



**GPU Nuclear**

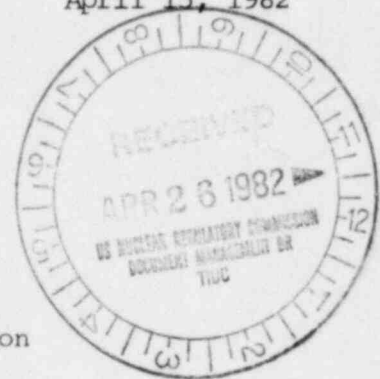
P.O. Box 388  
Forked River, New Jersey 08731  
609-693-6000  
Writer's Direct Dial Number:

April 15, 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. 50-219/82-19/01T



This letter forwards three copies of a Licensee Event Report to report Reportable Occurrence No. 50-219/82-19/01T in compliance with paragraph 6.9.2.a.3 of the Technical Specifications.

Very truly yours,

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. 50-219/82-19/01T

Report Date

April 15, 1982

Occurrence Date

March 31, 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 drain valves V-1-106, V-1-107 and V-1-110, V-1-111.

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 the cold shutdown condition at the time the occurrence was identified. However, the reactor was in various modes between the time period of January, 1980 until the date of identification of the event.

Description of Occurrence

On February 8, 1982, while performing local leak rate tests on Main Steam isolation drain valves V-1-106 and V-1-107 the results were found to be outside of the acceptable limit provided in the test procedure. On March 18, 1982, the remaining pair of Main Steam isolation drain valves, V-1-110 and V-1-111, were leak tested, and the results of this test were also outside of acceptable limits. Since V-1-110 and V-1-111 were tested as a pair, it could not be determined which valve was leaking. An additional test was performed and valve V-1-111 was determined to be the leaking valve.

In the piping configuration containing these valves, a leak in either V-1-106 or V-1-107 and a leak in V-1-111 provides an abnormal flow path from the reactor vessel to the Condenser hotwells.

Apparent Cause of Occurrence

The cause of the leakage was deterioration of valve internals.

### Analysis of Occurrence

Isolation valves V-1-110 and V-1-111 were tested as a pair with a resulting total leakage past these valves in excess of 100 SCFH. (The actual total leakage was not measurable due to the inability of the testing equipment to measure leakage greater than 100 SCFH.) The total leakage past isolation valves V-1-106 and V-1-107 when leak rate tested as a test pair was 49 SCFH. Therefore, the total leak past the four isolation valves in series or the primary containment penetration was 49.0 SCFH.

If a LOCA had occurred during the event period, leakage from this primary containment penetration would be directed to the condenser "A" hotwell due to the piping configuration associated with these penetration isolation valves. At the condenser, primary containment steam leakage would be condensed, mixed with, and contained in the condensate water system. Any primary containment non-condensable gas leakage would be contained in the condenser and off-gas systems.

### Corrective Action

Valve V-1-106 was repaired and passed the subsequent leak rate test. Valve V-1-111 was replaced and leak tested successfully. A further investigation into the suitability of these valves will be performed and additional corrective action taken, if required.