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SHIELDS L. DALTROFF
VICE PRESIDENT
ELECTRIC PRODUCTION

February 21, 1985

Docket No. 50-352

Mr. Richard W. Starostecki, Director
Division of Project and Resident Programs
U.S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Starostecki:

Your letter of January 22, 1985, forwarded Inspection Report No. 50-352/85-01 which cited an activity at our Limerick Generating Station which did not appear to be in full compliance with NRC requirements and a deviation from an FSAR commitment. The event which identified this apparent noncompliance was reported to the NRC in Licensee Event Report No. 84-042 dated January 30, 1985. The apparent violation and deviation are restated below followed by our response.

Violation

10 CFR 50, Appendix B, Criterion V requires that activities affecting quality to be performed in accordance with appropriate procedures or drawings.

Electrical Engineering Division Procedure EE 11.11 requires that inspections and tests be performed to determine that control circuit contact development is in accordance with appropriate drawings and authorizes the implementation of Rework Notices to correct nonconforming contact development conditions.

Quality Control Inspection Instruction 002 requires that inspections of wire connections made during the implementation of Field Engineering Rework Notices be made using the applicable project drawings as the acceptance criterion.

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Contrary to the above, between April 23 and April 30, 1984, testing, rework and inspection of the contact development shown on drawing E-519 were not performed using the appropriate drawing in that these activities were performed using revision 2 of drawing E-519 without first resolving the effects of General Electric FDDR HH1-2835 on that revision of the drawing.

This is a Severity Level IV violation (Supplement II).

Response

On December 31, 1984, a surveillance test was being performed on the "C" Refueling Floor Exhaust Duct Radiation Monitor. The monitor failed to pass the surveillance test, was declared inoperable, and the channel was placed in the "trip" condition in accordance with Technical Specification 3.3.2-1. During verification by the control room operators that the affected system isolation valves were closed, it was discovered that two of the valves, SV-026-190A and SV-026-190C, had not closed when the monitor was placed in the tripped condition. The operators manually closed the valves to isolate the affected containment penetration. Additionally, the two valves which were redundant to the inoperable ones were verified as operable. Likewise, all other isolation valves which are operated by the Refueling Floor Exhaust Duct Radiation system were verified as operable.

An immediate investigation indicated that the condition of the wiring for the valve closure circuits did not agree with design drawing E-519, Revision 3, in that the closure circuits were connected to a contact controlled by relay B21H-K120C instead of a contact from the proper relay B21H-K101C. The valve closure control circuits were restored to the configuration described by E-519, Revision 3, and the "C" Refueling Floor Exhaust Duct Radiation Monitor was returned to service by 7:00 p.m. on December 31, 1984.

The circuit connections for SV-026-190A and SV-026-190C did not agree with the current drawing E-519, Revision 3, as the result of personnel error during preoperational testing. A PECo field engineer mistakenly implemented a change to the circuit during initial checkout testing of the circuits on drawing E-519.

Earlier revisions to E-519 required circuit connections for SV-026-190A and SV-026-190C to relay B21H-K120C. Between December, 1983 and January, 1984, Design Change Packages (DCP) 176 and 186 and General Electric (GE) Field Deviation Disposition Request (FDDR) HH1-2835 were approved to change E-519 circuits in conjunction with the implementation of ATWS modifications.

As a result of the DCP's and GE FDDR's, the circuits on E-519 were changed in early 1984 such that the closure of SV-026-190A and SV-026-190C would be controlled by relay B21H-K101C. Because of the broad scope of the DCP's and FDDR, the final closure of the design changes did not occur until July 1984. As a result, all the affected drawings were not revised until September 1984.

In the meantime, circuits shown on E-519 were Blue Tag tested by PECo field engineering in April, 1984. The PECo field engineer involved with this testing used Revision 2 to E-519. This revision was the most current revision to the drawing at the time. However, a traveler form had been issued with Revision 2 to E-519 to indicate that GE FDDR HH-1-2835 affected that revision. This field engineer did not resolve the interface between the FDDR and the drawing before performing his testing. He discovered, during the testing, that the closure circuits for SV-026-190A and SV-026-190C were connected to relay B21H-K101C instead of relay B21H-K120C as shown on E-519, Revision 2. Therefore, he initiated a Rework Notice 79E-7 to permit a change to the circuits to connect them to relay B21H-K120C. Rework Notice 79E-7 was approved and implemented in April, 1984. Subsequent inspection of the work and testing of the system were completed without identifying this error which reversed the intended modification.

When the cause of this event was identified, a three-part investigation was initiated to verify operability of primary and secondary containment isolation valves. This investigation consisted of:

1. A Rework Notice review by personnel from the PECo Electrical Engineering Division Field Engineers.

2. An independent review of Rework Notices by the PECo Engineering & Research Department QA Division.

3. A review of documentation for testing of primary and secondary containment isolation valves by members of the plant technical staff.

The PECo Field Engineering group performed a detailed review of the Field Engineering Rework Notices written to date. No further hardware related discrepancies were noted as a result of this review.

Engineering and Research QA personnel performed a detailed review of approximately ten (10) percent of the QC Inspection Reports written to date, relating to Field Engineering Rework Notices, and no further hardware related discrepancies were noted.

The plant technical staff reviewed documentation associated with testing of primary and secondary containment valves listed in Technical Specification Tables 3.6.3-1, 3.6.5.2.1-1, and 3.6.5.2.2-1 to verify containment isolation capability. Surveillance tests were used to verify containment isolation instrument operability. Surveillance tests, preoperational tests, and "Blue Tag " test records were used to verify that each containment isolation valve closes in response to each of its required isolation signals. No operability discrepancies were identified during the review.

Based upon the results of this three-part investigation it is our conclusion that the wiring error associated with these two valves was an isolated incident.

In order to prevent recurrence of this isolated incident, the Electrical Field Engineering group and the Quality Control Inspectors have been retrained emphasizing the procedural requirements for review of the latest design documents, including any additions or attachments, prior to performance of any work activities.

Deviation

B. Final Safety Analysis Report (FSAR) section 14.2.12 indicates that tests conducted during the preoperational test program are identified and described in the test abstracts contained in FSAR Table 14.2-4.

The test method description in the test abstract for preoperational test 1P59.1 for the Containment Isolation and Nuclear Steam Supply Shutoff System indicates that during the test, the actuator trip relay for each containment isolation valve would operate the corresponding isolation valve.

Contrary to the above, test 1P59.1 did not demonstrate that the actuator trip relays associated with the refueling area ventilation exhaust high radiation conditions operated each corresponding isolation valve in that the operation of valves SV-26-190A and 190C was not verified.

Response

The test abstract contained in FSAR Section 14.2.12, Table 14.2-4 for preoperational test 1P59.1, Containment Isolation and Nuclear Steam Supply Shutoff System, contains a unique general instruction requiring that each isolation signal be verified from actuator trip relay to the associated isolation valve. Although the test method utilized in the performance of 1P59.1 did not explicitly meet this requirement, the test was written and performed with the understanding and agreement of those groups involved in the performance and review of the preoperational test program to demonstrate that the logic of the system was proper and met the FSAR Acceptance Criteria for system isolation. Demonstration of valve operation from actuator relay was to be demonstrated during the prerequisite "Blue Tag" component testing, hence the stated test methodology would be met implicitly. Generally, this actuation was re-verified during the separate preoperational tests of those systems which contained the particular isolation valves.

A review of the preoperational tests for each of those systems which contain valves which receive containment isolation signals (including those shown on FSAR Table 6.2-17) has been performed to determine if all valves were verified from their individual actuating relays as part of the preoperational test. This review revealed two instances in which this was not the case. The first instance included valves SV-026-190A, B, C, D of which the A & C required rewiring and retesting. The retest of valves SV-026-190A & C is documented in a special partial surveillance test, ST-1-072-103, conducted 12/31/84. A review of the test records for the B&D valves indicates that they were satisfactorily tested as a preoperational test prerequisite. The second instance was for valve HV-059-131 which a review of the valve test record reveals to have also been satisfactorily tested as a prerequisite to preoperational testing.

Based on the results of our investigation, it is concluded that the testing of the SV-026-190A & C isolation valves was an isolated case and that the test methodology utilized successfully demonstrated the functionality of the

Containment Isolation and Nuclear Steam Supply Shutoff System.

Should you have any questions or require further information, please contact us.

Very truly yours,

cc: Dr. T. E. Murley, Administrator
Mr. J. T. Wiggins, Senior Resident Inspector
See Attached Service List

cc: Judge Helen F. Hoyt
Judge Jerry Harbour
Judge Richard F. Cole
Troy B. Conner, Jr., Esq.
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Atomic Safety & Licensing Appeal Board
Atomic Safety & Licensing Board Panel
Docket & Service Section (3 Copies)
James Wiggins
Timothy R. S. Campbell

January 16, 1985