Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

MEMBER OF THE General GPU ELECTRIC POWER COMPANIES SYSTEM Public Utilities Corporation

June 12, 1974

Mr. Robert T. Carlson, Chief
Facility Construction and Engineering, Support Branch
United States Atomic Energy Commission
Directorate of Regulatory Operations, Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Carlson:

Subject: Oyster Creek Station
Docket No. 50-219
RO Inspection Report No. 50-219/74-8

This is in reply to your letter of May 16, 1974 to Mr. I. R. Finfrock, Jr. regarding the inspection conducted by Mr. Walton on May 6-7, 1974 at our Oyster Creek Nuclear Generating Station.

1. AEC Concern

Criterion IX, Appendix B, 10 CFR 50 states, in part, "Measures shall be established to assure that special processes, including...nondestructive testing, are controlled and accomplished...using qualified procedures..."

Contrary to the above, nondestructive test procedures were approved by the licensee for use during in-service inspection with words in the procedure which state, "This procedure is not applicable and shall not be used for in-service inspection."

JCP&L Reply

The referenced procedure is a Magnaflux Testing Laboratory (Magnaflux) procedure and is not normally utilized for in-service inspection. The procedure is written to satisfy the requirements of Sections III, V, and VIII of the ASME Boiler and Pressure Vessel Code which are the sections invoked for in-service inspection at Oyster Creek.

June 12, 1974

Prior to its use, Magnaflux Procedure 13N, as well as all other applicable procedures, was reviewed by an independent JCP&L ASNT-TC-1A Certified Level III Representative and was found to be acceptable for in-service inspection at Oyster Creek. During the above review, JCP&L recognized the referenced statement, but it was determined that the procedure was adequate. The fact that the statement was not deleted from the procedure is an administrative oversight and JCP&L has since received Amendment No. 1 to Procedure 13N from Magnaflux which deletes Paragraph 13.2.2 Note 1.

2. AEC Concern

Amendment 68 to "Application for Reactor Construction Permit and Operating License", Paragraph 4.2.4, Criterion 32, states, in part, "...All piping, pumps, and valves defined by ASME Boiler and Pressure Vessel Code Section XI January 1, 1970 issue, to be part of the primary coolant pressure boundary and all components of the reactor vessel...are accessible for inspection during refueling."

ASME Section XI 1S 2.3.2 states, in part, ..."Ultrasonic examination, using the pulse-echo method, shall be conducted in accordance with the provisions of Appendix IX, IX-300 and referenced in IX 340..."

Section III, Paragraph IX-343(b) of the ASME Code states, in part, "...Drilled holes shall be used as basic calibration reflectors...these holes shall be located either in the production material or in a basic calibration block..." Paragraph IX-343(c) of the ASME Code states, "...In lieu of the above, other calibration reflectors are permitted, provided equivalent response is demonstrated..."

Contrary to the above, a calibration block was used which was not fabricated in accordance with the above requirements.

JCP&L Reply

The significance of your reference to Amendment 68 in this case is unclear. JCP&L does use the pulse-echo method for in-service inspection and does have and use calibration blocks for equipment calibration. The calibration block, which is used for the reactor vessel skirt weld inspection, was not available initially so Magnaflux provided a suitable block for calibrating their equipment. The Magnaflux 11W calibration block had a known 0.060 diameter side-drilled hole. The UT unit was calibrated using a 20 db factor while the actual weld was examined using 10 db, which we considered to be a conservative approach. The examination was conducted with the understanding that if there were any reflections noted, it would not be possible to correlate the results with previous examination results. However, since the examination did not reveal any reflections in either the weld zone or the base metal, there was no need for correlation. In any event, the Magnaflux IIW calibration block is in accordance with the required codes and the equipment calibration method was adequate and correct.

To preclude any uncertainty, the JCP&L calibration standard was located and the UT examination of the reactor vessel skirt was redone after recalibrating the equipment to the JCP&L standard. No defects were found and the results of both UT examinations are part of the 1974 in-service inspection records.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

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31 MAY 1974

H. D. Thornburg, Chief
Field Support and Enforcement Branch
Directorate of Regulatory Operations, HQ

JERSEY CENTRAL POWER AND LIGHT COMPANY
OYSTER CREEK - BWR
ABNORMAL OCCURRENCE AO 74-34

The subject abnormal occurrence report is forwarded for action.

Based on our review of the licensee's preliminary report, it is recommended that RO Headquarters transfer the lead responsibility to DL for review and evaluation of this penetration failure involving primary coolant leakage at the reactor vessel's bottom head. Additionally, DL should be requested to evaluate potential generic aspects of this problem.

RO:I will continue to provide field follow up and review of any forthcoming corrective measures by the licensee in this matter.

E. J. Brunner, Chief
Reactor Operations Branch

Enclosure
AO 74-34

Greenman	Caphton	Brunner	Tillou	O'Reilly
5/30/74	5/30		5-30-74	5/31

B/527

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To: James P. O'Reilly
Directorate of Regulatory Operations
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

From: Jersey Central Power & Light Company
Oyster Creek Nuclear Generating Station, Docket #50-219
Forked River, New Jersey 08731

Subject: Abnormal Occurrence Report No. 50-219/74/34

The following is a preliminary report being submitted in compliance with the Technical Specifications paragraph 6.6.2.

Preliminary Approval:

J T Carroll Jr. 5/30/74

J. T. Carroll, Jr. Date

cc: Mr. A. Giambusso

~~8004080489~~ fp

Report Date: 5/30/74 Occurrence: ______

OYSTER CREEK NUCLEAR GENERATING STATION
FORKED RIVER, NEW JERSEY 08731

Abnormal Occurrence
Report No. 50-219/74/ 34

IDENTIFICATION OF OCCURRENCE:

Violation of the Technical Specifications, paragraph N/A, Indications of coolant leakage existing in the area of an incore flux monitor reactor vessel housing located at core coordinate 28-05.

This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15E.

CONDITIONS PRIOR TO OCCURRENCE:

___ Steady State Power
___ Hot Standby
___ Cold Shutdown
_x_ Refueling Shutdown
___ Routine Startup Operation
___ Routine Shutdown Operation
___ Load Changes During Routine Power Operation
___ Other (Specify) ______

The reactor was in the REFUEL mode during a hydrostatic test at 850 psig pressure and with coolant temperature approximately 155°F.

DESCRIPTION OF OCCURRENCE:

On Tuesday, May 28, 1974, during a scheduled reactor vessel hydrostatic test to inspect the pressure boundary following refueling maintenance activities, leakage was observed in the vicinity of an incore flux monitor tube located at the bottom of the reactor vessel. Further investigation conducted on Wednesday, May 29, 1974, showed evidence of possible leakage in the area of an incore flux monitor housing penetration located in the reactor vessel bottom head. A second hydrostatic test was conducted at a pressure of 850 psig at approximately 7:00 p.m. on May 29, 1974, whereupon, water was observed leaking between the monitor housing and the reactor vessel. The leakage was measured under the conditions of 850 psig with a temperature of 164°F, and calculated to be on the order of approximately 0.02 gallons per hour.

APPARENT CAUSE OF OCCURRENCE:

____ Design	____ Procedure
____ Manufacture	____ Unusual Service Condition Inc. Environmental
____ Installation/ Construction	____ Component Failure
____ Operator	____ Other (Specify) ____

The cause of this event has yet to be determined.

ANALYSIS OF OCCURRENCE:

As stated in FDSAR Amendment #37, a postulated failure of the flux monitor tube would result in vessel leakage at a rate which would not cause excessive cladding temperatures and for which core reflooding is possible by engineered safety features. This situation is less severe than the design basis accident. To determine the consequences of a weld failure at a housing for an in-core monitor tube, it is assumed that the weld between the housing and the reactor vessel bottom head fails, allowing the housing and the in-core monitor tube to be ejected from the vessel. The hole provided in the bottom head for the housing has a diameter of two inches; this is the assumed break size. The hole has a break area of .0218 ft^2. Assuming worst conditions, this results in peak clad temperatures less than 1000°F, as updated in FDSAR Amendment #67. This value is well within acceptable limits of the applicable ECCS criteria.

CORRECTIVE ACTION: The nuclear steam supply vendor and the reactor vessel manufacturer have been contacted with regard to this condition. Discussion will ensue as to the proper course of action to be taken to resolve this matter. Recommendations will be forthcoming pending complete review of this event by the Plant Operations Review Committee.

Prepared by: John P. Sullivan Jr. Date: 5/30/74