ENCLOSURE NO. 1

Description of Violation

Jersey Central Power and Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
Docket No. 50-219
License No. DPR-16

One activity under your license appears to be in violation of AEC requirements as indicated below. This apparent violation is considered to be of Category II severity.

Paragraph 6.6.2 of the Technical Specification requires that you notify the Director of Regional Regulatory Operations Office in the event of an abnormal occurrence and that this notification be made by telephone and telegraph within 24 hours of your recognition of the unusual occurrence. It also requires that you submit a written report of the occurrence to the Director of Licensing within 10 days. An abnormal occurrence is defined, in Section 1.15 of the Technical Specification, as a failure of one or more components of an engineered safety feature or plant protection system that causes or threatenes to cause the feature or system to be incapable of performing its intended function.

Contrary to this requirement, you failed to notify the Director of the Regional Regulatory Operations Office, or report to the Director of Licensing, within the prescribed time limits, that 88 of 132 shock suppressors had been found defective between April 15 and June 5, 1973. Again on July 22, 1973 you failed to make timely notification and to submit a timely report when you discovered that 8 of the reconditioned shock suppressors had again been found to be defective. We note that these matters were ultimately reported to the Directorate of Licensing in your letter dated August 6, 1973.

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PDR FOIA
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Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N.J. 07960 • 539-6111

October 12, 1973

RECEIVED OCT 17 1973

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Main Steam Isolation Valve Failure

The purpose of this letter is to report a violation of Technical Specifications, paragraph 4.5.F.1.D., failure of main steam isolation valves NS04A and NS04B to meet acceptable leakage rate requirements. This event is also considered to be an abnormal occurrence as defined in the Technical Specifications, parag[illegible] 1.15.E. Notification of this failure, as required by the Technical Spec[illegible]ations, was made to AEC Region I, Directorate of Regulatory Operations, on Friday, September 28, 1973 and by telecopier on that same day.

Following completion of extensive maintenance and repair work on main steam isolation valve NS03B, it was possible to conduct a leakage test on the two main steam isolation valves (NS04A and NS04B) outside the drywell. As a result of the ensueing test, main steam isolation valve NS04B leakage rate was determined to be 15.2 SCFH and isolation valve NS04A leakage rate measured 96 SCFH.

It was necessary to operate both valves in order to provide adequate ventilation of the reactor vessel while performing maintenance work on NS03B. Thus, the outside isolation valves were operated after the plant was shut down and were not tested in the "as found" condition, as is normally the case. The air flow path established by utilizing the main steam lines was successful in minimizing the radiogas concentrations in the drywell; thereby, providing maximum radiological protection for maintenance people while repairing NS03B.

Investigation into the cause of leakage through the two outside isolation valves resulted in identifying the valve stem packing as the leakage path. Replacement of the packing and subsequent retesting of the valves indicated essentially all the leakage associated with NS04A and NS04B was through the stem packing region.

As an additional precautionary measure, the two inside main steam isolation valves were also repacked.

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In our letter dated September 21, 1973, we indicated that the safety significance of the failure of NS03B depended on the condition of the outside valve in the "B" steam line, i.e., NS04B. With the failure of NS04B to pass a leak rate test, neither valve in the "B" steam line was capable of satisfying the Technical Specifications leakage rate limit of 9.95 SCFH.

It should be recognized the leakage through the packing of NS04B was equal to 5.7% of the total allowable primary containment leakage; whereas, the allowable Technical Specifications leakage from any one penetration or isolation valve is 5% of this total allowable leakage from the primary containment. Therefore, in the event of a LOCA, release of fission products from the primary containment would not be greater than the release discussed in Table I.5-2 of Amendment 65 and Section 3.3 of Amendemnt 68 in the FDSAR.

It should be noted that the leakage through the packing of NS04B would, for the most part, be drawn into the reactor building ventilation system and released through the plant stack. The failure of NS04A represents a failure of one of two redundant valves in the main steam line "A". Leakage through the packing of NS04B would also be into the reactor building ventilation system and would still be a controlled release under hypothetical accident conditions.

It should be noted that this is the first time significant stem packing leakage existed in the outer main steam isolation valves.

Based on past experience with the main steam isolation valves, a failure of this nature has not been previously experienced. Considering this, we intend to investigate a preventative maintenance program whereby a schedule of complete main steam isolation valve inspection can be accomplished. A set frequency will be determined for this inspection in order that all four (4) main steam isolation valves be checked within a reasonable time period. This program should preclude future failures of this kind by identifying problems prior to their reaching a point where degradation of valve integrity occurs.

Enclosed are forty (40) copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

DAR:cs
Enclosures

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I

JERSEY CENTRAL POWER LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

PHONE 609 • 693-6051

P.O. BOX 388 • FORKED RIVER • NEW JERSEY • 08731

October 11, 1973

Mr. James P. O'Reilly
Directorate of Regulatory Operations
Region 1
U. S. Atomic Energy Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. O'Reilly:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Preliminary Abnormal Occurrence Report No. 73-26

Per a telephone conversation between J. L. Sullivan, Jr., and D. Caphton on October 10, 1973, we are reporting the attached event as an abnormal occurrence, although it is not clear that it is reportable.

Technical Specification 4.1, Table 4.1.1, Note 2, states, "At least daily during reactor power operation, the reactor neutron flux peaking factor shall be estimated and the flow-referenced APRM scram and rod block settings shall be adjusted, if necessary, as specified in Section 2.3, Specifications (1) (a) and (2) (a)." This estimate was, in fact, performed as specified and corrections were made as required.

Very truly yours,

J. L. Sullivan, Jr.
FOR
J. T. Carroll, Jr.
Station Superintendent

JTC/pd

cc: A. Giambusso

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Date: 10/6/73
Time: 2:00 p.m.

Preliminary
Abnormal Occurrence
Report No. 73-26

SUBJECT: Failure to set the Average Power Range Monitor scram and rod block set points to the conservative values specified in Technical Specifications 2.3(1)(a) and 2.3(2)(a).

This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15A. Notification of this event, as required by the Technical Specifications, paragraph 6.6.2.a, was made to AEC Region I, Directorate of Regulatory Operations by telephone on October 10, 1973, at 4:30 p.m., and by telecopier on October 11, 1973, at 1:15 p.m.

SITUATION: On October 6, 1973, at 2:00 p.m., the reactor startup to full power had been halted due to a lack of in-service condensate demineralizers. The core thermal output at this time was approximately 567 MWt and the recirculation flow rate was $30X10^6$ lbm/hr. At this time the Maximum Total Peaking Factor (PF) was estimated to be 4.54 and the Average Power Range Monitors (APRM's) were set conservatively such that 100% on the APRM's corresponded to 1200 MWt. This is equivalent to reducing the neutron flux scram by the amount 3.01/PF as specified in Technical Specification 2.3.1.a, with some added margin. The 100%/1200 MWt setting allows for a neutron flux peaking up to a value of 4.84.

SITUATION - Continued

At 5:30 p.m., after a heat balance calculation, the setting of ~~the APRM's was inadvertently set such that 100% of the APRM's~~
the APRMs was inadvertently set such that 100% of the APRMs
corresponded to 1400 MWt which accounts for peaking factors of only 4.15. Thus, the limiting safety system setting for the APRM Neutron Flux Scram and rod block were set less conservatively than specified in the Technical Specification 2.3.1.a and 2.3.2.a.

CAUSE: An investigation is yet to be conducted to determine the exact cause of this occurrence. However, at this time, it is believed that it was caused by a communication problem.

REMEDIAL ACTION:

At 10:00 a.m. on October 7, 1973, the reactor neutron flux peaking factor was estimated as required in Technical Specification 4.1, Table 4.1.1, Note 2, and found to be 4.71. The APRM's were then correctly adjusted to the conservative 100%/1200 MWt setting.

SAFETY SIGNIFICANCE:

Based on the Neutron Flux Peaking Factor of 4.71, as estimated at the time of the correction, the safety limit can be shown to be at 1228 MWt for the recirculation flow rate of $30X10^6$ lbm/hr. Using the 100%/1400 MWt setting of the APRM's, the reactor at this condition would have scrammed at 1200 MWt, if required. Thus, the safety limit would not have been exceeded.

R. E. Ferguson, Jr.

Prepared by: [signature] Date: 10/11/73