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July 3, 1991

C321-91-2181

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report

This letter forwards one copy of Licensee Event Report 91-002.

Sincerely,

John J. Barton
Vice President and Director

JJB/TB:jc
(WP\51:Covltrs)
Enclosure

cc: Administrator, Region 1
Senior NRC Resident Inspector
Oyster Creek NRC Project Manager

IE22
1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oyster Creek, Unit 1	0 5 0 0 0 2 1 9	1 OF 0 4

TITLE (4): Local Leak Rate Test Results in Excess of Limits due to Valve Degradation

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 2	2 1	9 1	9 1	0 0 2	0 0	0 7	0 3	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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

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)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
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 AREA CODE	
Donald J. Sheehan, Engineer Level III OPS ENG, PEB, OC	6 0 9	9 7 1 - 2 4 9 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	S B	I S V	A 5 8 5	Y	B	B F	I S V	F 1 3 0	Y
B	B F	I S V	A 5 8 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		1 2	1 5	9 1

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

During the 13R Refueling Outage, local leak rate testing (in accordance with 10CFR50, Appendix J) identified a Main Steam Isolation Valve (MSIV) with a leak rate in excess of the acceptance criteria of 12.08 SCFH at 20 psig as specified in Technical Specifications 4.5.F.2. The safety significance of this discovery is considered minimal since the other MSIV in the same header met the local leak rate test requirements.

Subsequent local leak rate testing identified a pair of isolation valves (in series) with a potential leak rate in excess of the acceptance criteria of 60% of the maximum allowable limit (La) at 35 psig as specified in Technical Specifications 4.5.F.1. Since the root cause for the excessive leakage past the Reactor Building-to-Torus vacuum breakers has not been identified, a root cause evaluation will be performed and a follow-up LER will be forwarded when the evaluation is completed.

These leaking valves were repaired and subsequent local leak rate testing verified that leakage rates were within the acceptance criteria.

Due to an improper interpretation of reporting requirements implemented by a recent Technical Specification Amendment, this condition was not reported within 30 days as required by 10CFR50.73.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oyster Creek, Unit 1	0 5 0 0 0 2 1 9	9 1	0 0 2	0 0	0 2 OF 0 4

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

Dat- of Occurrence

A Main Steam Isolation Valve failed a local leak rate test on February 21, 1991, at 0220 hours.

Two primary containment isolation valves in series failed a local leak rate test on March 7, 1991, at 1400 hours.

Identification of Occurrence

A Main Steam Isolation Valve (MSIV, EIIS Component SB-ISV) exceeded the leak rate criteria specified in Technical Specification 4.5.F.2.

Two primary containment isolation valves (in series) potentially exceeded the leak rate criteria specified in Technical Specification 4.5.F.1.

Inese conditions are considered reportable under 10CFR50.73(a)(2)(i).

Conditions Prior to Occurrence

The plant was in cold shutdown for a refueling outage.

Description of Occurrence

Local leak rate testing (LLRT) performed during the 13R refueling outage revealed two instances of reportable leakage. A Main Steam Isolation Valve (MSIV) had an individual leak rate in excess of the acceptance criteria of 12.08 SCFH at 20 psig as specified in Technical Specification (Tech Spec) 4.5.F.2. The test was performed at 0220 hours on February 21, 1991. This valve later passed an LLRT on May 20, 1991, at 2300 hours.

On March 7, 1991, at 1400 hours, an LLRT was performed on the Reactor Building-to-Torus vacuum breakers (V-26-17 and V-26-18, EIIS Component BF-ISV). These two primary containment isolation valves are in series, with the test connection between the two valves. Since both valves V-26-17 and V-26-18 were tested simultaneously, the leak rate past each individual valve was not measured. The test revealed an unquantifiable leak rate. Valve V-26-17 was determined to be [illegible]king by observing flow downstream of the valve. For this reason, V-26-17 was [illegible]pected. A soft-disc seat was replaced, and V-26-17 was re-assembled. On [illegible] 26, 1991, at 2200 hours a Post Maintenance Test (PMT) for V-26-17 revealed [illegible] although the leakage downstream of V-26-17 had stopped, leakage was still excessive. It was assumed that the remaining leakage was due to V-26-18. V-26-18 was removed, the soft-seat boot was replaced, and V-26-18 was re-assembled. As leakage was not quantified, it is assumed that this primary containment penetration had a potential leak rate that exceeded the limit of 60% La at 35 psig per Tech Spec 4.5.F.1. This penetration successfully passed an LLRT on May 26, 1991, at 1600 hours.

A Deviation Report (DR) was submitted to document the failed LLRT for the Reactor Building-to-Torus vacuum breaker penetration. While this condition was being analyzed, it was discovered that the LLRT failure of the MSIV on February 21, 1991, should have been reported. These conditions were not reported in a timely fashion (as specified in 10CFR50.73) due to misinterpretation of the Technical Specifications following an amendment update. On June 7, 1991, at 1400 hours a DR was submitted to notify appropriate management personnel that an LER for the MSIV LLRT failure had not been submitted.

Apparent Cause of Occurrence

The apparent cause of excessive leakage past the MSIV was misalignment of valve internals as evidenced by grooves found in the valve guide ribs.

The apparent cause of excessive leakage past V-26-17 was attributed to a damaged soft-disc seat. The apparent cause of leakage past V-26-18 was attributed to a damaged soft-seat boot. Since the root cause for the excessive leakage past the Reactor Building-to-Torus vacuum breakers has not been identified, a root cause evaluation will be performed and a follow-up LER will be forwarded when the evaluation is completed.

During cycle 12, Amendment 132 to the Oyster Creek Technical Specifications was issued by the United States Nuclear Regulatory Commission (USNRC). Amendment 132 significantly revised the reporting criteria of LLRT results. The new requirements were not clearly understood, and resulted in exceeding the 30-day period defined in 10CFR50.73 for submittal of Licensee Event Reports.

Analysis of Occurrence and Safety Assessment

The MSIVs are containment isolation valves designed to minimize coolant loss from the vessel, and the resulting offsite dose, in the event of a main steam line break accident. The safety significance of the excessive leak rate past the MSIV is considered minimal as the other MSIV in the same header met the leak rate acceptance criteria of Tech Spec 4.5.F.2.

Specifications concerning leakage have been established to determine operability status of the primary containment systems and valves. This is done to ensure their availability to control the release of radioactivity in the event of an accident condition. The Reactor Building-to-Torus vacuum breakers (V-26-17 and V-26-18) are primary containment isolation valves, and are in the same penetration. The extent of leakage past both the valves could not be quantified due to the method of testing, however, the penetration was degraded and the ability of the penetration to control the release of radioactive material following an accident was reduced. A conservative dose projection calculation was performed. The results of this calculation show that 10CFR100 exposure limits would not have been exceeded with the degraded Reactor Building-to-Torus vacuum breaker penetration.

Corrective Action

Because of a history of LLRT failures of the MSIVs, the manufacturer, Atwood and Morrill, modified the original design to provide assurance that the MSIVs would meet the LLRT requirements and to ensure reliable and acceptable valve seat tightness. The modification includes an improved main poppet design for better valve seating and alignment. This modification was performed on the leaking MSIV, NS04A. A satisfactory leak rate test was performed on May 20, 1991, at 2300 hours.

The Reactor Building-to-Torus vacuum breakers V-26-17 and V-26-18, were inspected and repaired. V-26-17 had a new soft-seat disc installed. V-26-18 had a new soft-seat boot installed. A new test connection was provided that will allow V-26-18 to be leak rate tested independently of V-26-17. This penetration satisfactorily passed a local leak rate test on May 26, 1991, at 1600 hours. Additional corrective action, if any, will be specified in the follow-up report pending root cause determination.

A memorandum was written that more clearly defines the reporting requirements for LLRT results. This memorandum was distributed to all personnel with responsibility for making reportability determinations. Reportability requirements will be addressed in the Deviation Report.

Failure Data

NS04A Angle Globe Valve, 24"
Manufacturer - Atwood and Morrill Company Inc./Xomox Corporation

V-26-17 Swing Check Valve, 20"
Manufacturer - Atwood and Morrill Company Inc./Xomox Corporation

V-26-18 Butterfly Valve, 20"
Manufacturer - Fisher Controls Company Incorporated

Similar Events

The following LER's are similar to this report:

- 82-14 - LLRT Failure
- 82-19 - LLRT Results on MSIV's Outside Limits
- 82-20 - LLRT Results on MSIV's Greater Than Limits
- 83-09 - LLRT Results
- 86-35 - Containment Penetration Found Degraded Due to Isolation Valves Actuator/Valve Linkages Out of Adjustment