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 3	05000423	1 of 6

TITLE (4)
Non-Conservatism in the Low Temperature Overpressure Protection and Pressure/Temperature Limit Curves in the Technical Specifications

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
05	01	97	97	-- 030 --	00	05	30	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 5
POWER LEVEL (10): 000

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)
David A. Smith, MP3 Nuclear Licensing Manager	(860)437-5840

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 May 1, 1997, at 1030 hours, with the unit in Mode 5, it was identified that the current Reactor Coolant System heatup and cooldown pressure and temperature (P/T) limit curves and the Power-Operated Relief Valve (PORV) setpoint curves for the Cold Overpressure [Protection] System (COPS) contained in Technical Specifications (TS) 3.4.9.1, "Pressure/Temperature Limits," and 3.4.9.3, "Overpressure Protection Systems," respectively, were non-conservative. From initial startup until May 9, 1997, the P/T limits and the PORVs setpoint (COPS) curves contained within the TS did not properly account for instrumentation and system configuration uncertainties. Since 1993, interim administrative controls were imposed within operating procedures to account for the pressure drop across the core with the RCPs in operation. However, the PORV setpoint curves were not revised to account for this effect or the other uncertainties recently identified and therefore, PORV operation may not have occurred in the past when required. This condition is being reported pursuant to 10 CFR 50.73(a)(2)(ii)(B) as a condition outside the unit design basis.

Inadequate design verification and validation was performed by the Nuclear Steam Supply System vendor and the utility.

Administrative precautions on system operation minimize the likelihood and severity of COPS events.

A review of system operational aspects will be performed to determine if the 10 CFR 50 Appendix G limits have been met since startup. Based on preliminary information Appendix G limits are considered to have been met.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 3	05000423	97	-- 030 --	00	2 of 6

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

I. Description of Event

On May 1, 1997, at 1030 hours, with the unit in Mode 5, it was identified that the current Reactor Coolant System (RCS) heatup and cooldown pressure and temperature (P/T) limit curves and the Power-Operated Relief Valve setpoint curves for the Cold Overpressure [Protection] System (COPS) contained in Technical Specifications (TS) 3.4.9.1, "Pressure/ Temperature Limits," and 3.4.9.3, "Overpressure Protection Systems," respectively, were non-conservative. The non-conservatism in the P/T limit and Power-Operated Relief Valve (PORV) setpoint (or COPS) curves was due to incorrect accounting of the instrument uncertainties, instrument loop uncertainties, and system hydraulic considerations. The major uncertainties were:

- PORV instrumentation uncertainties: The current COPS curves provided by the Nuclear Steam Supply System (NSSS) vendor did not include any pressure or temperature uncertainty. When detailed system specific uncertainties were calculated, the temperature uncertainty increased by approximately 15 degrees Fahrenheit and the pressure uncertainty increased by approximately 85 psi.

- Pressure drop across the vessel with the Reactor Coolant Pump(s) (RCPs) in operation: Nuclear Regulatory Commission (NRC) Information Notice (IN) 93-58, "Non-Conservatism in Low Temperature Overpressure Protection [LTOP] for Pressurized-Water Reactors," discusses the pressure drop across the core with the RCPs in operation. The 10 CFR 50 Appendix G, "Fracture Toughness Requirements," pressure and temperature limiting area is the vessel downcomer region, while the pressure is measured in the RCS Hot Leg. The present TS curves do not include additional margin to account for the pressure difference between the instrument and the location of concern. In 1993 interim administrative controls were imposed to account for the pressure drop across the core with the RCPs in operation. Recently performed calculations resulted in new TS proposed curves which account for this effect.

- Difference in pressure between the relief (PORV or RHR suction relief) valves location and location of the pressure measuring instrument: Recently performed calculations resulted in new TS proposed curves which also account for this effect.

TS Limiting Condition for Operation (LCO) 3.4.9.3 requires that at least one of three combinations of relief valves (either two PORVs, two RHR suction relief valves, or one PORV and one RHR suction relief valve) be operable, or that the RCS be depressurized with a vent area of 5.4 inches or greater. On September 18, 1996, as discussed within Licensee Event Report 96-034-00, it was identified that the 440 psig setpoint for the RHR suction relief valves, was not in compliance with the setpoint of 450 psig specified in LCO 3.4.9.3. The RHR suction relief valve setpoint of 440 psig is the correct setpoint and is conservative. The RHR suction relief valves were declared inoperable and have remained inoperable pending receipt of a Technical Specification setpoint change to revise their relief setpoint to 440 psig. The RHR suction relief valves while declared inoperable, were still available, and could perform their relief function if an LTOP event were to occur.

On May 1, 1997, at 1150 hours, it was identified that the PORV setpoint curves were nonconservative. Both trains of COPS were declared inoperable. With none of the three relief valve combinations of LCO 3.4.9.3 operable, Action c was entered. Action c requires "with both of the required relief valves inoperable, complete depressurization and venting the RCS through at least a 5.4 square inch vent within 8 hours." At 1716 hours on May 1, 1997, a pressurizer safety valve was removed establishing the vent path and LCO Action c was exited because the overpressure protection system was restored to compliance with the LCO.

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

From initial startup until May 1, 1997, the pressure and temperature limits and the PORVs setpoint (COPS) curves contained within the TSs did not properly account for instrumentation and system configuration uncertainties. Since 1993 interim administrative controls were imposed within operating procedures to account for the pressure drop across the core with the RCPs in operation. However, the PORV setpoint (COPS) and the heatup and cooldown curves were not revised to account for this effect or the other uncertainties recently identified, discussed above, and therefore, the unit was operated without adequate protection against anticipated operational occurrences. This condition is being reported pursuant to 10 CFR 50.73(a)(2)(ii)(B) as a condition outside the unit design basis.

II. Cause of Event

Inadequate design verification and validation was performed by the Nuclear Steam Supply System (NSSS) vendor and the utility. Design inputs that should have been used to generate the original 10 CFR 50 Appendix G curves were not included by the vendor and not identified in the review process by the NSSS vendor or by the utility.

III. Analysis of Event

The components of the RCS are designed to withstand the effects of cyclic loads due to system pressure and temperature and rate of change considerations. Above 350 degrees Fahrenheit, cold overpressurization is not a concern because the reactor vessel is not susceptible to non-ductile failure. NRC Branch Technical Position RSB 5-2, requires that a Cold Overpressure Protection System be operable during low temperature reactor operation (below 350 degrees Fahrenheit for this unit) to ensure that the 10 CFR 50, Appendix G requirements are not inadvertently exceeded as a result of anticipated operational occurrences. The PORVs and the RHR suction relief valves are credited in the safety analyses as part of the COPS to control the RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature limits of 10 CFR 50, Appendix G.

The additional considerations included in the recently determined new P/T limits and PORV setpoint curves require PORV actuation at lower pressures and temperatures than previously prescribed and hence PORV operation may not have protected against a COPs event occurring in certain P/T regions. This inoperability of the PORVs is significant because it represents a condition outside the design basis of the plant. However, based on preliminary information it is expected that the 10 CFR 50 Appendix G limits have not been exceeded.

The period of time where the susceptibility to overpressure events occurs is generally of short duration and administrative precautions, i.e., restrictions on RCP operation, restrictions on cooldown and heatup rates, and strict restrictions on injection and makeup system operations are imposed in operating procedures to minimize both the likelihood and magnitude of a COPs event. Therefore, the likelihood of a COPS event and its severity was minimized by the administrative controls in place.

IV. Corrective Action

The following corrective actions were taken:

1. On May 1, 1997, a pressurizer safety valve was removed to establish a vent path that did not rely on operation of the PORVs or the RHR suction relief valves.

2. Composite cold overpressure protection curves were developed using the most conservative values from the present Technical Specification PORV setpoints and the recently calculated proposed Technical Specification PORV setpoints. These curves were approved by The Plant Operations Review Committee and added to Section 3.4.9.3 of the Technical Requirements Manual, effective May 9, 1997, to provide interim administrative controls.

The following corrective actions will be taken:

1. A review of system operational aspects will be performed to determine if the 10 CFR 50 Appendix G limits have been met since startup. Review and evaluation will be completed prior to entering Mode 4. If Appendix G limits are found to have been violated a supplement to this LER will be submitted.

2. A Technical Specification change will be submitted to revise the existing pressure and temperature limits and the PORV setpoint curves by July 1, 1997.

V. Additional Information

None

Similar Events

LER 96-034-00 "Residual Heat Removal (RHR) Pump Suction Relief Valve Setpoint Not In Accordance With TS"

On September 18, 1996 with the plant in Mode 5, during a review conducted by plant engineering personnel, it was determined that the actual setpoint for the RHR pump suction relief valves was not in accordance with the requirements of Technical Specification (TS) 3.4.9.3. The TS require that the RHR pump suction relief be set at 450 psig in order to provide adequate over pressure protection when the temperature of any RCS cold leg is less then 350 degrees Fahrenheit. Contrary to this requirement, the actual lift pressure for the RHR pump suction relief had been set at 440 psig. This condition was reported pursuant to 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plant's TS.

LER 97-007-00 "Non-Conservative Assumptions Used in Technical Specification Shutdown Margin Curve"

On January 14, 1997, as the result of an engineering review, it was postulated that an incorrect assumption had been made in determining the degree of error in a Cycle 6 Shutdown Margin (SDM) Curve which had previously been found to be in error. Subsequent Engineering evaluation concluded that the Cycle 6 SDM - Mode 3 curve had been non-conservative. On January 23, 1997, with the unit in Mode 5, a prompt event report was made to the Nuclear Regulatory Commission of this historical

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 3	05000423	97	-- 030 --	00	5 of 6

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

condition pursuant to 10CFR50.72(b)(2)(iii)(A) as an event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor and maintain it in a safe shutdown condition.

LER 97-015-00 "Potential Vortexing of Recirculation Spray System Pumps"

On February 4, 1997, with the unit in Mode 5, a review of design calculations identified that the calculated minimum water level in the containment sump at the time of Recirculation Spray System (RSS) pumps start (approximately eleven minutes after a Containment Depressurization Actuation signal) would be below the level of the containment sump vortex suppression gratings during a large break Loss of Coolant Accident (LOCA). Vortex formation and the resulting air entrainment, could result in cavitation of the RSS pumps, such that they might be unable to perform their intended safety function(s) in an accident. Following evaluation, this condition was immediately reported on February 7, 1997, pursuant to 10CFR50.72(b)(1)(ii)(B), and is being reported pursuant to 10CFR50.73(a)(2)(ii)(B) as a condition outside the design basis of the plant.

LER 97-028-00 "RSS Suction Line Loss Greater Than Available NPSH"

On April 10, 1997, with the unit in Mode 5, a review of design calculations for Net Positive Suction Head (NPSH) identified that a potential for flashing existed in the suction lines to the Recirculation Spray System (RSS) pumps (3RSS*P1A,B,C&D). Calculations for the RSS System had originally determined that sufficient NPSH existed at the centerline of the first stage of the RSS pump impellers utilizing a saturated sump model. However, these calculations contained an inherent assumption that the fluid would remain single phase. This assumption is inappropriate when utilizing the saturated sump model. Flashing, resulting in insufficient NPSH, would result in cavitation of the RSS pumps, such that they might be unable to perform their intended safety functions during a design basis event (DBE). In this condition the RSS system would not be capable of supplying the minimum volume of cooling water credited in the containment analysis. Following evaluation, this condition was immediately reported on April 16, 1997, pursuant to 10CFR50.72(b)(2)(iii)(D) as a loss of safety function. It is being reported pursuant to 10CFR50.73(a)(2)(v)(D) and 10CFR50.73(a)(2)(ii)(B), as a loss of safety function and as a condition outside the unit design basis.

LER 97-029-00 "Design Basis Concern on SGTR Analysis for MSPRBV"

On March 17, 1997, with the unit in Mode 5, it was determined that between February 8, 1996 and March 18, 1996, the minimum number of Main Steam Pressure Relief Bypass Valve (MSPRBV) flowpaths required to depressurize the Reactor Coolant System (RCS) to satisfy the steam generator margin to overfill analysis may not have been available in the event of a Steam Generator Tube Rupture (SGTR) concurrent with a single failure. Further investigation confirmed on April 17, 1997 that the minimum number of MSPRBV flowpaths required after a SGTR combined with a single failure, could not be met. This condition existed for a period of approximately one month. This event is being reported pursuant to 10CFR50.73(a)(2)(ii)(B) as a condition that resulted in the plant being outside the design basis. The cause of this event was a failure to recognize design and single failure requirements necessitating that the MSPRBV be included in the Technical Specifications.

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

Manufacturer Data

EIIS System Code

Residual Heat Removal System (PWR)........................BP
Reactor Coolant System...AB

EIIS Component Code

Valve, Relief..RV