Northeast
Nuclear Energy

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The Northeast Utilities System

APR 19 1996

Docket No. 50-336
B15664

Re: 10 CFR 50.73

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

This letter forwards Licensee Event Report (LER) 96-017-00 documenting an event that occurred at Millstone Nuclear Power Station, Unit No. 2 on March 20, 1996. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(ii).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

P. M. Richardson
Director - Millstone Unit No. 2

Attachment: LER 96-017-00

cc: T. T. Martin, Region I Administrator
P. D. Swetland, Senior Resident Inspector, Millstone Unit No. 2
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2

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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 2	05000336	1 of 3

TITLE (4)

Hydrogen Monitoring System does not Meet Single Failure Criteria

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
03	20	96	96	017	00	04	19	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	0%	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)
G. P. van Noordennen, Nuclear Licensing Supervisor	(860)440-2084

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

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

On March 20, 1996 at 1655 hours, with the plant in mode 5, at 0% power, it was discovered that the hydrogen monitoring system does not meet the single failure criterion required by Regulatory Guide 1.97. The corrective actions for LER 95-038 provided procedural guidance that would establish alternate electrical power, by installing a temporary jumper, to the respective outside containment isolation valve that could be potentially de-energized following a loss of a single DC bus. It was concluded that the operator actions required by the procedural guidance do not meet the single failure criterion of Regulatory Guide 1.97. This event is being reported pursuant to the requirements of 10 CFR 50.73(a)(2)(ii) "any event or condition that resulted in the condition of the nuclear power plant being in a condition that was outside the design basis of the plant."

The cause of this event was a to failure update the design basis to reflect the deviation of Regulatory Guide 1.97 requirements that existed subsequent to implementing the corrective actions for LER 95-038-00. Corrective actions include a design change to permanently re-power the outside containment sample line isolation valves to meet the single failure requirements of Regulatory Guide 1.97, and a continuing investigation to verify that the post accident sampling system (PASS) and the hydrogen monitoring system comply with their design basis requirements. There were no automatic or manually initiated safety systems actuated as a result of the condition.

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

I. Description of Event

On March 20, 1996 at 1655 hours, with the plant in mode 5, at 0% power, it was discovered that the hydrogen monitoring system does not meet the single failure criterion required by Regulatory Guide 1.97. The hydrogen monitoring equipment is designated as Category 1 and requires redundancy. LER 95-038 previously identified that following a Loss of Coolant Accident (LOCA) coincident with the loss of a DC Bus, a hydrogen monitoring flowpath could not be established due to the configuration of the power supply to the valves. The corrective action for LER 95-038 was to provide procedural guidance that would establish alternate electrical power, by installing a temporary jumper, to the respective outside containment isolation valve that could be potentially de-energized following a loss of a single DC bus. It was concluded that the operator actions required by the procedural guidance does not meet the single failure criterion of Regulatory Guide 1.97.

On March 20, 1996 at 1708 hours, with the plant in mode 5, at 0% power, a report was submitted pursuant to the requirements of 10 CFR 50.72(b)(1)(ii)(B), "any event or condition during operation that results in the condition of the nuclear power plant being in a condition that is outside the design basis of the plant."

There were no automatic or manually initiated safety systems actuated as a result of this event. Additionally, no operator actions required in response to this event.

II. Cause of Event

The cause of this event was a failure update the design basis to reflect the deviation to RG 1.97 requirements that existed subsequent to implementing the corrective actions for LER 95-038-00.

III. Analysis of Event

LER 95-038-00 previously identified that following a Loss of Coolant Accident (LOCA) coincident with the loss of a DC Bus, a hydrogen monitoring flowpath could not be established due to the configuration of the power supply to the valves. The corrective action for LER 95-038-00 was to provide procedural guidance that would establish alternate electrical power, by installing a temporary jumper, to the respective outside containment isolation valve that could be potentially de-energized following a loss of a single DC bus. It was concluded that the operator actions required by the procedural guidance does not meet the single failure criterion of Regulatory Guide 1.97. This event is being reported pursuant to the requirements of 10 CFR 50.73(a)(2)(ii) "any event or condition that resulted in the condition of the nuclear power plant being in a condition that was outside the design basis of the plant."

Both hydrogen monitors have similar flowpaths. The hydrogen monitoring system and PASS utilize the normal containment radiation monitor flowpath as the sampling flowpath post LOCA. The facility 1 hydrogen monitor has a facility 1 suction line isolation valve inside containment and a facility 1 return line isolation valve outside containment powered by facility 1 DC power. The outside containment suction isolation valve is powered by facility 2 DC power. The containment isolation signals are the same facility as the power to the valves. The discovery of this condition was reported in LER 95-038-00.

The event postulated is a LOCA, with the resulting initiation of a containment isolation signal, coincident with the loss of either the facility 1 or facility 2 DC bus. The current Post Incident Hydrogen Control Procedure provides guidance for the installation of a jumper that would either reroute facility 1 power to the facility 2 isolation valve or facility 2 power to the facility 1 isolation valve. By providing operator guidance for the

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

installation of an electrical jumper to allow opening of both isolation valves on a single hydrogen monitoring train, the containment isolation valves for hydrogen monitoring post accident sampling could be performed when required by the emergency operating procedures. However, the investigation concluded that this does not meet the single failure requirements for Regulatory Guide 1.97, Category 1, equipment requiring redundancy.

The containment isolation function of these valves was not affected by this event.

LER 96-009-00 identified that the original FSAR description reflecting a 12 hour sample time was never updated to reflect the current design basis, which requires the capability to analyze a sample within 3 hours following the decision that a sample is required. The safety evaluation that was prepared for the procedure change that implemented the guidance to install the electrical jumper was based on the 12 hour sampling time. The investigation concluded that the operator actions could be performed within 3 hours, therefore the FSAR discrepancy had no implications in this event.

The actual and potential safety significance of this event is considered low. Analysis confirmed that the connections to re-power the appropriate valves can be made within the control room in sufficient time (3 hours) to satisfy the post accident sampling requirements.

IV. Corrective Action

The following actions will be completed prior to the end of the current outage:

A design change to permanently re-power the outside containment sample line isolation valves to meet the single failure requirements of Regulatory Guide 1.97, for hydrogen monitoring, will be implemented.

An investigation is continuing to verify that PASS and the hydrogen monitoring system comply with their design basis requirements. This investigation and any modifications deemed necessary will be completed during the current outage.

V. Additional Information

Similar Events

LER 96-009-00
LER 95-038-00

Manufacturer Data

None