GPU
NUCLEAR

GPU Nuclear, Inc.
U.S. Route #9 South
Post Office Box 388
Forked River, NJ 08731-0388
Tel 609-971-4000

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

6730-97-2171
June 20,1997

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report 96-011, Revision 1
Primary Containment Leak Rate in Excess of Tech Spec
Requirements due to Incorrect Reassembly of Valve Cover

Enclosed is Licensee Event Report 96-011, Revision 1. Bars have been placed in the right margins to indicate the revised sections. This event did not impact the health and safety of the public.

If any additional information or assistance is required, please contact Mr. John Rogers of my staff at 609-971-4893.

Very truly yours,

for Michael B. Roche
Vice President and Director
Oyster Creek

030069

MBR/JJR

Enclosure

cc: Administrator, Region I
NRC Project Manager
NRC Sr. Resident Inspector

IE22 1/1

NRC FORM 366
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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OYSTER CREEK, UNIT 1	50-219	1 OF 6

TITLE (4)

Primary Containment Leak Rate in Excess of Tech Spec Requirements Due to Incorrect Assembly of Valve Cover

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
Month	Day	Year	Year	Sequential Number	Revision	Month	Day	Year	Facility Name	Docket Number
10	26	96	96	-- 011 --	01				FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	0.0

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

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36©(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36©(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
James E. Frank, System Engineer	609-971-4114

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	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	NO x				

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

At 11:30 PM on October 26, 1996, a pressure drop test was performed on the primary containment. This evaluation was completed at 1:30 AM on October 27, 1996, which indicated that the leak rate was above the Technical Specification limit. A Torus to drywell vacuum breaker cover was found to be leaking. The valve cover was repaired and gross containment leakage was calculated to be below Technical Specification allowable leakage. Containment integrity had been required for a five day period prior to the discovery.

The cause of this event was determined to be an improper valve reassembly on V-26-5 during the 16R outage. Procedure changes were made to emphasize self checking to ensure the cover plate is level during and after the torquing sequence. An evaluation of the methodology used in performing Local Leak Rate Testing of the torus to drywell vacuum breakers will be done.

Additionally, this specific human performance error, as well as others, was discussed at weekly craft/management interface meetings.

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

DATE OF DISCOVERY

Primary containment leak rate was calculated to be above Technical Specification limits on October 27, 1996, at approximately 0210 hours.

IDENTIFICATION OF OCCURRENCE

Containment (EIIS JM) Torus to Drywell (EIIS BF) vacuum breaker V-26-5 (EIIC RV) cover was found to be leaking while containment was pressurized.

The leakage path was present during five days of reactor operation following the 16R outage.

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

CONDITIONS PRIOR TO OCCURRENCE

At the time of the discovery, the plant was shutdown, less than 212° and vented to atmosphere. Primary containment was not required to be in effect.

The reactor had been operated in a startup mode with containment integrity in effect for approximately five days prior to the discovery. Primary containment was inerted and pressurized with nitrogen for 40 hours prior to the manual plant shutdown on October 25, 1996.

DESCRIPTION OF OCCURRENCE

During the 16R outage all torus to drywell vacuum breaker valves were disassembled for mechanical surveillance and limit switch calibration. All passed the cover plate local leak rate tests after reassembly.

On October 20, 1996, the reactor was started and containment integrity was required. The plant was then shutdown to repair a steam leak, and subsequently restarted on October 22, 1996. An annotated chronology is presented below:

<u>10/22/96</u>

6:20 AM	Placed Mode Switch to STARTUP. (This requires primary containment to be intact.)
9:10 AM	Reactor Critical
-----	All Day – Reactor startup and heatup continued.
6:44 PM	Personnel enter containment to perform 1000 lb. inspection.

DESCRIPTION OF OCCURRENCE (Cont.)

10/23/96

1:25 AM Personnel exited the containment and the drywell was secured. Reactor power was approximately 2-5%.
5:07 AM Commenced inerting the Drywell with nitrogen.
11:20 AM Commenced pressurizing the Drywell.
11:47 AM Completed inerting drywell at approximately 1 psig.. Commenced inerting the torus.
4:26 PM Completed inerting the Torus.

Over the next few hours, questions developed for the first time as to whether the amount of nitrogen being introduced into the drywell was greater than normal.

6:30 PM Secured all nitrogen inerting to inspect the nitrogen makeup line for possible leakage.
8:04 PM Operations Director directs the Shift Technical Advisors to monitor nitrogen makeup and evaluate the condition of the primary containment.

A program to monitor drywell conditions and nitrogen makeup was initiated. An ongoing analysis of the leakrate indicated that it was not excessive. At this point the primary containment was evaluated as intact.

10/24/96

10:10 AM Nitrogen compressors were placed into service.

The reactor and drywell parameters were constantly changing as plant conditions approached equilibrium. The containment continued to be evaluated as intact. The accuracy of the calculated leakage was low as containment conditions continued to vary during the power ascension. Also, a significant contributor to the make up flow was identified at the nitrogen compressor outside of containment. This compressor leakage is not containment leakage. Further monitoring of the leakage continued during power ascension.

10/25/96

5:07 AM The calculated leak rate was greater than those previously calculated.

It was suspected that the leak was in the nitrogen compressor crankcase. The system engineer was contacted and repair plans were initiated. Therefore, it was believed that if a leak existed, it was from the source of the nitrogen, and not from the containment.

5:56 AM The reactor was at full power.
11:59 AM A manual reactor scram was initiated in response to a main generator (EIIS EL) runback.

DESCRIPTION OF OCCURRENCE (Cont.)

<u>10/26/96</u>

1:40 AM The reactor was in COLD SHUTDOWN. Primary containment was not required. Primary containment integrity had been required for 133 hours.

5:00 PM Engineering Action Plan 96-51 was initiated to determine if a primary containment leak existed and to attempt to quantify the leakage if it did exist.

11:30 PM Completed Action Plan 96-51. Engineering commenced evaluating data.

<u>10/27/96</u>

1:30 AM Engineering determined that the containment leak rate exceeded Technical Specification limits

Although the accuracy of the collected data could not be confirmed, it was conservatively assumed to be correct at approximately double the Technical Specification limit.

2:21 AM Plant walkdowns and inspections were initiated to locate the leak.

8:45 AM A leak was identified in the cover of vacuum breaker V-26-5. Additional smaller leaks were noted.

12:30 PM Leak testing and searching was halted to depressurize the drywell and repair the leaks.

The containment was depressurized and V-26-5 was disassembled. The O rings were replaced and the valve was reassembled.

5:18 PM Repairs were completed. Local leak rate test on the cover of V-26-5 was completed satisfactorily.

7:22 PM Containment was pressurized with nitrogen to evaluate gross containment leakage.

<u>10/28/96</u>

1:45 AM Engineering determined that the gross containment leakage was reduced and was within Technical Specification limits.

APPARENT CAUSE OF OCCURRENCE

The elevated containment leakage was caused by improper valve assembly during the previous refueling outage. The valve body to cover leakage was due to improperly seated O rings. The inner O ring was found pinched between the cover and the valve body. This caused the outer O ring to not properly seat. The root cause of this occurrence was personnel error in that improper valve maintenance was performed. A contributory cause to this event was that the Local Leak Rate Test performed is not valid unless the cover is properly seated (square). In this case, the pinched o-ring prevented the cover from seating.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The safety significance of this occurrence has been determined to be minimal.

The function of the V-26-5 cover is to maintain primary containment integrity. If a design basis accident had occurred, automatic systems would have re-directed Reactor Building exhaust and V-26-5 cover leakage through the Standby Gas Treatment System (SGTS) for high efficiency filtering. The leak rate of the cover was calculated to be less than 875 scfh adjusted to a 35 psig pressure. This flow rate is well below the SGTS flow rate of 2600 scfm which would ensure that any release from this path would be filtered and monitored.

To bound the potential for offsite releases for this event and plant conditions, a calculation was performed using the calculated leak rate. A total calculated primary containment leak rate of 931 SCFH was assumed (compared to the Technical Specification limit of 426 SCFH). It was further assumed that a design basis Loss of Coolant Accident had occurred. The result was that offsite thyroid dose consequences would have been approximately 66% of the Oyster Creek design basis accident, or approximately 32% of the 10 CFR 100 limit.

Although not required by design bases, a calculation had previously been performed which assumed that the Standby Gas Treatment System would not automatically start, and had to be manually started 30 minutes after a design basis accident. This calculation assumed that 100% of the core inventory of radioiodines was available for release, and that the only credit taken was for a 0.03 reduction for sprays, and a 0.4 reduction for plate out. Exfiltration from the reactor building was assumed to be directly connected to the source. Thus, no credit was taken for dilution in the reactor building. No credit was taken for source blowdown and scrubbing in the torus. No credit was taken for time dependent release conditions. (It is highly improbable that the full extent of assumed fuel damage could occur with the initial blowdown to the drywell, therefore, the initial pressure spike should not have been associated with the full source term.) No credit was taken for the subsequent pressure decrease caused by blowdown into the drywell nor was credit taken for pressure decrease caused by quenching in the suppression pool. A drywell leak rate of 0.5% and a source term of four times the Regulatory Guide 1.3 requirements were assumed.

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ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT (Con't)

Additionally, it was assumed that the Standby Gas Treatment System heaters failed for 1.5 hours after the 30 minute delay in initiation. This calculation was modified to change the drywell leak rate from the original 0.5% to the calculated leak rate for this event. The offsite dose rose from 32% of the 10 CFR 100 limits, to 33% of the 10 CFR 100 limits. This value was also well within the Oyster Creek specific design basis accident limits.

Reactor building radiation levels were normal during the entire time containment integrity was required.

CORRECTIVE ACTIONS

IMMEDIATE ACTIONS

A containment walkdown was performed and the leak was identified.

SHORT TERM ACTIONS

Valve, V-26-5 O rings were replaced and the valve was properly reassembled. A subsequent local leak rate test was performed with satisfactory results.

Primary containment leakage was calculated to be within Technical Specification limits upon subsequent containment pressurization utilizing the same procedure which identified the V-26-5 cover leakage.

LONG TERM ACTIONS

The specific human performance error of this event, along with others selected from 1996, was discussed by upper management with maintenance supervisory and craft personnel.

The procedure for limit switch calibration and mechanical surveillance was revised to emphasize self checking to ensure that the vacuum breaker is properly reassembled during the torquing sequence.

An evaluation of the methodology for performing LLRTs on the torus to drywell vacuum breakers will be done to determine if changes would be appropriate.

SIMILAR EVENTS

LER 94-022; December 19, 1994, Primary Containment Leak Rate in Excess of Technical Specification Requirements Due to Maintenance Procedure Non-compliance Sequence of Events.