**NORTHEAST UTILITIES**

The Connecticut Light And Power Company
Western Massachusetts Electric Company
Holyoke Water Power Company
Northeast Utilities Service Company
Northeast Nuclear Energy Company

General Offices·Selden Street, Berlin Connecticut

P.O.BOX 270
HARTFORD, CONNECTICUT 06141-0270
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February 23, 1994
MP-94-136

DONALD B. MILLER, Jr.
*SENIOR VICE PRESIDENT - MILLSTONE*

Re: 10CFR50.73

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

Reference: Facility Operating License No. NPF-49
Docket No. 50-423
Licensee Event Report 94-004-00

Gentlemen:

This letter forwards Licensee Event Report 94-004-00 required to be submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(i) and 10CFR50.73(a)(2)(ii).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Donald B. Miller, Jr.
Senior Vice President - Millstone Station

DBM/BM:dlr

Attachment: LER 94-004-00

cc: T. T. Martin, Region I Administrator
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2 and 3
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3

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NRC Form 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES: 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Millstone Nuclear Power Station Unit 3 | 05000423 | 1 OF 3 |

TITLE (4)
Feedwater Isolation Valves Potentially Inoperable as a Historical Condition

| 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 |
| | | | | | | | | | | 05000 |
| 01 | 26 | 94 | 94 | - 004 - | 00 | 02 | 23 | 94 | FACILITY NAME | DOCKET NUMBER 05000 |

| OPERATING MODE (9) | 1 | THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | 20.402(b) | | 20.405(c) | | 50.73(a)(2)(iv) | | 73.71(b) |
| POWER LEVEL (10) | 85% | | 20.405(a)(1)(i) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | 73.71(c) |
| | | | 20.405(a)(1)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | OTHER |
| | | | 20.405(a)(1)(iii) | X | 50.73(a)(2)(i) | | 50.73(a)(2)(viii)(A) | | (Specify in Abstract below and in Text, NRC Form 366A) |
| | | | 20.405(a)(1)(iv) | X | 50.73(a)(2)(ii) | | 50.73(a)(2)(viii)(B) | | |
| | | | 20.405(a)(1)(v) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| William J. Temple, Site Licensing | (203) 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 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 January 26, 1994, with the plant in MODE 1 at 85% power, an engineering review determined that the Feedwater Isolation Valves may not have met design and Technical Specification closure requirements prior to 1989.

This is a report of a historical condition. It is reported as a condition outside the design basis, because there may not have always been sufficient nitrogen accumulator volume and/or pressure, which is needed for the valves to meet the closure times that are credited in the accident analysis. It is also reported as a non compliance with Technical Specifications, because the surveillance procedures may not have been adequate to determine if there was always sufficient nitrogen accumulator volume and/or pressure prior to 1989.

There is no current safety significance. Historically, if the valves may not have met the closure times credited in the accident analysis, then the mass and energy releases to the containment from a postulated feedwater or steamline rupture might have exceeded analyzed values, assuming a single active failure.

The historical concern was identified as a result of questions raised by the plant offsite safety review committee. This led to a review of the valve accumulator design adequacy. It was determined during the engineering review on January 26, 1994, that the design deficiency could have existed, and the NRC was promptly notified.

The root cause involved program failure and an inadequate design of the original Feedwater Isolation Valve nitrogen accumulator. No corrective actions are needed because design changes and procedure changes were made that have precluded the potential condition since 1989.

I. Description of Event

On January 26, 1994, with the plant in MODE 1 at 85% power, an engineering review determined that the Feedwater Isolation Valves may not have met the design and Technical Specification closure requirements prior to 1989. It was determined that there may not have always been sufficient nitrogen accumulator volume and/or pressure, which is needed for the valves to meet the closure times that are credited in the accident analysis. Also, the surveillance procedures may not have been adequate to determine if there was always a sufficient nitrogen accumulator volume prior to 1989.

Historically, if the valves may not have met the closure times credited in the accident analysis, then the mass and energy releases to the containment from a postulated feedwater or steamline rupture might have exceeded analyzed values, assuming a single active failure.

The Feedwater Isolation Valves (FWIVs) are hydraulically operated valves designed by Anchor/Darling with two nitrogen accumulators per valve. A hydraulic piston is located within each accumulator with nitrogen on one side and hydraulic oil on the other. A pressure switch and gage is connected to the nitrogen side of each accumulator. The nitrogen volume provides the required motive force to fast close the FWIVs in 5 seconds under full feedwater flow conditions. The accumulators are initially precharged with nitrogen to a pressure determined based on accumulator surface temperature and then the installed hydraulic pump is used to raise and maintain the accumulator pressure at 5000 psig. This process results in the movement of the hydraulic piston and the compression of the nitrogen gas. If a subsequent reduction in pressure is caused by a nitrogen leak, the hydraulic pump will maintain the required pressure by making up with hydraulic oil and displacing the hydraulic piston. As the piston travels it eventually bottoms out in the accumulator and can no-longer compress the nitrogen to maintain 5000 psig. At that point any further reduction cannot be made up by hydraulic oil and a low pressure alarm will occur. The problem is that before this condition occurs the volume of nitrogen required for fast closure has already been lost. For that reason during the third refueling outage completed in April 1991, a design change was implemented to add a stop tube cylinder inside the accumulator to limit piston travel and allow a low pressure alarm to occur while there is still enough nitrogen to permit fast closure.

On April 12, 1989, Anchor/Darling, responding to the plant's request, indicated that the setpoints on the FWIV actuator accumulator pressure switches should be 4750 psig. The original setpoint was 2250 psig. The fast closure capability of the isolation valves would be lost if the accumulator nitrogen pressure drops below 4650 psig. A Setpoint Change Request (SCR) was generated and approved to accomplish that change. In addition it was determined that the pressure switches were not seismically qualified. A Plant Design Change (PDCE) was generated to change the pressure switches to a model that meets the system seismic requirements. The PDCE and the SCR were completed during the third refueling outage in April 1991.

Prior to April 1989, there were occasions when low pressure alarms (2250 psig) were received due to nitrogen leaks. Both the reduced pressure and the preceding movement of the hydraulic cylinder would in the opinion of Anchor/Darling probably have prevented the fast closure capability of the valves. From April 1989 until the implementation of the identified design changes in April 1991, procedure changes were implemented that required monitoring of the nitrogen pressure and the hydraulic oil level on each accumulator each shift. Following the installation of the stop tubes and the new pressure switches set at 4750 psig during the third refueling outage, the FWIV's have fully met all operational requirements.

As a result of an offsite safety committee review and an engineering review, Anchor/Darling was requested to provide information regarding the original alarm setting of 2250 psig and information that would help in determining an assessment of the design basis. The 2250 psig appears to have been associated with the original plant proposed Greer actuators that were never installed on the Millstone 3 valves. Anchor/Darling responded that with reduced pressure and/or displacement of the hydraulic piston the valves may not have closed within five seconds and may not have closed completely.

II. Cause of Event

There are two root causes for the historical condition that may have led to the failure of the FWIV's to close. First, there was a program failure/technical error that did not pick up the requirement for a change in the alarm setpoint during the transition from the Greer operator (2250 psig) to the Anchor/Darling operator (4750 psig). The second root cause was an inadequate design of the original Feedwater Isolation Valve nitrogen accumulator. The lack of a stop tube may have resulted in an inadequate nitrogen volume to meet all valve operational requirements.

III. Analysis of Event

This is a report of a historical condition. It is reported under 10CFR50.73(a)(2)(ii)(B) as a condition outside the design basis, because there may not have always been sufficient nitrogen accumulator volume and/or pressure, which is needed for the valves to meet the closure times that are credited in the accident analysis. It is also reported under 10CFR50.73(a)(2)(i)(B) as a noncompliance with Technical Specifications, because the surveillance procedures may not have been adequate to determine if there was always sufficient nitrogen accumulator volume and/or pressure prior to 1989.

There is no current safety significance. Historically, if the valves may not have met the closure times credited in the accident analysis, then the mass and energy releases to the containment from a postulated feedwater or steamline rupture might have exceeded analyzed values, assuming a single active failure. However, there were no actual cases of feedwater isolation following a reactor trip when the valves failed to close as required.

IV. Corrective Action

It was determined during the engineering review on January 26, 1994, prior to receiving confirmation from Anchor/Darling, that the potential design deficiency could have historically existed, and the NRC was promptly notified under 10CFR50.72. This design deficiency was unique to the Feedwater Isolation Valves. Design changes and procedure changes were made that have precluded the potential condition since 1989. No other corrective actions are needed.

V. Additional Information

There have been no LERs similar to this event.

EIIS Codes

| System | Component |
|---|---|
| SJ (Feedwater) | ISV (Valve, Isolation) |