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

October 22, 1990
LIC-90-0809

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 90-23 for the Fort Calhoun Station

Please find attached Licensee Event Report 90-23 dated October 22, 1990. This report is being submitted pursuant to requirements of 10 CFR 50.73(a)(2)(ii)(B).

If you should have any questions, please contact me.

Sincerely,

W. G. Gates for
Division Manager
Nuclear Operations

WGG/tcm

Attachment

c: R. D. Martin, NRC Regional Administrator
A. Bournia, NRC Project Manager
R. P. Mullikin, NRC Senior Resident Inspector
INPO Records Center
American Nuclear Insurers

IE22
1/1

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

LICENSEE EVENT REPORT (LER)

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)	PAGE (3)
Fort Calhoun Station Unit No. 1	0 5 0 0 0 2 8 5	1 OF 0 5

TITLE (4)

Safety Injection Piping and Relief Valves Outside Design Basis

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 9	2 1	9 0	9 0	–	0 2 3	–	0 0	1 0	2 2	9 0		0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
1	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
1 0 0	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)(ix)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
	AREA CODE
Doug Molzer, Shift Technical Advisor	4 0 2 5 3 3 1 - 6 8 0 4

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)

[ ] 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 September 21, 1990, at 1425 hours CDT, the Plant Review Committee (PRC) determined that, for each of the four trains, the Safety Injection (SI) piping between the SI tank isolation valves and the first SI check valve to the RCS was outside the plant design basis. The setpoint of the relief valve in this section of pipe has been significantly higher than the design pressure of the piping. It was determined that the piping is operable since the most limiting component within the pressure boundary has a pressure rating at a conservative temperature (550 degrees F) of 395 psig, equal to the setpoint of the relief valve. During power operation, the SI tank isolation valves are open, and the SI piping is protected by the relief valves on the SI tanks. Further, it is unlikely the piping integrity would be challenged since the piping could only be subjected to a pressure of 395 psig when the SI tank isolation valves are closed during transition to or from cold shutdown.

This condition has existed since plant construction, and resulted from design and analysis deficiencies by the original plant Architect/Engineer.

Corrective actions include incorporation of safety related relief valves into the Inservice Testing Program, hydrostatic testing of the affected piping, and updating of applicable design basis documents.

The Fort Calhoun Station Updated Safety Analysis Report (USAR) designates the design code for the Safety Injection (SI) System piping as United States of America Standard (USAS) B31.7-1968. The Safety Injection system contains four SI tanks which are designed to provide the core with borated water following a depressurization of the Reactor Coolant System (RCS). The section of piping located between each SI tank isolation valve and the first check valve to the RCS is protected from overpressurization by a relief valve (the piping section and relief valve are similar for each SI tank). Each of these relief valves (SI-278, -279, -280, -281) is designed to relieve the liquid leaking past the SI check valve (SI-207, -211, -215, -219) when the SI tank isolation valve (HCV-2914, -2934, -2954, -2974) is closed and the RCS is pressurized. During power operation, the SI tank isolation valves are open, and the SI piping is protected by the relief valves on the SI tanks.

On September 10, 1990, the NRC Resident Inspector, while reviewing the training manual on the Safety Injection System, noticed the setpoint (1500 psig) for each SI line relief valve was significantly higher then the design pressure for the associated piping. He contacted the Assistant Plant Manager and the System Engineer to determine if the parameters he noted were correct.

On September 14, 1990, the Omaha Public Power District (OPPD) Design Engineering Department was requested by System Engineering to investigate the discrepancy between the relief valve setpoint and the design pressure of the SI piping. Investigation by Design Engineering revealed that the SI piping in question is Class 301 piping with a design pressure and temperature of 250 psig and 200 degrees F. This information was obtained from historical documentation and the original construction code hydrostatic test conducted on October 15, 1972. The piping had been hydrostatically tested to 312 psig based on a design pressure of 250 psig, in accordance with USAS B31.7-68. It was also found that the SI line relief valves have a setpoint of 395 psig, not 1500 psig as noted in the system training manual. The 395 psig setpoint was obtained from the original construction specification sheet for the valves.

After evaluation of design documentation and calculations, Design Engineering determined that the SI piping in question was operable in the as-built configuration, assuming the relief valves opened at their designated setpoint and check valve backleakage did not exceed the relief valve flow capacity. This conclusion was based on the fact that the pressure in the piping would not exceed the rating of the pipe flange, the most limiting component within the pressure boundary. According to USAS B16.5-1968, a 304SS, 300# flange has a pressure rating of 395 psig at 550 degrees F. Calculations also indicated that the piping temperature pressure rating would be 840 psig at 550 degrees F.

On September 21, 1990, at 1425 hours CDT, the Plant Review Committee (PRC) determined that the Safety Injection piping between the SI tank isolation valves and the first SI check valve to the RCS was outside construction code requirements and thus outside the plant design basis. A one-hour notification was made to the NRC on September 21, 1990 at 1521 hours CDT pursuant to 10 CFR 50.72(b)(1)(ii)(B). This written report is submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B).

Fort Calhoun Technical Specification 2.3(1)i requires that the SI tank isolation valves be open and power removed from the motor operators, prior to the reactor being taken critical. During power operation, the SI tank isolation valves are open, and the SI piping is protected by the relief valves on the SI tanks. When the SI tank discharge valves are open the maximum pressure this section of piping would be subjected to is limited by the 275 psig setpoint of the SI tank relief valves. Since system integrity has been verified to 312 psig by the original hydrostatic test, no degradation of the piping would result by the pressurization of the piping to the 275 psig setpoint. Therefore, overpressurization of this piping is only a concern when the SI tank isolation valves are closed, such as during transition to cold shutdown or startup from cold shutdown.

Apparently the 395 psig relief valve setpoint was based on the most limiting rating for the class 301 piping system rather than on the piping design pressure as expected. Flange ratings are typically the most limiting on a piping system. Using the flange rating table from USAS B16.5-1968, since this was the governing standard for original equipment at Fort Calhoun, a 304 SS, 300# flange has a pressure rating of 395 psig at a design temperature of 550 degrees F. The data sheet from the original valve specification requires a design temperature of 547 degrees F., which is the maximum assumed temperature for the piping if both check valves from the RCS were to leak.

It was discovered that the design pressure for the SI piping in question should have been increased to 275 psig from 250 psig during plant construction. In the late 1960's, questions were raised by the Atomic Energy Commission (AEC) concerning the effectiveness of the Emergency Core Cooling System on all similar plants. It was decided that the operating and design pressure in the SI tanks needed to be raised to ensure that maximum cladding temperature would not be exceeded during a RCS depressurization event. As a result of this concern, design pressure in the SI tanks was increased from 250 psig to 275 psig. At the time, the design pressure of the SI piping in question should have also been increased to 275 psig as the piping is subjected to tank pressure when the SI tank isolation valves are open. The reason the design pressure for the SI piping was not revised to reflect the change in design pressure for the SI tanks is indeterminate.

This reportable condition is attributed to design and analysis deficiencies by the original plant Architect/Engineer (A/E). Investigation determined that this condition has existed since plant construction.

Safety Analysis for Operability (SAO) 90-10 was issued on October 3, 1990 to justify continued plant operation. It was determined that the piping is operable since the most limiting component within the pressure boundary has a pressure rating at a conservative temperature (550 degrees) of 395 psig, equal to the setpoint of the relief valve. Further, it is unlikely the piping integrity would be challenged since the piping and relief valve could be subjected to a pressure of 395 psig only when the SI tank isolation valves are closed during transition to or from cold shutdown.

The one-hour notification made to the NRC on September 21, 1990 indicated that the shutdown procedures would be revised to ensure RCS pressure is reduced below 312 psig before the SI tank isolation valves are closed. Subsequent engineering evaluation has determined that changing the shutdown procedures is unnecessary. Calculations show the piping involved would not be overstressed if subjected to pressure equal to the relief valve setpoint of 395 psig.

The following corrective actions have been taken:

(1) A walkdown of the affected piping was performed with no visible deformation or damage to the piping or hangers noted.

(2) The SI System Training Manual has been revised to reflect the actual relief valve setpoint.

(3) The piping stress analysis was reviewed for the increased pressure and found to be acceptable.

The following corrective actions will be implemented:

(1) A hydrostatic pressure test will be performed on the applicable sections of piping during the 1991 refueling outage to establish and confirm the design pressure of 395 psig. Applicable design basis documents will be updated.

(2) The ISI Program has been upgraded to include testing of safety related relief valves. These valves will be tested on a frequency of at least once every five years in accordance with Relief Valve Surveillance Test Procedure PE-ST-VX-3001. During the verification and validation of this procedure, the relief valve setpoints will be compared to existing design basis documentation in order to ensure that the setpoints are consistent with the design basis documents. This verification and validation of PE-ST-VX-3001 is expected to be completed by March 31, 1991.

(3) The system engineers will be briefed on events noted in this report to heighten their sensitivity to other similar discrepancies which may exist. This action will be completed by November 30, 1990.

LER's 90-03, 90-07 90-09, 90-16, and 90-20 also describe conditions outside the design basis of the plant due to design deficiencies by the original A/E.