

Maine Yankee

RELIABLE ELECTRICITY SINCE 1972

329 BATH ROAD • BRUNSWICK, MAINE 04011 • (207) 798-4100

September 1, 1994

MN-94-95

JRH-94-215

UNITED STATES NUCLEAR REGULATORY COMMISSION

Attention: Document Control Desk

Washington, DC 20555

Reference: (a) License No. DPR-36 (Docket No. 50-309)

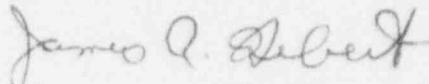
Subject: Maine Yankee Licensee Event Report 94-016, Emergency Feedwater
Isolation Valve Leakage

Gentlemen:

Please find enclosed Maine Yankee Licensee Event Report 94-016. This report is submitted in accordance with 10CFR50.73(a)(2)(ii).

Please contact us should you have any questions regarding this matter.

Very truly yours,



James R. Hebert, Manager
Licensing & Engineering Support Department

JRH

Enclosure

c: Mr. Thomas T. Martin
Mr. J. T. Yerokun
Mr. E. H. Trottier
Mr. Patrick J. Dostie

09044

9407140098 940902
PDR ADOCK 05000309
S PDR

IEP2
11

LICENSEE EVENT REPORT (LER)

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

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 (MNB 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) Maine Yankee Atomic Power Company	DOCKET NUMBER (2) 50-309	PAGE (3) 1 OF 4
--	-----------------------------	--------------------

TITLE (4)
Emergency Feedwater Isolation Valve Leakage

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
08	04	94	94	-- 016 --	00	09	02	94	NA	NA
									FACILITY NAME	DOCKET NUMBER
									NA	NA

OPERATING MODE (9) 3	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)										
POWER LEVEL (10) NA	20.402(b)			20.405(c)			50.73(a)(2)(iv)			73.71(b)	
	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)(vi)			OTHER	
	20.405(a)(1)(iii)			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 George Stowers, Nuclear Safety Specialist	TELEPHONE NUMBER (Include Area Code) (207) 882-6321
---	--

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)		
YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO		MONTH	DAY	YEAR
				NA	NA	NA

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

At 1220, on August 4, 1994 with the reactor in a cold shutdown condition, plant operators determined that an Emergency Feedwater isolation valve for #1 Steam Generator was leaking by. Further investigation identified similar leakage in the Emergency Feedwater supplies to #2 & #3 Steam Generators. Subsequently it was determined that under accident conditions which require isolation of Emergency Feedwater, isolation valve leakage could exceed Safety Analysis assumptions.

Maintenance activities have been performed to reduce Emergency Feedwater valve leakage. In addition, administrative controls have been implemented to ensure Emergency Feedwater leakage is maintained within the bounds of Safety Analysis assumptions during accident conditions.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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)	LER NUMBER (6)			PAGE (3)
Maine Yankee Atomic Power Company	50-309	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 OF 4
		94	-- 016 --	00	

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

On August 4, 1994 Maine Yankee was in a cold shutdown condition making preparations to restart the plant following a maintenance outage to correct Steam Generator tube leakage. At approximately 1220 while performing a leak test of the Emergency Feedwater (BA) Isolation Valve (ISV) for #1 Steam Generator, it was determined that EFW-A-338 leaked by at a rate of approximately 400 GPM. Leakage past #1 Steam Generator Emergency Feedwater Regulating Valve (FCV) EFW-A-101 at 75 GPM was also identified. These leakrates were determined using normal system instrumentation. Subsequent investigation determined that the actuator for EFW-A-338 was coupled to the disk approximately 180 degrees out of alignment. This misalignment is believed to have occurred during valve maintenance performed during our 1992 Refueling Outage. Maintenance activity was initiated to reposition the actuator/disk to the correct orientation.

At 1700 on August 5, 1994 it was determined that Maine Yankee's Steam Line Break Safety Analysis assumes zero leakage past Emergency Feedwater isolation and regulating valves. Therefore, at 1957 on August 5, the NRC was appraised of the situation via the Emergency Notification System in accordance with the provisions of 10 CFR 50.72 (b)(2)i. A follow up notification was made at 1151 on August 6, 1994 after leakage was identified past the Emergency Feedwater Isolation Valve for #3 Steam Generator, EFW-A-340.

On August 10, 1994 comprehensive testing was performed on the Emergency Feedwater isolation and regulating valves for all three Steam Generators using precision test equipment. These tests revealed leakrates for various individual valves and combinations of valves from 0 to 36 GPM per Steam Generator (See Attachment A). Safety Analysis sensitivity studies conducted in parallel with these tests established the following revised acceptance criteria:

- 1) Leakage past individual isolation/regulating valves should be less than 40 GPM. The purpose of this limit is to prevent excessive cooldown following a Main Steam Line Rupture with failure of either the isolation valve or regulating valve to close.
- 2) Leakage past both the isolation and regulating valve to each Steam Generator should be less than 10 GPM. The purpose of this limit is to provide at least 30 minutes for operators to ensure EFW is isolated to a faulted Steam Generator following a Steam Generator Tube Rupture. Such action is taken to ensure SG safety valves will not be lifted due to overfill.
- 3) Zero leakage to a Steam Generator with a tube rupture within 30 minutes. The purpose of this limit is to prevent lifting SG safety valves due to overfill following a Steam Generator Tube Rupture.

In order to meet the revised acceptance criteria, EFW-A-101 was adjusted to reduce leakage past EFW-A-101 & 338 to zero GPM and administrative controls were implemented to ensure termination of any potential future leakage to a ruptured Steam Generator within 30 minutes using manual isolation valves (V).

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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)	LER NUMBER (6)			PAGE (3)
Maine Yankee Atomic Power Company	50-309	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	3 OF 4
		94	-- 016 --	00	

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

An investigation is being conducted to identify the root cause(s) of this event and to identify actions which should be taken to prevent recurrence. Significant activities associated with this investigation include the following:

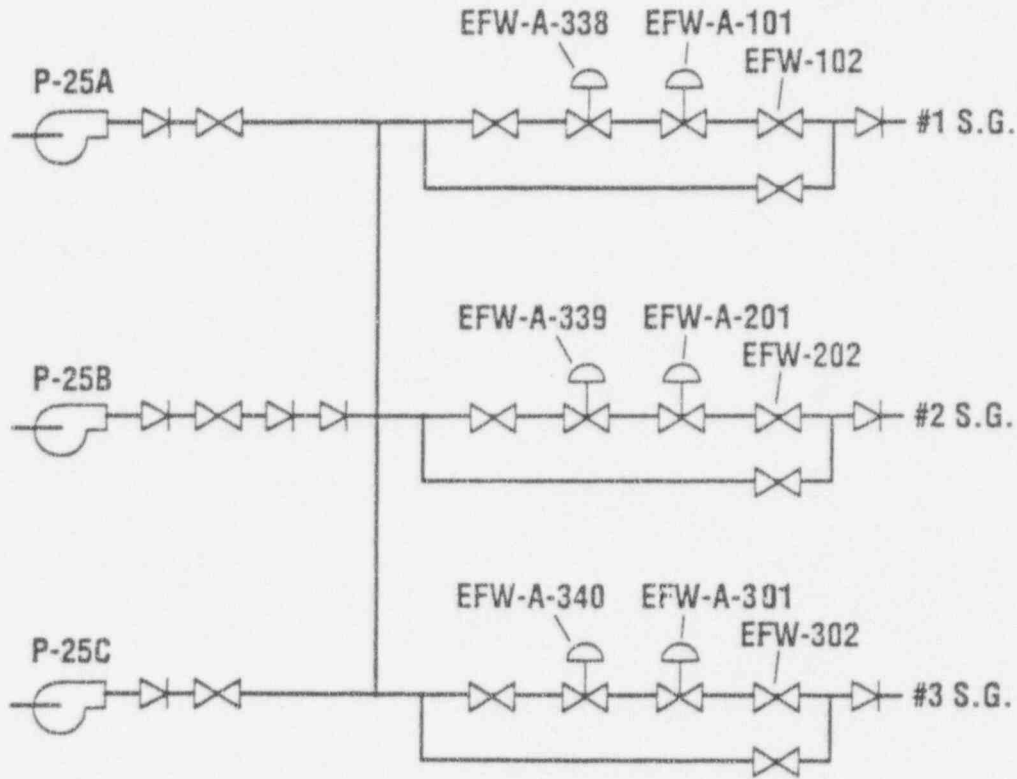
- o Performance of a root cause determination to identify and correct the causal factors contributing to the improper assembly of EFW-A-338
- o Continuation of an recently initiated task to improve the process used by Maine Yankee to ensure Safety Analysis inputs and assumptions are appropriately identified, disseminated, and periodically reverified.

LER 94-002, Inadequate Configuration Control in Design and Operation of Steam Generator Blowdown System, and LER 94-003, Service Water Flow Measured Outside Design Basis, describe prior events at Maine Yankee which are similar to those presented here.

Attachment A

Emergency Feedwater System

(Simplified Diagram)



SG	Valve(s) Closed	As Found 08/04/94	Test Results 08/10/94	As Left 08/11/94
1	EFW-A-101	75 GPM	36 GPM	0 GPM
	EFW-A-338	400 GPM	20 GPM	24 GPM
	EFW-A-101 & 338	-	15 GPM	0 GPM
	EFW-A-101, 338 & EFW-102	-	0 GPM	0 GPM
2	EFW-A-201	-	0 GPM	0 GPM
	EFW-A-339	-	17 GPM	17 GPM
	EFW-A 201 & 339	-	0 GPM	0 GPM
	EFW-A-201, 339 & EFW-202	-	0 GPM	0 GPM
3	EFW-A-301	-	0 GPM	0 GPM
	EFW-A-340	-	36 GPM	23 GPM
	EFW-A-301 & 340	-	4 GPM	0 GPM
	EFW-A-301, 340 & EFW-302	-	0 GPM	0 GPM