GPU Nuclear

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April 16, 1982

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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. 50-219/82-20/01T

This letter forwards three copies of a Licensee Event Report to report Reportable Occurrence No. 50-219/82-20/01T in compliance with paragraph 6.9.2.a.3 of the Technical Specifications.

Very truly yours,

PBFiedler

Peter B. Fiedler
Vice President & Director
Oyster Creek

PBF/kdk
Enclosures

cc: Director (40)
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 (1)
Oyster Creek Nuclear Generating Station
Forked River, N. J. 08731

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OYSTER CREEK NUCLEAR GENERATING STATION
Forked River, New Jersey 08731

Licensee Event Report
Reportable Occurrence No. 50-219/82-20/01T

Report Date

April 16, 1982

Occurrence Date

April 1, 1982

Identification of Occurrence

It was identified that an abnormal degradation of the primary containment existed based on the results of leak rate testing performed on the Main Steam Isolation Valves NS03A and NS04A.

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

Conditions Prior to Occurrence

The reactor was in cold shutdown condition at the time the occurrence was identified, with the reactor coolant temperature less than 212°F and the reactor vented. The reactor was in various operating modes since the last surveillance was performed satisfactorily.

Description of Occurrence

On February 8, 1982, while performing local leak rate testing on Main Steam Isolation Valves, the leak rate for valve NS03A was found to be 100 SCFH, which was greater than the acceptable limit of 11.9 SCFH. On March 28, 1982, the other Main Steam Isolation Valve in the "A" line, NS04A, was leak tested and resulted in a measured leakage of 22.9 SCFH, due to a packing leak. In the "A" line piping configuration, a packing leak in the NS04A and excessive leakage through NS03A provides an abnormal flow path from the reactor vessel to the trunnion room, which is considered part of Secondary Containment.

Apparent Cause of Occurrence

NS03A: The cause of the excessive valve leakage is attributed to a valve poppet pad wearing on the lower valve body rib. The body rib, which acts as an alignment guide for the valve poppet, prevented the poppet from properly positioning itself into the valve seat during closing. The cause of the valve deterioration is not fully understood at this time and will be the subject of an engineering evaluation.

Apparent Cause of Occurrence (Continued)

NS04A: The cause is attributed to packing leakage.

Analysis of Occurrence

As indicated in the Apparent Cause of Occurrence section, the leakage path for NS04A was through the valve packing. Because NS04A is downstream of NS03A, the leakage rate from primary containment was limited to 22.9 SCFH and confined within the trunnion room (part of Secondary Containment). The trunnion room is closed and sealed during power operation to maintain secondary containment. During power operation there were no indications of abnormal trunnion room temperatures which confirms the leakage to secondary containment from NS04A was minimal. Based on the above, the significance of this occurrence is considered minimal.

Corrective Action

Valve NS03A was completely disassembled and inspected both visually and dimensionally. A new valve poppet and poppet pad was installed in the valve and a stellite overlay weld repair was made to the worn section of the valve body rib. In addition, the valve stem and lantern ring were replaced due to indications of wear. The cause of NS03A valve deterioration will be evaluated and corrective actions taken as deemed necessary.

Valve NS04A was repacked.

After repairs were completed, both NS03A and NS04A were retested satisfactorily.

The present acceptance criteria of 11.9 SCFH appears to be overly conservative for a Main Steam Isolation Valve and is currently being evaluated. A Technical Specification change request will be submitted, if the evaluation warrants one. 10 CFR 50, Appendix J does not require a leakage limit on individual isolation valves.