lift pressure versus temperature as the safety valve heated up. The in-situ testing showed that set pressure varied with temperature changes in the valve body. The pressurizer safety valves were then sent to Wyle Laboratories for disassembly, inspection and set pressure qualification.

In an attempt to substantiate the effect of valve temperature distribution on set pressure, the plant insulation was sent to Wyle with the valves to be used during testing. An experimental low temperature test was devised in an attempt to test the set pressure with the valves as close to the field measured temperatures as possible. The results of this test were inconclusive because the valve temperatures could not be kept stable and constant when high pressure saturated steam was applied. This was partially due to test equipment limitations. Qualitatively, as the valve temperatures rose, there was an initial increase in the set pressure (due to nozzle thermal growth) and then a decrease (due to body thermal growth). However, when the valve was tested with the valve in thermal equilibrium with the high pressure steam, the valve "pop" pressure was repeatable within +/- 1%.

Previous set pressure testing at Wyle (including testing performed in July 1992, following a previous Pressurizer Safety Valve lift), had utilized an arrangement in which an environmental chamber was installed over an uninsulated bonnet, and fiberglass cloth was placed over the valve body. A revised testing method was used for the August, 1992 tests and was conducted at Wyle with the plant insulation installed on the valves and the valves at equilibrium temperatures. This testing indicated a lower effective set pressure for both valves. It appears that these valves were set at a lower effective set pressure during previous testing because of the different methods used in controlling temperature distribution throughout the valve during testing. The set pressures of both safety valves were raised by approximately one flat (approximately +75 psi, or about 3%).

It was concluded that the root cause of the premature lift of RC-142 was that the previous valve (laboratory) test environment did not provide an adequate representation of the actual field environment. Specifically, the temperature distribution throughout the valve body and bonnet differed significantly from the as-installed temperature distribution and was the cause for the effective lift pressure being lower than the test set pressure. Insulating the valve using the valve's actual plant insulation during the most recent set pressure tests provided a more accurate representation of the actual field environment.

A contributing factor is the fact that the FCS Pressurizer Safety Valves have a carbon steel valve body and bonnet and a stainless steel nozzle, disc and spindle. This material difference is believed to accentuate the problem because of different thermal expansion coefficients.

Significant differences in temperature distribution during set pressure testing in comparison to installed conditions is only expected to be a concern with insulated safety valves that are installed on loop seals in high temperature systems, or with uninsulated valves on high temperature systems that may be subjected to forced cooling from ventilation drafts. At Fort Calhoun Station, the only safety valves that this is applicable to are the Pressurizer Safety Valves, RC-141 and RC-142.

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.

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

CORRECTIVE ACTIONS

The following corrective actions have been or will be completed:

1) Engineering Change Notice (ECN) 92-308 was installed prior to plant startup to provide power directly from the power bus within Panel AI-50 to pressure transmitters PT-939, PT-943, PT-944 and PT-945. Panel AI-50 receives its power from the Permanent Magnet Generator with back-up from the Station 120V AC.

2) The setpoint for initiating a High Pressurizer Pressure trip in the RPS has been decreased to 2350 psia and the setpoint for PORV operation has been decreased to 2350 psia for Cycle 14 to increase the available margin between the PORVs and the Pressurizer Safety Valve set pressures. The setpoint change was justified by Engineering Analyses EA-FC-92-066 and EA-FC-92-067 and a 10 CFR 50.59 evaluation. Appropriate EOPs and Abnormal Operating Procedures (AOPs) were revised and Operator training conducted to address the setpoint change.

3) The test procedures for set pressure testing of the Pressurizer Safety Valves have been revised to ensure that adequate control of valve temperature distribution is maintained during set pressure testing. The set pressures of RC-141 and RC-142 were adjusted using the revised qualification procedure.

4) A failure analysis of the failed Acopian AC to DC power module will be performed by March 31, 1993.

5) The EHC System has been reviewed and single failure components (i.e., components whose single failure could cause a Reactor trip) have been identified. Upgrades to the Preventive Maintenance Program, monitoring enhancements or possible modifications/replacements which would enhance system reliability are being factored into ongoing programs.

6) An evaluation will be performed by December 31, 1992 of the options for possible relocation of the Pressurizer Safety Valves to eliminate the loop seal (reference LER 92-023).

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

PREVIOUS SIMILAR EVENTS

This is the second occurrence of a premature lift of a pressurizer safety valve at Fort Calhoun Station. The first occurrence was on July 3, 1992 and involved the same valve, RC-142. The initial investigation revealed that RC-142 had lifted and that it sustained damage to its internals including indications of valve chatter, failure of the bellows assembly and malfunction of the setpoint adjusting bolt locknut. Previous industry test data regarding temperature effects on setpoint indicated that no more than a 1% setpoint shift would be expected due to temperature effects. Discussions with the valve manufacturer had indicated that only bonnet temperatures would be significant with respect to setpoint shift. Additionally, the obvious malfunction involving the adjusting bolt (which allowed the valve setpoint to shift during the event) tended to mask the significance of the temperature related effect. The Pressurizer Safety Valve bonnets were instrumented with thermocouples to verify the valve operating conditions following startup from the event.

The July 3, 1992 event was initiated by an inverter problem which caused a momentary loss of power to the instrument bus that supplies power to the Turbine EHC System, resulting in closure of the Turbine Control Valves. The event was reported under LER 92-023. LER 86-001 also reported a Reactor trip following a loss of power to the Turbine EHC panel.