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

APPROVED BY OMB NO. 3150-0104

EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	1 OF 3

TITLE (4)

Non-conservative Assumptions Identified in Analysis for Peak Secondary System 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 NAME	DOCKET NUMBER
10	03	96	96	-- 031 --	01	04	03	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5	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)
POWER LEVEL (10)	000		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 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)
R. G. Joshi, MP2 Nuclear Licensing	(860) 440-2080

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)

On October 3, 1996, it was discovered that potentially non-conservative assumptions used in an analysis could result in exceeding the ASME Code maximum relief valve accumulation for the steam generators (SG) and main steam line piping during an analyzed design basis event. During a review of the analysis for the single main steam isolation valve (MSIV) closure event, it was discovered that potentially non-conservative assumptions were made in the modeling of the main steam line, main steam safety valves (MSSV), and SGs. A preliminary analysis of these conditions has been performed. The results indicate that the 110 percent of design rating of the SGs would be exceeded for the single MSIV closure and loss of load design basis events.

The cause of this event was an inadequate design of the MSSV inlet piping and the failure to adequately assess the piping pressure losses to the MSSVs.

As a result of this event, corrective actions will be implemented to ensure that the plant response to analyzed events will not result in exceeding the design requirements of the SG. These actions will include any necessary reanalysis and plant modifications. These actions will be completed prior to plant restart from the current outage.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	96	031	01	2 OF 3

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. Description of Event

On October 3, 1996, it was discovered that potentially non-conservative assumptions used in an analysis could result in exceeding the ASME Code maximum relief valve accumulation for the steam generators (SG) and main steam line piping during an analyzed design basis event. At the time of discovery of this event, the unit was in Mode 5 at 0 percent power.

During a review of the analysis for the single main steam isolation valve (MSIV) closure event, it was discovered that potentially non-conservative assumptions were made in the modeling of the main steam line, main steam safety valves (MSSV), and SGs. It has been determined that the piping pressure losses between the SG and the MSSV inlets were not fully evaluated with respect to the MSSV performance and not fully addressed within plant safety analyses involving MSSV actuation.

Additionally in reviewing this event, it was identified that the relieving capacity of the MSSVs was significantly reduced due to the design of the connective piping between the main steam line and the MSSVs. The 6 inch connective piping is XXS grade piping with an inside diameter of less than 5 inches. The original analysis also did not identify peak steam generator pressure at the limiting location, resulting in a pressure slightly lower than the actual peak steam generator pressure. However, this error does not significantly affect the analysis.

A preliminary analysis of these conditions has been performed. The results indicate that the 110 percent of design rating of the SGs would be exceeded for the single MSIV closure and loss of load design basis events.

II. Cause of Event

The cause of this event was an inadequate design of the MSSV inlet piping and the failure to adequately assess the piping pressure losses to the MSSVs. The MSSVs were not capable of relieving their rated capacity at an acceptable pressure due to the smaller diameter installed connective piping. Additionally, subsequent analysis failed to account for the effects of the inadequate design in the analysis of peak pressures for the design basis events.

The original design basis for Unit 2 did not require analysis for the single MSIV closure event. Analysis of this event was included at a later time and was originally performed by Westinghouse Electric Corporation. The identified potential non-conservative assumptions are believed to have existed in the original analysis also. The current analysis was performed by Siemens Power Corporation.

III. Analysis of Event

The Single MSIV Closure Event is the limiting event for secondary side pressure. The closure of a single MSIV during operation will decrease the heat removal by the secondary system. Upon cessation of steam flow to the turbine, the pressure in the affected steam generator will increase above the opening setpoint of the MSSVs. The peak analyzed SG dome pressure for this event is 1096 psia.

Due to the non-conservative assumptions identified in the analysis, the peak secondary side pressure would be greater than previously calculated. Based on the preliminary analysis, the peak secondary side pressure would exceed the design pressure rating of the SGs and main steam line piping. This would also be true for the Loss of Load Event. Therefore, this event is considered to be safety significant.

The effect of this condition on a small break loss of coolant accident (LOCA) was evaluated since the SG pressure is controlled by the MSSVs during the initial phases of the LOCA. The effect of the inlet losses on SG

pressure is not significant if all the MSSV banks have not opened. Since the net loss of capacity caused by the inlet losses is less then the capacity of 1 MSSV, the maximum increase in SG pressure would equal the maximum difference in MSSV opening setpoint, or 10 psi. This magnitude change in SG pressure was evaluated by Siemens Power Corporation to be insignificant to peak fuel cladding temperature.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(ii)(B), any event or condition that resulted in the condition of the nuclear power plant, including the principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant. This event was reported in accordance with 10 CFR 50.72(b)(1)(ii) on October 3, 1996.

IV. Corrective Action

As a result of this event, corrective actions will be implemented to ensure that the plant response to analyzed events will not result in exceeding the design requirements of the SG. These actions will include any necessary reanalysis and plant modifications. These actions will be completed prior to plant restart from the current outage.

V. Additional Information

None

Similar Events

LER 91-010: On October 18, 1991, a reportability determination was made concerning a reanalysis of the main steam line break event inside the containment. The reanalysis confirmed that the assumptions made for the existing (1979) main steam line break (MSLB) analysis were non-conservative with respect to power level, break size, and single active failure. Using more restrictive assumptions, design limits for containment pressure and temperature could be exceeded. A multi-disciplinary task force was established to investigate containment response to postulated MSLBs. Plant modifications required to ensure an acceptable containment pressure response for a main steam line break inside the containment were installed and tested.

Manufacturer Data

None

EIIS Codes

SG - Steam Generator
RV - Main Steam Safety Valve
SB - Main Steam System
ISV - Main Steam Isolation Valve