GPU Nuclear

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January 18, 1995
C321-95-2028

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 94-022

Enclosed is the Licensee Event Report 94-022.

If there are any questions please contact Mr. John Rogers at 609.971.4893.

for John J. Barton
Vice President and Director
Oyster Creek

JJB/JJR
Attachment

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

NRC FORM 366N
(5-92)

LICENSEE EVENT REPORT (LER)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Primary Containment Leak Rate in Excess of Technical Specification Requirements Due To Maintenance Procedure Noncompliance

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
12	19	94	94	-- 022 --	00	01	18	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N

POWER LEVEL (10): 59

THIS REPORT IS 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)
	20.405(a)(1)(i)		50.36(c)(1)	X	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)
Robert Barbieri, System Engineer	609.971.4303

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

On December 19, 1994, while performing a reactor startup following the 15R refueling outage, it was noted that the amount of nitrogen being added to the primary containment was higher than normal. It was subsequently determined that leakage from the primary containment exceeded the Technical Specification limits. A reactor shutdown was initiated and an Unusual Event was declared. A subsequent evaluation determined that the Drywell vent and purge valves were leaking past their seats. The valves were declared inoperable. The reason for the seat leakage was determined to be valve actuator misadjustment.

Immediate corrective action was taken to readjust the valve stroke length and perform a local leak rate test required to declare the valves operable. The reactor shutdown and Unusual Event were terminated.

The cause of this event was determined to be failure to comply with existing maintenance administrative procedures. Long term corrective actions are planned which will raise the sensitivity and understanding of maintenance personnel to the need for procedure compliance.

DATE OF DISCOVERY

The containment leak rate was determined to be in excess of Technical Specification limits on December 19, 1994 at approximately 10:00 am.

IDENTIFICATION OF OCCURRENCE

Containment (EIIS JM) Isolation Valves (EIIS ISV) V-27-1 and V-27-2 were found to be leaking while in the closed position. The calculated leak rate exceeded the Technical Specification limit for maximum allowable leak rate (L_a).

A reactor shutdown was commenced in accordance with Technical Specification 3.5.A.7.

This event is considered reportable in accordance with 10 CFR 50.73(a)(2)(i), 10 CFR 50.73(a)(2)(ii), and 10 CFR 50.73(a)(2)(v).

CONDITIONS PRIOR TO DISCOVERY

At the time of the discovery of this event, the plant was operating at approximately 59% power. A plant startup following the 15R refueling outage was in progress.

DESCRIPTION OF OCCURRENCE

On December 18, 1994, during the plant startup following the 15R refueling outage, it was noted that the frequency and duration of nitrogen makeup to the primary containment appeared higher than normal. Efforts were initiated to determine the reason for the makeup. On December 19, 1994 at approximately 10:00 am, it was determined that the makeup was due to a leak from the primary containment, and the leak rate exceeded Technical Specification limits. A plant shutdown was initiated and additional efforts were employed to identify the source of leak.

DESCRIPTION OF OCCURRENCE (cont)

It was subsequently determined that the repairs could not be completed in 8 hours and an Unusual Event was declared. At approximately 2:50 pm, the leak was identified as passing through the seats of containment isolation valves V-27-1 and V-27-2. These are two eighteen inch butterfly valves in series in the Drywell vent exhaust path.

Further review revealed that maintenance had been performed on these valves during 15R after the acceptable as-found local leak rate test had been completed. Following the maintenance, no local leak rate tests were performed.

An immediate maintenance crew was assembled, briefed, and assigned to readjust the valves. Adjustment resulted in an immediate decrease in the rate of leakage. Following the adjustment, a local leak rate test was performed satisfactorily. At 9:06 pm, the valves were declared operable, and the reactor shutdown and Unusual Event were terminated.

APPARENT CAUSE OF THE OCCURRENCE

The root cause of this event was failure to comply with existing administrative procedures relating to the control of emergent work.

ANALYSIS OF OCCURRENCE AN SAFETY ASSESSMENT

The safety function of these two valves is to close upon a containment isolation signal and maintain leak tightness within the Technical Specification limits. Based on calculations performed to evaluate this occurrence at normal drywell conditions, the leakage was approximately 900 scfh, while the leakage limit from the containment at design basis conditions would have been approximately 426 scfh. The leakage path was the design release pathway through the exhaust stack and is fully monitored. No radioactive release through this path occurred during this event. If a design basis accident had occurred, automatic systems would have redirected the flow through the Standby Gas Treatment System for high efficiency filtering.

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ANALYSIS OF OCCURRENCE AN SAFETY ASSESSMENT (cont)

Additionally, the magnitude of the release would have been well within design limits as: 1) The reactor was at less than full power with no significant power history; 2) there are no fuel failures or other sources of increased activity which would have contributed to an elevated initial source term; and 3) the nuclear steam supply system had recently passed an ASME Code pressure test and subsequent 1000 psi visual inspection prior to the plant startup. The actual consequences of this event were considered to be minimal as no radioactive release occurred.

CORRECTIVE ACTIONS

Upon determination that the leakage from the drywell was in excess of Technical Specification limits, a reactor shutdown was commenced and an Unusual Event was declared. Immediate corrective actions were initiated to readjust the valves to leak tightness, and a post maintenance local leak rate test was performed for verification. Additionally, the Maintenance Director issued a letter reinforcing the existing administrative requirements dealing with work scope changes.

The following long term corrective actions have also been initiated:

1. Senior Maintenance Management will meet with maintenance supervision to discuss this event and lessons learned to reinforce the need for procedure compliance and personal accountability. This action is scheduled to be completed by January 31, 1995.

2. Cyclic training will be provided to maintenance personnel to reinforce the areas of procedure use and adherence, equipment control, and administrative controls for job orders. This action is scheduled to be completed by March 31, 1995.

3. A letter will be provided by the Maintenance Director to all maintenance personnel addressing the lessons learned from this event and re-emphasizing the previous communication regarding management expectations. This action is scheduled to be completed by January 31, 1995.

SIMILAR EVENTS

None.