Northeast Utilities System

Millstone Offices • Rope Ferry Rd., Waterford, CT

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

October 25, 1996

Docket No. 50-423
B15941

Re: 10CFR 50.73(a)(2)(vii)(D)

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

This letter forwards Licensee Event Report 96-036-00, documenting a condition that was determined at Millstone Unit No. 3 on September 29, 1996. This LER is submitted pursuant to 10CFR 50.73(a)(2)(vii)(D).

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

B15941-01: Safety related Air Operated Valves (AOVs) controlled by non-safety-grade circuits will be reviewed to determine the effect of mis-positioning of components on safety system accident performance. This review will be completed prior to entry into mode 4 from the current outage.

B15941-02: Corrective actions will be implemented as required to correct deficiencies identified during the review which would adversely affect safety systems accident performance. These actions will be completed prior to entry into mode 4 from the current outage.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

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

Attachment: LER 96-036-00

IE22 1/1

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

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 4

TITLE (4)
Safety Related Valves Controlled by Non-Safety Equipment

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	29	96	96	036	00	10	26	96	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)	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)	[X] 50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
J.M. Peschel, 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)

[X] YES (If yes, complete EXPECTED SUBMISSION DATE). [ ] NO

EXPECTED SUBMISSION	MONTH	DAY	YEAR
	03	31	97

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

On September 29, 1996, with the plant in mode 5 of an extended outage, while performing an engineering evaluation, it was concluded that the High Pressure Safety Injection (SIH) and Low Pressure Safety Injection (SIL) systems were subject to degraded performance due to possible mis-positioning of normally closed safety related air operated valves (AOVs). Mis-positioning of these valves is postulated to occur as a result of possible failures related to non-qualified power and control circuits. IEEE Std 379 as supplemented by RG 1.53, requires that components which are not qualified for seismic events or accident environments, and non-safety-grade components and systems should be assumed to fail to function if such failure adversely affects protective system performance. Contrary to this requirement, several components within the SIH and SIL systems were not properly analyzed for all potential failures. As a result, the potential diversion of SIH and/or SIL flow under accident conditions may be more than the margin allowed within the LOCA analysis.

This condition was reported at 1434 on September 29, 1996, pursuant to 10CFR50.72(b)(1)(iii)(D) as a condition that could have prevented the fulfillment of the safety function of a system needed to mitigate the consequences of an accident.

Safety Related Air Operated Valves (AOVs) controlled by non-safety-grade, non-qualified components which could adversely impact safety system performance will be reviewed. The effect of mis-positioning the subject AOVs on the LOCA analysis will be evaluated. Based on the outcome of this review, corrective actions will be identified and implemented as required. The plant will not enter mode 4 until the analysis has been performed and the required corrective actions have been completed.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Millstone Nuclear Power Station Unit 3	05000423	96	--	036	--	00	2 of 4

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

I. Description of Event

On September 29, 1996, with the plant in mode 5 of an extended outage, while performing an engineering evaluation, it was concluded that the High Pressure Safety Injection (SIH) and Low Pressure Safety Injection (SIL) systems were subject to degraded performance due to possible mis-positioning of normally closed, safety related, air operated valves (AOVs). Mis-positioning of these valves is postulated to occur as a result of possible failures related to non-qualified power and control circuits. The valves affected are the accumulator fill valves (3SIL*AV8878A/B/C/D), the SIH to SIL isolation valves (3SIL*AV 8872A/B/C/D), accumulator test line isolation valves (3SIL*AV 8877A/B/C/D), accumulator injection fill line isolation valves (3SIL*AV8879A/B/C/D), charging pump test line isolation valve (3SIH*AV8882), SIH pump hot leg test line isolation valves (3SIL*AV8889A/C), and the Residual Heat Removal system pump hot leg test line isolation valves (3SIH*AV8889B/D).

This condition was reported at 1434 on September 29, 1996, pursuant to 10CFR50.72(b)(1)(iii)(D) and is being reported pursuant to 10CFR50.73(a)(2)(vii)(D) as a condition that could have prevented the fulfillment of the safety function of a system needed to mitigate the consequence of an accident.

II. Cause of Event

The cause of the reported condition is a design error. The initial plant design did not adequately consider the effect of mis-positioning of these valves on the SIH and SIL system performance under accident conditions. This design error appears to have been a result of inadequate interface between the Nuclear Steam Supply System (NSSS) provider and the Architect/Engineer.

III. Analysis of Event

3SIL*AV8878A to D are normally closed valves and are located in each accumulator fill line inside the containment. The valves can be opened by an operator from the main control board when borated makeup water is required to be added to the accumulators. These valves are Category I components for the purpose of maintaining pressure boundary integrity. However, they are powered and controlled via circuitry which is not class 1E.

3SIL*AV8889B/D, 3SIL*AV8877A/B/C/D, 3SIL*AV8879A/B/C/D, 3SIL*AV8872A/B/C/D, and 3SIH*AV8882, 3SIH*AV8889A/C, are located in check valve test lines and are normally closed except for check valve leakage monitoring. The valves are opened by an operator from the main control board for the purpose of performing these checks. These valves are designed with a restricted port to limit the maximum flow rate when fully open with full system pressure across the valve. These valves are Category I components for the purpose of maintaining pressure boundary integrity. However, they are powered and controlled via circuitry which is not class 1E.

An interpretation of the IEEE standards conducted by the NSSS provider had determined that there were no credible failure mechanisms for the non-safety related control system which operates these valves. However, no supporting documentation of such an interpretation can be found. Therefore, it can not be assured that these valves will remain in their proper fail safe position in the event of a design basis accident. During a design basis accident (DBA), a failure could potentially lead to the diversion of safety injection flow and possibly result in pump runout.

The unit's Safety Evaluation Report (SER) states that the unit conforms to the design basis requirements of IEEE standard 379. IEEE Std 379 as supplemented by RG 1.53 requires that components which are not qualified for seismic events or accident environments, and non-safety-grade components and systems be assumed to fail if such failure adversely affects safety system performance. In general, the lack of equipment qualification may serve as a basis to assume failure. With all possible failures of non-safety-grade, non-qualified equipment assumed, the safety

system must then be capable of performing those functions required to mitigate the consequences of the specific event.

IEEE Standard 379 requires that the system single failure analysis be pre-conditioned with failure of all non-qualified control circuits. Contrary to this requirement, the SIH and SIL systems were not properly analyzed for all potential failures. As a result, the potential diversion of SIH and/or SIL flow under accident conditions may be more than the margin allowed within the LOCA analysis.

The safety implications and potential safety consequence of this condition are significant in that credible mechanisms exist that could result in the simultaneous failure of these non-safety-grade, non-qualified power and control circuits causing a loss of the ability to mitigate the consequences of an accident. This would result in off site dose consequences potentially in excess of 10CFR100 limits. However, no event occurred. Also the likelihood of such a common mode failure is remote.

IV. Corrective Action

Safety related Air Operated Valves (AOVs) controlled by non-safety-grade components which could adversely affect safety system performance will be reviewed. The effect of mispositioning the subject AOVs on the safety system accident performance and LOCA analysis will be evaluated. Based on the outcome of this review, corrective actions will be implemented as required. The plant will not enter mode 4 until such time as the analysis has been performed and the necessary corrective actions implemented. A supplemental report detailing the outcome of this review and the resultant corrective actions will be submitted.

V. Additional Information

None

Similar Events

LER 96-007-00 Containment Recirculation Spray and Quench Spray System Outside Design Basis due to Design Errors

On April 3, 1996, at 13:55, with the plant in Mode 5 at 0-percent power, it was determined that the Containment Recirculation System (RSS) spray piping and supports were not adequately designed for loads resulting from accident temperatures. It was initially determined that the higher RSS temperatures could result from a postulated loss of service water to one or more RSS heat exchangers. It was subsequently determined that: a) unacceptable stresses in the RSS and Quench Spray System (QSS) piping and supports could also result from the design basis accident temperatures inside containment, and b) the original design basis piping analyses utilized support anchor movements which were nonconservative.

As corrective actions, design reviews of the QSS and RSS were performed, appropriate design improvements made, and the systems restored to appropriate design basis requirements prior to declaring the systems operable. As action to prevent recurrence, other systems which are required to mitigate the consequences of a design basis accident were reviewed in order to determine whether or not they are susceptible to the same types of design deficiencies.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)
		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Millstone Nuclear Power Station Unit 3	05000423	96	--	036	--	00	4 of 4

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

LER 96-013-00 Residual Heat Removal System Design Deficiency Due to Nonconservative Original Design Assumption

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. 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.

Manufacturer Data

EIIS System Code:
High Pressure Safety Injection - BQ
Low Pressure Safety Injection - BP

EIIS Component Code:
Valve, Solenoid, Flow- TSV