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 5

TITLE (4)

Potential Loss of Pressurizer Pressure Transmitters Following Pressurizer Spray Break

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	0 9	8 8	8 8	0 0 5	0 1	0 9	3 0	8 8	N	0 5 0 0 0
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OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 7 0	[ ] 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)	[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 (AREA CODE)	
Robert F. Mehaffey - Supervisor Electrical I & C	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)		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)

This LER is being provided as a supplement to LER-88-005. Changes to the original LER are denoted by vertical lines in the right hand margin.

At 1600 hours on March 9, 1988, with reactor power at 70 percent, Omaha Public Power District (OPPD) discovered a condition by which a specific Loss of Coolant Accident (LOCA) could incapacitate instrumentation feeding a protective function. Analysis performed indicates that a LOCA where the break occurs at specific pressurizer spray line locations could possibly render two channels of pressurizer pressure inoperable, leaving only two channels operable. With one of the unaffected channels initially out of service, sufficient instrumentation would not exist to complete the required 2-out-of-4 logic for reactor trip and safeguards initiation from the decreasing pressurizer pressure.

Additional evaluation per LER 88-005, Rev. 0, has determined that pressurizer loop spray piping from RCS cold leg loops 1B and 2A is routed outside the biological shield. The location of this piping is such that jet impingement and pipe whip due to a line break could damage safety related electrical equipment which must function to mitigate the consequences of this spray line break (small break LOCA).

OPPD has analyzed these situations and concluded that safe operation may be continued due to: (1) availability of alternate leak detection methods, (2) operator awareness of the potential consequences of a spray line break, and (3) the low probability of a catastrophic break.

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This LER is being provided as a supplement to LER-88-005. Changes to the original LER are denoted by vertical lines in the right hand margin.

At 1600 hours on March 9, 1988, with reactor power at 70 percent, Omaha Public Power District (OPPD) discovered a condition by which a specific Loss of Coolant Accident (LOCA) could incapacitate instrumentation feeding a protective function. This is outside the design basis as detailed in Updated Safety Analysis Report (USAR) section 7.2.9 and 5.8.4. It was determined that this condition presented no safety consequences, but was reportable under 10 CFR 50.72 (b)(1)(ii)(B). The NRC was notified at 1647 hours on March 9, 1988.

Condition No. 1 - Pressurizer Level Transmitters

This potential condition was discovered during a review of high energy line concerns resulting from a modification to the pressurizer level transmitters. The review indicated that a pressurizer spray line break at specific locations could possibly render two channels of pressurizer pressure inoperable. Pressurizer pressure is used as an input to the thermal margin/low pressure (TM/LP) and high pressurizer pressure trips of the Reactor Protective System (RPS), and also feeds the pressurizer pressure low signal (PPLS) in the Engineered Safety Features (ESF) logic. Both the RPS and ESF function through a 2-out-of-4 logic system for actuation.

USAR section 7.2.9 states that the plant is protected against common-mode failures by the selection of instrumentation sensor locations and lines which provides physical separation of the channels. With one of the operable pressurizer pressure channels (i.e., not subjected to steam impingement from the break) out for maintenance or already inoperable (as allowed by Technical Specifications), the plant would be placed in a condition outside the design basis of USAR section 7.2.9 after a pressurizer spray line break.

If the line failure is a leak, the leak would be detectable through various means. The leak would show up in the daily performance of Surveillance Test ST-RLT-3, "Reactor Coolant System Leak Rate Test", as a noticeable jump or increasing trend. The maximum unknown leakage allowed by ST-RLT-3 before corrective action is required is 0.3 gallons per minute (gpm); the Technical Specification limit is 1.0 gpm. Depending on the size of the leak and the length of time it exists, abnormal trends may also be observed in the containment sump level, containment dewpoint, and on containment radiation monitors RM-050 and RM-051. A LOCA-qualified ambient temperature detector, normally available on Emergency Response Facilities computer as point T888, is located in the vicinity of the pressurizer pressure transmitters, and could be used to detect local heating from a leak. Upon discovery of a leak, a controlled shutdown could be initiated.

If a catastrophic break were to occur, three failure modes of the transmitters (Foxboro Model N-E11GM-HIE2-ADL) are possible: low, high, and as-is.

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

If the transmitters with two operable channels acting as backups, fail low, they would fulfill their design function by tripping the reactor on as the reactor coolant system (RCS) pressure falls below the TM/LP setpoint. ESF will also be actuated, as designed, by the PPLS signals generated by both the failed and operable transmitters. Failure toward low pressure is the most probable failure mode, based on information from the transmitter manufacturer.

If the transmitters fail high, they would cause a simultaneous reactor trip and opening of the Power Operated Relief Valves (PORV's) due to the perceived high pressurizer pressure. The PORV's or their block valves must be manually closed by a control room operator to minimize the pressure loss in the RCS. ESF is not actuated by high pressurizer pressure, so until containment pressure reaches the setpoint for the Containment Pressure High Signal (CPHS), there will be no containment spray or safety injection. The CPHS will initiate safety injection, containment isolation, and steam generator isolation, but due to the ESF logic, containment spray will not be initiated without a PPLS along with the CPHS. The containment isolation signal will, however, open the component cooling water valves to the containment air cooling units, and therefore assist in controlling containment pressure. At approximately the same time, each of the two operable pressurizer pressure transmitters will see the RCS pressure fall and send a PPLS to the ESF circuitry to initiate ESF functions. If one of the unaffected pressure channels is initially out of service, containment spray must be actuated manually by a control room operator if required to control containment pressure. With high failure of the transmitters, manual actuation of ESF may be required if CPHS is not received early enough in the transient, or if one of the unaffected pressure channels is out of service. Manual actions may include closing the PORV's or their block valves, actuation of containment spray, and actuation of the entire ESF system.

In the case of the transmitters failing as-is, or anywhere within the setpoints of TM/LP on the low side and high pressurizer pressure on the high side, the two remaining pressure channels will trip the reactor on TM/LP. If one of the unaffected pressure channels is initially out of service, the reactor will trip on CPHS, due to the pressurization of the containment building. As in the case of high failure, CPHS will initiate safety injection, containment isolation, and steam generator isolation, but manual actuation of containment spray will be necessary if required to control containment pressure.

It must be noted for the high and as-is failure modes, that a reactor trip on CPHS and associated ESF actuations will not occur as early as a trip on TM/LP, following a pressurizer spray line break.

Licensed operations personnel have been alerted to this problem, the possible consequences, and recommended operator actions by means of training hotline 88-021. The actions recommended in the event of a spray line break (ensuring reactor has tripped and ESF has actuated) are already part of the Emergency Operating Procedures (EOPs) for reactor trip and LOCA; thus, the EOP's already provide guidance for this scenario.

The plant can continue to be safely operated in light of this design deficiency due to the following factors:

1. A catastrophic pressurizer line break is considered unlikely; failure of the line is more likely to start as a crack, propagate into a leak, and ultimately break if no corrective actions are taken. A leak could be detected by the previously discussed indications and a plant shutdown could be initiated.

2. If the line does break, the expected failure mode of the transmitters is toward low pressure, thereby allowing the transmitters to fulfill their design function.

3. If the transmitters fail high or as-is, operator awareness of the potential problems associated with a pressurizer spray line break, and proper utilization of the EOPs, will assist in ensuring appropriate actions are taken.

Condition No. 2 - LPSI, HPSI, SI Equipment

Contrary to statements made in the USAR Sections 5.8.4 "Protection of Safety Systems from Missiles", and 7.2.9, "Physical Separation", which indicate that no Reactor Coolant System (RCS) piping is located outside the biological shield, it was discovered that pressurizer loop spray piping from RCS cold leg loops 1B and 2A is routed outside the biological shield. The location of this piping is such that jet impingement and pipe whip due to a line break could damage safety related electrical equipment which must function to mitigate the consequences of this spray line break (small break LOCA).

The piping in question is routed in the vicinity of safety injection tanks SI-6A and SI-6C on the 1013 ft. elevation of containment and in the vicinity of the pressurizer Power Operated Relief (PORVs) and Block Valves, and pressurizer auxiliary spray valves on the 1045 ft. elevation.

A break near SI-6A could cause the loss of two Low Pressure Safety Injection (LPSI) valves, the loss of three High Pressure Safety Injection (HPSI) valves, the diversion of Safety Injection Tank flow from the RCS and failure to initiate Engineered Safety Features (ESF) systems. These failures would be beyond those failures assumed in the USAR for these systems. A break near SI-6C could cause the loss of one LPSI valve, two HPSI valves, and the potential diversion of flow from safety injection tank SI-6C. These failures would be beyond those assumed in the USAR. On the 1045 level, the potential losses include the PORV valves and associated block valves and the pressurizer auxiliary spray valves.

Of the equipment on the 1045 ft. elevation, the pressurizer auxiliary spray valves function to provide hot leg injection after a cold leg break to prevent boron precipitation. Since the break in question would be at the 1045 level, hot leg injection would not be required and would not result in a plant safety concern. If the PORVs were to fail open, the RCS would depressurize through both the break and pressurizer. This would enhance flow through the core and still remain well within safety injection capacity, and thus is not a significant safety concern. Based on this evaluation only breaks on the 1013 ft. elevation will be considered.

An evaluation of the piping on the 1013 ft. elevation using NRC Generic Letter 87-11 "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements" and its related technical document Branch Technical Position (BTP) MEB 3-1 was performed. The results of this evaluation indicated that pipe stresses are such that no pipe breaks need to be postulated and thus pipe whip and jet impingement need not be considered. Further review of the pipe stresses indicated that the stresses may be low enough to conclude that a pipe crack need not be postulated. A pipe stress evaluation for additional piping is presently under further investigation. In the unlikely event that a crack should occur, the RCS leak detection system as described in the USAR Section 4.3.14 and as covered by Technical Specification 2.1.4 of 1 gpm for unidentified leakage is adequate to detect the crack and permit OPPD to begin mitigating actions as required in the Technical Specifications for RCS leakage.

The pipe stress evaluation to determine if cracks must be considered for additional piping will be completed by January 29, 1989. At that time, OPPD will determine the results of the analysis and any further actions that may be required. Please note that additional spray piping below the 1013 ft. elevation is being evaluated.

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

September 30, 1988
LIC-88-879

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 88-005 Revision 1 for the Fort Calhoun Station

Please find attached Licensee Event Report 88-005 Revision 1 dated September 30, 1988. Changes to this LER are marked with a vertical line in the right hand margin. This report is being submitted per requirements of 10 CFR 50.73.

This supplement provides further evaluation of piping routed in the vicinity of safety injection tanks.

If you have any questions, please contact us.

Sincerely,

KJMo

K. J. Morris
Division Manager
Nuclear Operations

KJM/rh

Attachment

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