Northeast
Utilities System

Millstone Offices • Rope Ferry Rd., Waterford, CT

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June 14, 1996

Docket No. 50-423
B15753

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

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

This letter forwards Licensee Event Report 96-013-00, documenting a condition that was determined at Millstone Unit No. 3 on June 12, 1996. This LER is submitted pursuant to 10CFR50.73(a)(2)(ii)(B).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

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

Attachment: LER 96-013-00

cc: T. T. Martin, 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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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 3

TITLE (4)
Residual Heat Removal System Design Deficiency Due to Nonconservative Original Design Assumption

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
05	15	96	96	013	00	06	14	96	FACILITY NAME	DOCKET NUMBER

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

OPERATING MODE (9)	5	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 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)
William J. Temple, Nuclear Licensing Supervisor	(860)437-5904

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	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		08	16	96

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

On June 12, 1996, with the plant in Mode 5 at 0-percent power, an engineering evaluation determined that a design deficiency in the Residual Heat Removal System (RHS) was a condition that was outside the design basis of the plant. A loss of control air could cause the RHS control valves to fail open. If this condition occurred during the initial phase of a plant cool down, the Reactor Plant Component Cooling Water System (CCP) temperatures could go above the 125°F used in the system stress analysis.

The Safety Grade Cold Shutdown (SGCS) design requirements specify that the unit be capable of being brought to Cold Shutdown with limited operator action outside the control room. If RHS heat exchanger operation is initiated at a 350°F RCS temperature as currently assumed in the analysis, and if the RHS throttle control valves 3RHS*HCV606/607 were to fail open, the RHS heat exchanger CCP outlet temperature is estimated to be 250°F. This would have created the potential for the CCP piping to not meet the ASME Appendix F stress criteria. This condition was reported June 12, 1996, as a condition outside the design basis of the plant, pursuant to 10CFR50.72(b)(1)(ii)(B).

The original plant design did not consider that the RHS flow control valves failing open on a loss of air, could create unacceptably high RHS heat exchanger discharge temperatures.

The corrective actions will be described in a supplement to this LER.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Millstone Nuclear Power Station Unit 3	05000423	96	--	013	--	00	2 of 3

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

I. Description of Event

On June 12, 1996, with the plant in Mode 5 at 0-percent power, an engineering evaluation determined that a design deficiency in the Residual Heat Removal System (RHS) was a condition that was outside the design basis of the plant. A loss of control air could cause the RHS control valves 3RHS*HCV606 and/or 3RHS*HCV607 to fail open. If this condition occurred during the initial phase of a plant cool down, the Reactor Plant Component Cooling Water System (CCP) temperatures could go above the 125°F used in the system stress analysis.

The Safety Grade Cold Shutdown (SGCS) design requirements specify that the unit be capable of being brought to Cold Shutdown with limited operator action outside the control room. In the SGCS design basis, the only operator action outside the control room (with no single failure to overcome) is repowering the RHS isolation valves. These valves are de-energized to preclude the possibility of a spurious opening.

The potential effects of higher CCP temperatures was first questioned by system engineers on May 15, 1996, during a review of plant design documentation. A review determined that if RHS heat exchanger operation is initiated at a 350°F RCS temperature, as assumed in the SGCS analysis, then the RHS heat exchanger CCP outlet temperature could be 250°F, if 3RHS*HCV606/607 failed open. It was subsequently determined that under the resultant conditions the CCP piping may not have met the ASME Appendix F stress criteria.

The design of the RHS flow control is as follows. Flow control through each RHS train is provided by a normally open control valve downstream of each heat exchanger (3RHS*HCV606 and 607) in conjunction with a normally closed control valve located in a bypass line around each heat exchanger (3RHS*FCV618 and 619). Should 3RHS*HCV606 and 607 fail open, the original plant design credited plant operators using only one train and operator control of the RHS pumps to control the RCS cooldown rate.

II. Cause of Event

The original design did not consider that the failure mode of the RHS flow control valves failing open, could create unacceptably high RHS heat exchanger CCP discharge temperatures.

III. Analysis of Event

This condition is reported as a condition outside the design basis of the plant, pursuant to 10CFR50.73(a)(2)(ii)(B). Specifically, the SGCS design basis analysis (without instrument air available) was not properly coupled to the CCP piping stress analysis assumptions. There was a potential that the CCP piping would not have met ASME Appendix F stress criteria given the potential high operating temperatures caused by instrument air unavailability and RHS System operation from the control room as required by SGCS design requirements.

Normal Cooldown

The RHS system operating practice is to normally have one RHS train aligned for shutdown cooling while one train is aligned for Safety Injection (SI) when high RCS temperatures exist. Therefore, only one train is likely to be adversely affected by high operating temperatures given a loss of instrument air.

If an RHS flow control valve failed open, the operator is alerted by a process computer CCP high temperature alarm (common alarm annunciator with specific alarm output). The 250°F maximum temperature is based on the CCP heat exchanger outlet temperature increasing from normal conditions (60 to 95°F) to 160°F. If RHS heat exchanger CCP inlet temperature is 95°F, the minimum CCP outlet temperature is limited to approximately 190°F. Therefore, there is

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

sufficient time available for operator action to attenuate the peak CCP operating temperature. The RHS pump is controlled from the control room and the operators would stop the pump to prevent temperatures from exceeding excessive values which would challenge CCP piping system operability.

Given a loss of instrument air and damage to one CCP train due to high temperature damage, it is determined that plant operators would use local manual action to place the unaffected train into service and then complete the cooldown, or remain in hot standby conditions, restore the instrument air system, then complete the cooldown.

In summary, based on operating practices and engineering judgment, at least one RHS and CCP train would not be subjected to high temperatures and would remain available for shutdown cooling. The potentially affected RHS and CCP train would also remain operable based on the limited operator action needed to attenuate the peak CCP system temperatures which would keep piping stresses below ASME III, Appendix F limits.

Safety Grade Cold Shutdown

For the SGCS analysis, the time that the RHS is placed into service is approximately 10 hours after reactor shutdown. If instrument air is not available, plant operators would not attempt to conduct the cooldown from the control room. The RHS flow control would be done locally, if instrument air could not be restored. Therefore, the safety function could have been accomplished and the postulated condition has low safety significance.

This condition is associated with conformance to the Branch Technical Position RSB 5-1, which the plant is designed to meet. The SGCS design basis analysis (without the non-safety grade instrument air available) was not properly coupled to the CCP piping stress analysis The nonconformance to the SGCS design scenario is the basis for reporting this condition as outside the design basis of the plant.

IV. Corrective Action

The corrective action for the condition will be described in a supplement to this LER.

V. Additional Information

Similar Events

LER 96-006-00, "Plant Shutdown Required by Technical Specifications, for Auxiliary Feedwater Containment Isolation Valves Declared Inoperable." This LER involved an original plant design discrepancy with a containment isolation valve not being capable of remaining closed against maximum accident pressure.

LER 96-007-00. "Containment Recirculation Spray and Quench Spray System Outside Design Basis due to Design Errors." This LER involved an original plant design deficiency with piping and supports not being adequately designed for loads resulting from accident temperatures.

Manufacturer Data

EIIS System Codes
Residual Heat Removal System - BP
Reactor Plant Component Cooling Water System - CC