GPU Nuclear

GPU Nuclear
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Writer's Direct Dial Number:

May 20, 1982

Mr. Ronald C. Haynes, Administrator
Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Haynes:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report
Reportable Occurrence No. 50-219/82-14/01T

This letter forwards three copies of a Licensee Event Report to report Reportable Occurrence No. 50-219/82-14/01T in compliance with paragraph 6.9.2.a.3 of the Technical Specifications.

The reporting of failures to meet acceptance criteria discovered during the performance of the recently completed local leak rate testing program has been accomplished through the submittal of this and two previous Licensee Event Reports (Reportable Occurrence Nos. 82-19 and 82-20). In order to provide unified and cohesive reporting of future primary containment degradation, discovered during the performance of local leak rate tests, the first test failure will be reported as a Licensee Event Report in accordance with paragraph 6.9.2.a.3 of the Technical Specifications. Subsequent degradations discovered will be included as an update to the initial report.

Very truly yours,

P. B. Fiedler

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:PFC:lse

Attachments

IE22 1/1

cc: Director (40 copies)
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Director (3)
Office of Management Information and
Program Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, NJ 08731

OYSTER CREEK NUCLEAR GENERATING STATION
Forked River, New Jersey 08731

Licensee Event Report
Reportable Occurrence No. 50-219/82-14/01T

Report Date

May 20, 1982

Occurrence Date

February 6, 1982

Identification of Occurrence

The results of local leak rate testing identified five containment isolation valves and one gasket that failed to meet their acceptance criteria. This constitutes an abnormal degradation of primary containment integrity.

This event is considered to be a reportable occurrence as defined in Technical Specifications, paragraph 6.9.2.a.3.

Conditions Prior to Occurrence

The plant was in cold shutdown with reactor coolant temperature less than 212°F and the reactor vented.

Description of Occurrence

Local leak rate testing identified the following valves with leakage in excess of the acceptance criteria of 11.9 SCFH @ 20 psig. The results of the leak rate test program for those valves are as follows:

Description	Valve/Gasket	Date of Test	Leakage @ 20 psig
1-8 Sump Discharge	V-22-28, 29	2-12-82	14.72 SCFH
Drywell Purge	V-27-3	2-6-82	158.70 SCFH
Drywell Airlock	Outer Door Gasket	3-31-82	Would not pressurize
Torus to Reactor Bldg. Vacuum Breaker	V-26-16	2-7-82	Would not pressurize
Torus Vent Bypass	V-28-47	2-9-82	Would not pressurize

Apparent Cause of Occurrence

Inspection of the 1-8 sump discharge valve (V-22-28, 29) seats showed they had degraded during operation. The Drywell Purge Valve (V-27-3) and Torus Vent Bypass Valve (V-28-47) experienced minor shift in the linkage adjustment setting during operation. The Drywell Airlock outer door gasket was damaged while moving equipment through the Airlock doors during current outage Main Steam Isolation Valve maintenance. Inspection of the Torus to Reactor Building vacuum breaker revealed misalignment of the operator/butterfly disc linkage (See Reportable Occurrence No. 82-12).

Analysis of Occurrence

Each of the containment penetrations tested had at least one redundant valve that met the acceptance criteria. In the case of 1-8 sump discharge valves, the leak rate test is performed by pressurizing between the valves. It is possible that one of the two valves could have had a leakage greater than the 11.9 SCFH acceptance criteria. If this were the case, the other valve would have a maximum leakage rate of 2.82 SCFH or less. If both valves had exactly the same leakage rate, the maximum leakage for the penetration would have been 7.36 SCFH.

Therefore, the safety significance of these events is considered to be minimal.

Corrective Action

The seats of the 1-8 sump discharge valves were lapped, blue checked, and passed their subsequent leak rate test. Drywell Purge Valve (V-27-3) and Torus Vent Bypass Valve (V-28-47) linkages were adjusted and the valves passed their subsequent leak rate tests. The Drywell Airlock outboard door gasket was replaced and it passed the subsequent leak rate test. The Torus to Reactor Building vacuum breaker linkage was realigned and the vacuum breaker valve passed the subsequent leak rate test.

The Drywell Purge Valves and operators are scheduled for replacement during the Cycle 11 refueling outage.