OPPD
Omaha Public Power District
444 South 16th Street Mall
Omaha NE 68102-2247

February 21, 1997
LIC-97-0022

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Subject: Licensee Event Report 97-001 Revision 0 for the Fort Calhoun Station

Please find attached Licensee Event Report 97-001 Revision 0 dated February 21, 1997. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B). If you should have any questions, please contact me.

Sincerely,

S. K. Gambhir

S. K. Gambhir
Division Manager
Engineering & Operations Support

EPM/epm

Attachment

IE22 1/1

c: Winston and Strawn
L. J. Callan, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
INPO Records Center

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Station Unit No. 1	05000285	1 OF 5

TITLE (4): Main Steam Outside of Design Basis Due to an Error in Safety Valve Analysis

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 NAME	DOCKET NUMBER
01	22	97	97	001	00	02	21	97		05000
										05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Randel E. Lewis, Principal Engineer	(402)533-6508

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 15 single-spaced typewritten lines) (16)

Following the report by Millstone Unit 2 of a problem with the calculation of Main Steam Safety Valve (MSSV) setpoints on September 3, 1996, Fort Calhoun Station (FCS) began an investigation to determine if similar issues applied. It has been determined that on only one occasion, over the life of the plant, have the number of inoperable MSSVs exceeded the design basis as recalculated due to this incident. The design basis was recalculated when it was discovered that the pressure drop to the MSSVs was not being taken into account.

The root cause of this event was an inadequate vendor review of a CESEC code modeling. When the code modeling was originally developed, the analysis methodology should have accounted for piping pressure losses associated with flow. CESEC does not have the capability for directly modeling pressure drop in the piping. Therefore, the potential existed for pressure to exceed the code allowed during a single Main Steam (MS) isolation valve closure event.

OPPD has performed the necessary analysis to update the design basis of the plant to account for the error discovered and reported in this LER. Guidance has been provided to the operating staff to ensure that the design basis is maintained. A revision to the FCS Technical Specifications will be submitted to appropriately reflect the new design basis.

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BACKGROUND

The Fort Calhoun Station (FCS) uses two steam generators (SGs) to supply steam to a single turbine. The Main Steam (MS) system is protected from over-pressure by ten safety valves, five on each of the two MS headers. The MS Safety Valves (MSSVs) are located in Room 81 just outside of the containment, about 100 piping feet from the SGs.

EVENT DESCRIPTION

On September 3, 1996, Millstone Unit 2 reported that the pressure drop between their SGs and MSSVs had not been taken into account in determining the safety valve setpoints. They indicated that the SG pressure resulting from one of the Design Basis Accidents (DBAs) would exceed 110 percent of American Society of Mechanical Engineers (ASME) design pressure for the system. Omission of the pressure drop effects the loss of load, loss of feedwater and loss of coolant accident (LOCA) analyses, and creates a potential for MSSV chatter. Omaha Public Power District (OPPD) management determined that the issue should be reviewed for applicability to FCS. An Engineering Assistance Request (EAR) 96-153 was initiated to investigate this issue. The EAR was assigned to Design Engineering to determine if similar issues existed at FCS.

Calculation FC 06627 was performed to determine the pressure drop between the SGs and MSSVs for conditions of maximum design flow through the MSSVs coincident with MS Isolation Valve (MSIV) closure without a reactor trip. The calculated pressure drop is 35 pounds per square inch (psi) at maximum steam flow.

OPPD conducted a review to determine if problems similar to those reported by Millstone and subsequently elaborated upon by Asea Brown Boveri/Combustion Engineering (ABB/CE), formerly known as Combustion Engineering (CE) existed at FCS. OPPD reviewed the FCS and Millstone Unit 2 MSSV configuration and their implications on FCS MSSV operation. A review of FCS MSSV design basis, disclosed one calculation related to valve setpoints. Calculation FC 05586 Rev. 0, determined the minimum and maximum MSSV setpoints allowed by code, without consideration of inlet pressure drop. When the calculated pressure drop was considered in the response to EAR 96-153, it was concluded that the FCS MSSV configuration is not susceptible to the valve chattering reported by Millstone.

While updating the appropriate plant calculations with the previously mentioned data, the engineer compared his data with the pressure drop assumed in the plant accident analyses and discovered that the pressure drop between the SGs and MSSVs was also not accounted for in those analyses. Omission of the pressure drop, as previously noted, could adversely affect certain LOCA and non-LOCA safety analyses where the MSSVs are

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activated.

On January 14, 1997, engineering completed the calculation associated with the EAR 96-153 and concluded that pressure drops occurring between the SG and the inlet to the MSSVs was not accounted for in the appropriate Updated Safety Analysis Report (USAR) Chapter 14 (Safety Analysis) analyses. On January 15, 1997, a Condition Report (CR) was issued to document above findings and initiate the necessary corrective actions to resolve the deficiency.

CE had performed the initial LOCA and non-LOCA thermal hydraulic safety analyses for FCS. The non-LOCA analyses consist of the loss of load and loss of feedwater DBAs. The loss of load analysis was performed by CE until 1983 when OPPD started performing the analysis in-house. The CESEC (Combustion Engineering System Excursion Code) code is utilized to perform this analysis. The CESEC code does not have the capability to directly account for pressure drops in piping upstream or downstream of the MSSV. The piping losses can be accounted for within the CESEC code by adjusting other parameters, such as valve opening area and valve setpoints.

On January 17, 1997, ABB-CE issued a 10 CFR 21 notification that a reportable defect exists which is applicable to all safety analyses for nuclear power plants for which ABB-CE was the nuclear steam supply system vendor. The specific concern involves the piping losses between the SG and the MSSVs while the MSSVs are open. The CE part 21 notification stated that omission of the piping loss from calculations may adversely affect certain LOCA and non-LOCA safety analyses where the MSSVs are activated. The ASME code requirement states, in part, that "..in determining the setting pressures and discharge capacities required to comply with these rules, full account shall be taken of the pressure drop in both inlet and discharge side of the pressure relief devices at full discharge conditions. In addition back pressure arising from discharge of other devices through common discharge piping shall be considered..." This is the same issue that was independently discovered by OPPD and documented by the CR previously mentioned.

On January 22, 1997, this information was brought to the Plant Review Committee (PRC) for a reportability review. At 0850 Central Standard Time (CST), the PRC determined that this condition constituted a condition where the plant may have been outside of the plant design basis in the past. A one-hour non-emergency report was made to the NRC Operations Center pursuant to 10 CFR 50.72(b)(1)(ii)(B) on January 22, 1997, at 1005 Eastern Standard Time (EST). This report is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B).

Technical Specification (TS) 2.1.6(3) states that "Whenever the reactor is in power operation, eight of the ten main steam safety valves shall be operable..." Subsequent

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analysis has determined that the design basis for the plant will support one inoperable MSSV on each of the two MS headers.

A comprehensive review was conducted to determine the status of all the MSSVs over the life of the plant. It was determined that on only one occasion over the life of the plant had more than one MSSV exceeded 103 percent of set pressure on the same MS header. LER 88-023 Revision 1 reported a condition where three MSSVs failed to lift within their required lift range during testing. MS-277 and MS-278 are connected to the same MS header. The revised design basis, which considers pressure drop, will not support any two steam safety valves being inoperable on the same MS header. The 1988 as found data for the valves was:

Valve	Specified Set Pressure (psig)	As Found Lift Pressure (psig)
MS-275	1035	1045
MS-276	1025	1030
MS-277	1010	1140
MS-278	1000	1135
MS-279	1035	</= 1025
MS-280	1025	1020
MS-281	1010	1010
MS-282	1000	1125
MS-291	985	985
MS-292	985	990

Where psig is pounds per square inch gage. LER 88-023 Revision 1 reported that an analysis performed using the CE CESEC code indicated that even though three valves had been inoperable, the design basis of the plant was not effected. The analysis performed in 1988 did not account for the pressure drop discovered and reported in this LER. The conclusion of LER 88-023 Revision 1 that the design basis was not affected was incorrect. An evaluation using current data has been conducted. The results of this analysis are discussed below.

SAFETY SIGNIFICANCE

As indicated above, a comprehensive review of the plant history of MSSV operability has been conducted. While the plant TSs would have allowed two MSSVs on a single MS line to be inoperable, at only one time in the history of the plant has more than one valve in each MS line exceeded 103 percent of set pressure.

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The limiting DBA, loss of feedwater, was evaluated with the MSSVs lifting at the as found values indicated by the 1988 test results. This evaluation indicated that the maximum pressure that would have been seen in a design basis loss of feedwater accident would be about 1099 psia, which is less than 110 percent of the design MS system pressure. The SGs have been hydrostatically tested to 125 percent of design pressure without incident as required by ASME code. No incidents of MS pressure unintentionally exceeding the design limit have been recorded.

These evaluations demonstrate that the error discovered and reported by this LER does not pose a significant safety concern.

CONCLUSION

The root cause of this event was an inadequate vendor review of CESEC code modeling. When the original code modeling was developed, the analysis methodology should have accounted for piping pressure losses associated with flow. CESEC does not have the capability for directly modeling pressure drop in the piping. Therefore, the potential existed for allowing pressure to exceed the ASME code limit during a single MSIV closure event.

A contributing cause to this event is that CE performed transient analyses several times, but did not verify the input data regarding adjustment for piping loss. OPPD Engineering has performed loss of load and loss of feedwater analyses using the CESEC code since 1983 and the oversight was carried over from one revision to next revision.

CORRECTIVE ACTIONS

OPPD has performed the necessary analysis to update the design basis of the plant to account for the error discovered and reported in this LER. Guidance has been provided to the operating staff to ensure that the design basis is maintained.

OPPD will submit a revision to the FCS TSs to stipulate that no more than one MSSV may be inoperable on a MS line whenever the reactor is in power operation. The revision to TSs will be submitted by March 31, 1997.

PREVIOUS SIMILAR EVENTS

Other than the incident referenced above (LER 88-023), no other incidents have occurred at the FCS where MSSV setpoints have been affected by errors in vendor provided calculations.