



**Northeast  
Utilities System**

Millstone Offices • Rope Ferry Rd., Waterford, CT

P.O. Box 128  
Waterford, CT 06385-0128  
(203) 447-1791

October 18, 1996

Docket No. 50-423

B15939

Re: 10CFR 50.73(a)(2)(i)(B)

U. S. Nuclear Regulatory Commission  
Document Control Desk  
Washington, DC 20555

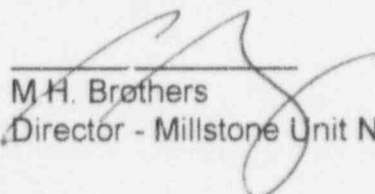
This letter forwards Licensee Event Report 96-034-00, documenting a condition that was determined at Millstone Unit No. 3 on September 18, 1996. This LER is submitted pursuant to 10CFR 50.73(a)(2)(i)(B).

The following are NNECO's commitments made within this letter:

B15939-01: A Technical Specification Change Request to revise the Residual Heat Removal pump suction relief valve setpoint from 450 psig to 440 psig will be submitted to the Nuclear Regulatory Commission for approval.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

  
M.H. Brothers  
Director - Millstone Unit No. 3

Attachment: LER 96-034-00

cc: H. J. Miller, Region I Administrator  
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3  
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3

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PDR ADOCK 05000423  
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## LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)

Millstone Nuclear Power Station Unit 3

DOCKET NUMBER (2)

05000423

PAGE (3)

1 of 3

TITLE (4)

Residual Heat Removal Pump Suction Relief Valve Setpoint Not in Accordance With Technical Specifications

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
09	18	96	96	034	00	10	18	96	FACILITY NAME	DOCKET NUMBER
OPERATING MODE (9)		5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more) (11)							
POWER LEVEL (10)		000	20.2201(b)		20.2203(a)(2)(v)		<input checked="" type="checkbox"/> 50.73(a)(2)(i)		50.73(a)(2)(viii)	
			20.2203(a)(1)		20.2203(a)(3)(ii)		<input type="checkbox"/> 50.73(a)(2)(ii)		50.73(a)(2)(x)	
			20.2203(a)(2)(i)		20.2203(a)(3)(iii)		<input type="checkbox"/> 50.73(a)(2)(iii)		73.71	
			20.2203(a)(2)(ii)		20.2203(a)(4)		<input type="checkbox"/> 50.73(a)(2)(iv)		OTHER	
			20.2203(a)(2)(iii)		50.36(c)(1)		<input type="checkbox"/> 50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A	
			20.2203(a)(2)(iv)		50.36(c)(2)		<input type="checkbox"/> 50.73(a)(2)(vii)			

## LICENSEE CONTACT FOR THIS LER (12)

NAME

R. T. Laudenat, Nuclear Licensing Supervisor

TELEPHONE NUMBER (include Area Code)

(860)444-5248

## 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)

<input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE).	<input checked="" type="checkbox"/> NO
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## EXPECTED SUBMISSION

MONTH DAY YEAR

## ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

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 Residual Heat Removal (RHR) pump suction relief valves was not in accordance with the requirements of Technical Specification 3.4.9.3.a.2 and 3.4.9.3.a.3. Technical Specifications 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 Reactor Coolant System (RCS) cold leg is less than 350 degrees Fahrenheit (°F). Contrary to this requirement, the actual lift pressure for the RHR pump suction relief had been set at 440 psig.

This condition is being reported pursuant to 10CFR50.73(a)(2)(i)(B) as an operating condition prohibited by the plant's Technical Specifications. This event has no adverse safety consequence in that the appropriate lift pressure for this valve, as documented in both the original pre-startup design change and the subsequent setpoint calculation, is 440 psig. All subsequent relief valve capacity calculations were based on the 440 psig setpoint. Over pressure protection of RHR piping and Low Temperature Over pressure/Cold Over Pressure Protection System (LTOP/COPPS) requirements also have utilized the 440 psig setpoint for all calculations performed following the original pre-startup design change. At no time were the relief valves incapable of performing their intended safety functions of providing RHR system over pressure protection or LTOP/COPPS protection.

A Technical Specifications Change Request is being processed to revise the value identified in Technical Specifications 3.4.9.3.a.2 and 3.4.9.3.a.3 from the current value of 450 psig setpoint to correct setpoint value of 440 psig.

LICENSEE EVENT REPORT (LER)  
TEXT CONTINUATION

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		YEAR	SEQUENTIAL NUMBER		REVISION NUMBER	
		96	--	034	-- 00	

Millstone Nuclear Power Station Unit 3

05000423

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

I. Description of Event

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 Residual Heat Removal (RHR) pump suction relief valves was not in accordance with the requirements of Technical Specification 3.4.9.3.a.2 and 3.4.9.3.a.3. Technical Specifications 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 Reactor Coolant System (RCS) cold leg is less than 350 degrees Fahrenheit (°F). Contrary to this requirement the actual lift pressure for the RHR pump suction relief had been set at 440 psig.

This condition is being reported pursuant to 10CFR50.73(a)(2)(i)(B) as an operation prohibited by the plant's Technical Specifications. The event had no adverse safety consequence in that the appropriate lift pressure for this valve, as documented in both the original design change and the subsequent setpoint calculation, is 440 psig. The testing on this valve has historically been performed successfully at the correct pressure of 440 psig.

II. Cause of Event

On March 26, 1985 during the construction of the unit, the Architect/Engineer (A/E) issued a design change, which lowered the setpoint for the RHR suction relief valve from 450 psig to 440 psig. This design change did not specify that a technical specification change was required. The design change also resulted in the issuance of a setpoint calculation change on April 29, 1985 which confirmed the change from 450 psig to 440 psig. Subsequently, the plant Maintenance Department established the relief valve test program in accordance with the requirements imposed by American Society Of Mechanical Engineers (ASME). In implementing the relief valve test program, reliance was placed on the information contained in setpoint calculations and design changes when establishing the appropriate values for testing of valves contained within the program.

III. Analysis of Event

The unit has continuously operated using the 440 psig setpoint as required by the original A/E design change and the associated the setpoint calculations. The 440 psig setpoint is the technically correct setpoint for the RHR suction relief valves to ensure over pressure protection for the RHR piping in accordance with ASME requirements. All subsequent relief valve capacity calculations were based on the 440 psig setpoint. Over pressure protection of RHR piping and Low Temperature Over pressure/Cold Over Pressure Protection System (LTOP/COPPS) requirements also have utilized the 440 psig setpoint for all calculations performed following the initial design change. This event had no adverse safety consequences in that at no time were the relief valves incapable of performing their intended safety functions of providing RHR system over pressure protection or LTOP/COPPS protection.

IV. Corrective Action

A Technical Specifications Change Request is being processed to revise the value identified in Technical Specifications 3.4.9.3.a.2 and 3.4.9.3.a.3 from the current value of 450 psig setpoint to correct setpoint value of 440 psig.

The programmatic conditions which allowed this event to occur were investigated and have been identified along with appropriate corrective actions. Specifically, the implementation of the original design change was not evaluated in terms of its impact on the unit's Technical Specifications.

**LICENSEE EVENT REPORT (LER)**  
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)				PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

The latest Design Control Manual (Rev. 03) and related Nuclear Group Procedures for safety evaluations provide additional guidance for considering the impact of modifications and design changes on the unit's Technical Specifications during the design change process. The use of an Integrated Safety Evaluation was implemented after this design change was implemented. The use of an Integrated Safety Evaluation is an additional barrier present today which is designed to prevent this type of error from occurring.

V. Additional Information

Not Applicable

Similar Events

None

Manufacturer Data

EIIS System Code: Residual Heat Removal (PWR) - BP  
EIIS Component Code: Valve, Relief - RV