Northeast
Nuclear Energy

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Millstone Nuclear Power Station
Northeast Nuclear Energy Company
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The Northeast Utilities System

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

Re: 10CFR50.73(a)(2)(i)

April 14, 1994

MP-94-264

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

Reference: Facility Operating License No. DPR-65
Docket No. 50-336
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).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

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

DBM/EF:ljs

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
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2

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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 2	05000336	1 OF 05

TITLE (4) Failure to Perform Adequate ASME Leakage Test

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
03	19	94	94	- 004 -	00	04	14	94	FACILITY NAME	DOCKET NUMBER 05000

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

OPERATING MODE (9)	1		20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	99.8		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)		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 March 19, 1994, at 1918 hours, with the plant in Mode 1 at 99.8% power, an adequate post work ASME leakage test was not performed on 2-FW-43A, one of two Auxiliary Feedwater Regulating Valves (AFRVs), as required. This event was discovered on March 21, 1994, at 1100 hours during a review of the test results. The test was conducted with the valve in the incorrect, closed position.

The pre-approved test plan requirements, for the valve to be open during leakage testing, were based on an evaluation of the valve's internal components. During the post work valve testing, a change to the test method was made by the Shift Supervisor and Test Coordinator without consulting the test plan originators. This change was based solely on the drawing of the valve included in the Maintenance AFRV overhaul procedure. The drawing showed the valve bonnet exposed to pressure with the valve in both the open and closed position. Therefore, the valve was leak tested using Auxiliary Feedwater pump discharge pressure with the valve closed. On March 21st, during closeout documentation review, the test method change was called into question. The valve vendor was contacted due to questions concerning the Maintenance procedure drawing. It was determined that the drawing was not representative of the valve installed. The valve was immediately declared inoperable, bypassed and isolated. Following re-testing in the open position, and with no leakage detected, the valve was returned to operable status.

The root cause is considered to be a combination of personnel error and procedural deficiency. A personnel error occurred when the Shift Supervisor and Test Coordinator changed the test method without consulting the test plan originators. Although this is not required by procedure, good work practices dictates that the originator and approvers of the test plan be consulted. The inaccurate drawing in the Maintenance Procedure is considered a procedural deficiency. The drawing is being changed to reflect the installed valve. Drawings in other component specific procedures for safety related equipment are to be reviewed for accuracy.

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

I. Description of Event

On March 19, 1994, at 1918 hours, with the plant in Mode 1 at 99.8% power, an adequate post work ASME leakage test was not performed on 2-FW-43A, one of two Auxiliary Feedwater Regulating Valves (AFRVs), as required. This event was discovered on March 21, 1994, at 1100 hours during an review of the test results. The test was conducted with the valve in the incorrect, closed position.

The pre-approved test plan required that the AFRV be open during leakage testing. This was based on an evaluation of the valve's internal components. During the post work testing, a change was made to the test method by the Shift Supervisor and the Test Coordinator without consulting the test plan originator. The change was based solely on the drawing of the valve included in the Maintenance AFRV overhaul procedure (MP 2705A5). The drawing indicated that the valve bonnet and packing would be pressurized with the valve in the open or closed position. Since surveillance test SP 2610A already pressurizes the valve in the closed position, the Shift Supervisor and Test Coordinator felt the conditions for both tests would be satisfied. The VT-2 examiner was contacted and after the required time interval, a VT-2 inspection was performed for valve bonnet and packing leakage. The ASME required documentation in accordance with Engineering procedure 21218 was completed and the valve was returned to operable status at 1918 hours, March 19, 1994. On March 21st, during the closeout documentation review, the test method change was called into question. Differences existed between the procedure drawing and the valve component evaluation that resulted in the test plan. The valve vendor was contacted for clarification. He stated that based on the known condition of the valve internals (i.e., sealing threads only at the bottom of the cage, see attached drawings), the drawing was in error. This information was received at 1100 on March 21st. Immediately following the discovery of the inadequate test, the valve was declared inoperable, removed from service, re-tested and placed in service by 1427 hours the same day. No leakage was detected during the retest.

There were no automatic or manually initiated safety systems actuated as a result of this event.

II. Cause of Event

The root cause of the event is considered to be a combination of personnel error and procedural deficiency. A personnel error occurred when the Shift Supervisor and the Test Coordinator changed the test method without consulting the test plan originator. Although this is not required by procedure, good work practices dictates that the originator and approvers of the test plan be consulted. The inaccurate drawing in the Maintenance procedure is considered a procedural deficiency, in that this component specific procedure contained a "typical" valve drawing that was not representative of the actual valve.

III. Analysis of Event

Based on event investigation, a determination was made that 2-FW-43A was returned to operable status following repair without an adequate ASME required leakage test. This is reportable under the criteria of 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications. Technical Specification 4.0.5 requires In-service Testing of ASME Code Class 1,2, and 3 valves to be performed in accordance with Section XI of the ASME code. The valve remained in-service without adequate testing for 40 hours.

There were no safety consequences as a result of this event based on the fact that no leakage was present during the retest.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	94	- 004 -	00	03 OF 05

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

IV. Corrective Action

Following the event discovery on March 21, 1994, immediate corrective action was to declare the valve inoperable. The valve was then bypassed and isolated for re-testing.

The valve drawing in the Maintenance procedure is being corrected to reflect the installed valve internals. Other component specific procedures for safety related equipment are to be reviewed to verify drawing accuracy.

Since Engineering procedure 21218 is a generic leak test procedure using normal system alignments, no provision was made in it to document specific test conditions. In this event, test requirements were documented on a Work Implementation Plan (WIP), a form developed as a temporary aid to communications and sequencing of work. The use of a WIP is not procedurally controlled, however, guidelines are provided for its use. These guidelines do provide for changes, however, not through a formal change process, such as reviews and approvals. Although not a cause of the event, documentation of specific test conditions in the generic procedure for ASME leakage testing, may have lead to the basis for the original test conditions through a more formal test change process. Therefore, procedure enhancements to provide for the control of test details for generic tests are being considered.

V. Additional Information

There were no failed components associated with this event.

Similar LERs

This event involved an ASME leakage test that was performed inadequately. Similar events only include one event, LER 93-010-00, where a post work in-service leakage test was performed, when an ASME leakage test was required. This also involved equipment returned to service with inadequate testing.

EIIS Codes

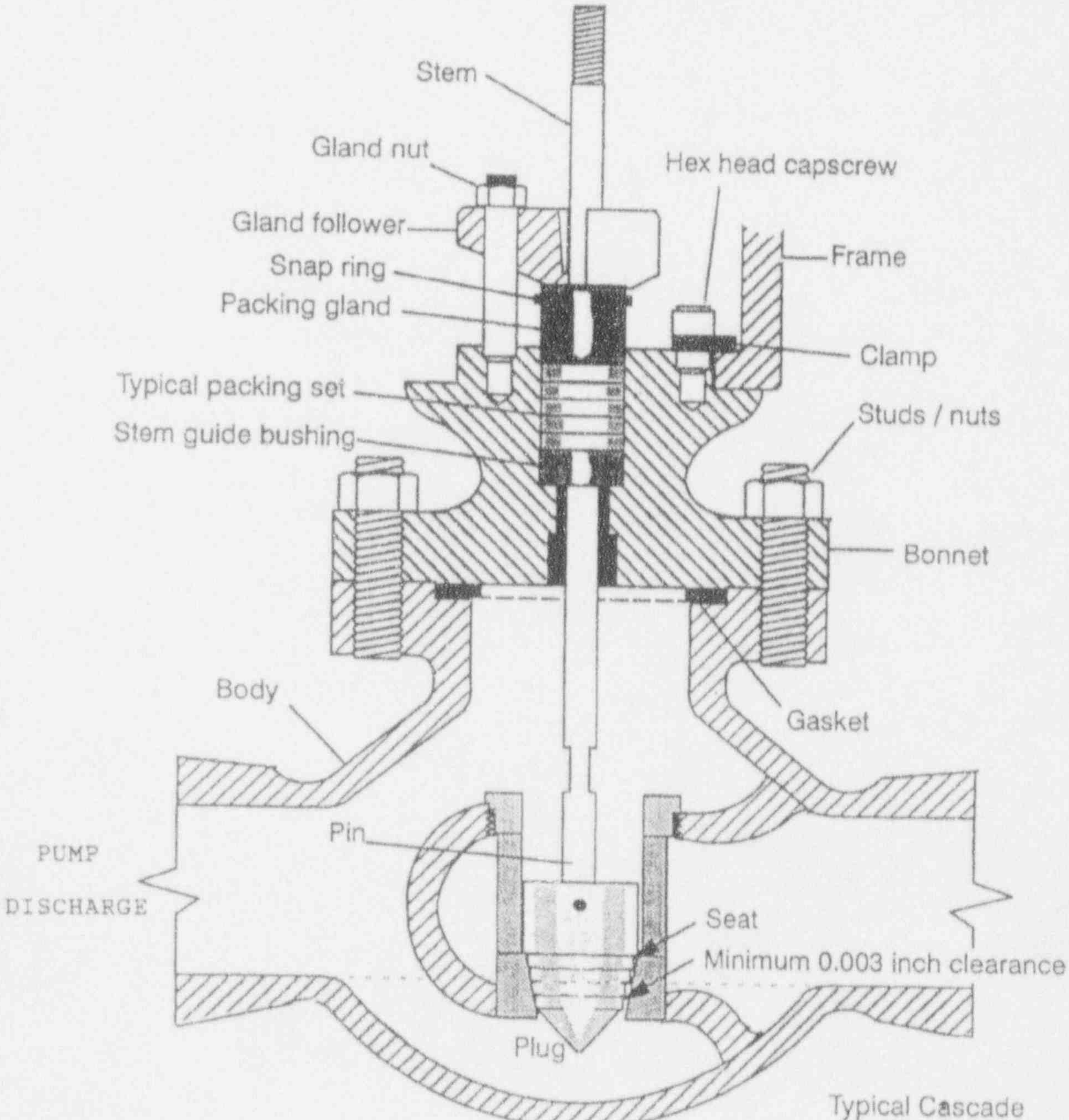
AFW Pumps: BA-P-T147 and BA-P-1075

AFRV: BA-FCV-C17

AFW system: BA

Attachments: MP2702A5 page 26 of 28, Existing Drawing and Required Drawing

EXISTING DRAWING



Typical drawing

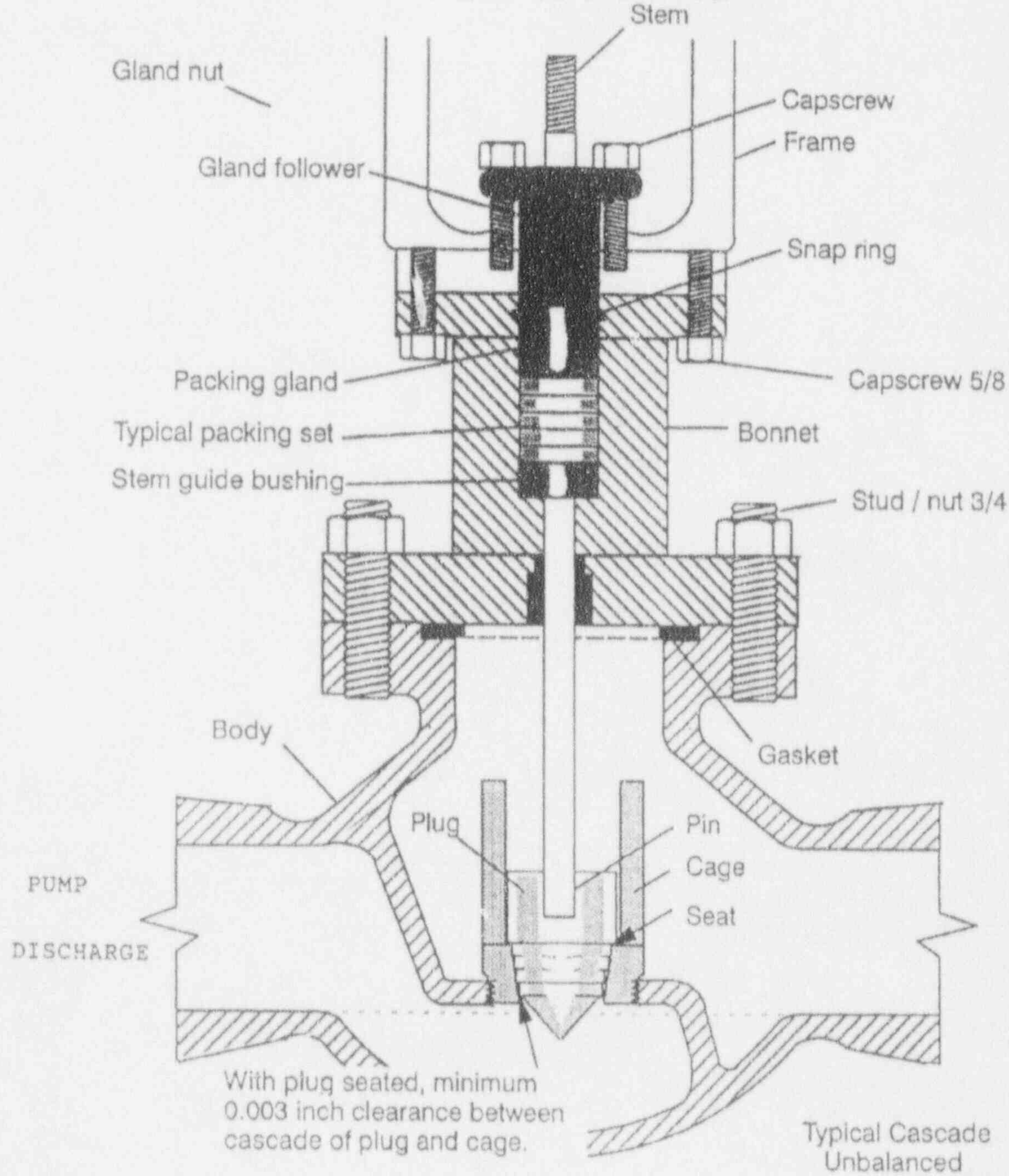
ATTACHMENT 1
(Sheet 1 of 2)

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	94	- 004 -	00	05 OF 05

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

REQUIRED DRAWING



Typical drawing

ATTACHMENT 1
(Sheet 2 of 2)