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SHIELDS L. DALTROFF
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May 19, 1982

Docket Nos. 50-277
50-278

Insp. Report 50-277/82-06
50-278/82-06

Mr. Richard W. Starostecki, Director
United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Starostecki:

Your letter of April 22, 1982, forwarded combined Inspection Report 50-277/82-06 and 50-278/82-06. Appendix A addresses several items which do not appear to be in full compliance with Nuclear Regulatory Commission requirements. These items are restated below along with our response.

A.1 Technical Specification 6.8 and Regulatory Guide 1.33 (November 1972) require implementation of procedures for control of measuring and test equipment. Licensee Procedure M65.4, Revision 5, June 15, 1981, Hydraulic Snubber Testing, requires a daily velocity meter calibration check prior to snubber testing.

Contrary to the above, on March 17, 1982, two hydraulic snubbers were tested without the specified prior check of the velocity meter calibration. Further, a substitute check used was not a valid calibration check.

This is a Severity Level IV Violation (Supplement I) applicable to DPR-44.

Response

Personnel error and a delay in the revision of the procedure for Hydraulic Snubber testing, M-65.4 caused this occurrence. On March 12, 1982, an improved peak holding velocity indicator for the snubber test machine was installed and calibrated. This installation also incorporated components to semi-automate the calibration check of the new velocity indicator. Further investigation of this item has revealed that although the new velocity indicator calibration check was not yet included in the M-65.4 procedure, the job leader did perform a valid calibration check of the indicator. This calibration check correlated preset snubber machine velocity, as read on the indicator, to distance, one inch of ram travel and time, as read on a stopwatch. The correlation consisted of converting snubber test machine inches/second to inches/minute and comparing the result to the preset snubber machine velocity indicator readout. Full details of this calibration check may not have been clearly communicated at the time of the inspection activity. This calibration check, used by the job leader prior to testing the two subject snubbers on March 17, 1982, is similar to that which was incorporated into the Hydraulic Snubber Testing Procedure M-65.4 on March 24, 1982. Test data supporting the operability of the two subject snubbers was therefore not compromised. Further, the two snubbers were rechecked on April 24, 1982 and were found within the acceptance criteria specified.

As stated above, the Hydraulic Snubber Testing Procedure M-65.4 was revised on March 24, 1982, to incorporate the semi-automated velocity indicator calibration check. Additionally, the personnel involved with this occurrence were counseled on the importance of timely procedure revision and the inappropriateness of using unapproved test methods.

A.3 Technical Specification 6.8 and Regulatory Guide 1.33 (November 1972) Appendix A Section 1.5 require procedural control of maintenance and repair work, including actions to be taken into account to minimize radiation exposure to workers.

Contrary to the above, on April 1, 1982, with the reactor shutdown, procedural controls did not adequately safeguard against worker radiation exposure; the blocking control for repacking the open but not backseated High Pressure Coolant Injection System inboard steam line isolation valve was to maintain reactor water level below the main steam lines, and

an on-duty reactor operator was not aware of the isolation valve work or that a rising reactor water level could result in spilling or spraying of 150 degrees F radioactive reactor coolant on or near workers in the drywell.

This is a Severity Level IV Violation (Supplement I) applicable to DPR-56.

Response

During the period covered by this Inspection Report, the concern expressed by the NRC inspector, regarding the blocking of the HPCI inboard steam line isolation valve was discussed with the operations engineer and on two occasions with the station superintendent. These preliminary discussions did not identify the inspector's concern associated with this item as a possible violation until the meeting held on April 20, 1982 to discuss the summary of preliminary findings. Until this last meeting, station management was aware of the NRC inspector's concern, but did not believe that a violation was indicated by the facts provided.

The detailed Inspection Report concerned with this finding indicates the following:

"The valve, which is the first valve in-line from the reactor vessel, was tagged open (but not backseated) for re-packing. Thus, part of the blocking for maintenance involved keeping reactor water level below the main steam lines from which the HPCI steam line taps are taken. The permit included a pertinent administrative control--a statement, 'Reactor level being maintained below the main steam lines.' The inspector questioned the reliability of this control, since it did not provide assurance of continued awareness of this blocking point. For example, a Unit 3 reactor operator on-shift on April 1 had not been informed that, should reactor water level control be lost and approach the main steam line penetrations, the HPCI valve block would be jeopordized (sic), potentially resulting in a radiological hazard (from spilling or spraying 150 degrees F radioactive water) to workers in the drywell."

During the exit interview with the resident inspectors on April 20, 1982, the resident inspector indicated that the Unit 3 reactor operator on shift on April 1, was knowledgeable of the proper reactor water level to be maintained during that shift. He was, however, not able to provide the inspector with the specific reason for maintaining the reactor water level at the specified normal level.

The resident inspector also stated, and the detailed Inspection Report indicates, that the permit which provided the blocking for maintenance to repack the subject valve included a pertinent administrative control in that the permit required "reactor level being maintained below main steam lines." The violation being assessed in this case is stated as "Procedural control did not adequately safeguard against work radiation exposure"..."and an on duty reactor operator was not aware that a rising reactor water level could result in spilling or spraying 150 degrees F radioactive reactor coolant on or near workers in the drywell." The violation, therefore, appears to be based on the fact that there was inadequate safeguard provided by the blocking permit and the reactor operator did not know why he was to maintain the level at the specified value. The facts, however, as stated in the detailed report indicate that an administrative control was provided to maintain the reactor water at the proper level, and the reactor operator was knowledgeable of level to be maintained, and how that level could be maintained during his shift. Philadelphia Electric Company, therefore believes that this finding has insufficient grounds to justify a Severity Level IV violation, and requests that the NRC reevaluate this finding.

Philadelphia Electric Company, however, acknowledges that improvements in information transfer to the reactor operator can be provided. In order to improve the knowledge of the reactor operator concerning primary coolant system integrity, and pressure and temperature limitations, an addition to the shift turnover list will be provided. This addition will indicate the temperature, pressure and level limits to be maintained, and the reason why these values should be maintained during that operating period. These changes will be implemented by June 1, 1982.

A.4 Technical Specification 6.8 and Regulatory Guide 1.33 require implementation of procedures for equipment control. Licensee Procedure A-8, Revision 4, January 22, 1980, Procedure for Control of Locked Valves, and Appendix A-8A require that valves on the Locked Valve Lists be locked with a padlock and chain when no related maintenance is in progress and that, if a lock is removed, an entry be made in the "Locked Valve Log."

Contrary to the above, on April 2, 1982, the following three Emergency Service Water valves listed in Appendix A-8A were not locked and no related maintenance was in progress: 504C, Inlet to 'C' Diesel; 519C, Inlet to 'C' Diesel; and 505C, Outlet from 'C' Diesel. Further, no entries regarding these valves had been made when the 'C' Diesel was blocked for maintenance on March 14 - 17, 1982.

This is a Severity Level IV Violation (Supplement I) applicable to DPR-44 and DPR-56.

Response

Upon discovery of the three properly positioned, but unlocked valves, the shift superintendent had them locked and all similar valves on the other three Diesel Generators checked per the administrative procedure for Control of Locked Valves, A-8. The three valves in question were the only valves found to be unlocked. Although the three valves were listed in the Procedure For Control of Locked Valves, Appendix A-8A, they were not listed as locked valves on the Diesel Generator blocking permit. Therefore, when the Diesel Generator was unblocked after maintenance on March 17, 1982, the valves were opened but not locked. Several actions have been taken or are planned to prevent recurrence.

A Shift Night Order Sheet was issued to have all operators review the procedure for control of locked valves. The Diesel Generator blocking sequence and check off list have been revised to require the valves to be locked. Additionally, all blocking sequences which deal with locked valves will be revised to provide provisions in the blocking sequence to help ensure that valves which are required to be locked are again locked when the permit is cleared. This review will be completed by December 30, 1982.

A.5 Technical Specification 6.8 and Regulatory Guide 1.33 (November 1972) require implementation of written procedures for plant operations. Administrative Procedure A-7, Revision 17, May 18, 1981, Shift Operations, requires that shift operations be conducted in accordance with approved procedures. Reactor Core Isolation Cooling (RCIC) System Lineup Procedure C.O.L. S.3.5.A, Revision 10, May 30, 1978, requires that RCIC turbine exhaust drain line isolation valves, AO-137 and AO-138, be OPEN during operations. Approved RCIC procedures permit deviation from the prescribed OPEN position for surveillance or maintenance but require these valves to be returned to the OPEN position when the reactor is above 100 psi and no system maintenance or testing is in progress.

Contrary to the above, RCIC valves AO-137 and AO-138 were found SHUT at 8:20 AM, April 9, 1982, when the reactor was above 100 psi and no system maintenance or testing was in progress.

This is a Severity Level IV Violation (Supplement I) applicable to DPR-56.

Response

The plant staff has completed an in-depth investigation into the circumstances associated with the RCIC valves AO-137 and AO-138 being found in the closed condition at 8:20 AM on April 9, 1982. As noted in the Inspection Report, surveillance testing of the RCIC System had been completed at 9:30 PM on the previous day. This Surveillance Test was completed satisfactorily with the double valve verification check-off list completed by two qualified operators. Because of other equipment difficulties, primary system pressure was reduced below 105 psig at 10:40 PM on this date. This reduced pressure condition was maintained until approximately 4:30 AM on April 9, 1982. Shift personnel were questioned and stated that no isolation of the RCIC System occurred due to the reduced pressure condition. Review of as-left calibration data on the pressure switches which would initiate an isolation of these valves indicates that the pressure in the primary cooling system was not low enough to have initiated an isolation. In addition, while performing the shift turnover front panel checks at midnight, the STA's and reactor operators observed the valves to be open at that time. The reactor operator on shift and control operator on midnight shift also stated that no isolation signal was initiated on their shift (11:30 PM, April 8, 1982 - 7:30 AM, April 9, 1982. In addition, the shift superintendent observed the valves in an open position at 6:00 AM on April 9, 1982.

As indicated above, the equipment deficiencies which required maintenance of the reactor coolant pressure below 105 psig were resolved prior to 4:30 AM on April 9, 1982. Reactor pressure increase was initiated and was in progress at the 7:00 AM shift change on April 9, 1982. During an increase in reactor pressure from 400 psig through 950 psig, reactor level control and reactivity control requires detailed and continuous operator attention. Because of this start-up activity, the front panel checks for day shift were not performed prior to the NRC inspector's identification of the valves and their control switches in improper positions. Normally, front panel checks are performed by the oncoming shift and the STA within the first hour following shift change.

Based on investigation and the above discussion, it appears that the valves were closed for a maximum of two hours and twenty minutes (6:00 AM through 8:20 AM) and they were not placed in this position by automatic operation. Our investigation was not able to identify the mechanism by which the valves were closed from the front panel switches.

As indicated in the Inspection Report, the deviation from a normal valve lineup did not render the RCIC System inoperable. Upon identification of this improper valve lineup by the NRC inspector, the reactor operator immediately placed the valves in a proper position. Operation of these valves at this time, therefore, proved proper manual operability.

In order to better insure proper system valve lineups, control room personnel and STA's were requested to temporarily increase the frequency of their front panel checks. These checks were performed two and three times per shift. In addition, shift supervision routinely performed front panel checks on every shift. No other significant system misalignments were identified during the increased surveillance period.

It is believed that this event is an isolated occurrence. Administrative procedures now in effect require double verification of major valves following surveillance testing and return of equipment after maintenance. Front panel checks are required to be performed at the beginning of each shift by control room operators and STA's. This requirement has been re-emphasized to individuals responsible.

Based on our analysis of this as an isolated instance and the adequacy of the present administrative controls, no further action is deemed necessary.

If any additional information is required, please do not hesitate to contact us.

Very truly yours,

COMMONWEALTH OF PENNSYLVANIA :

ss.

COUNTY OF PHILADELPHIA :

S. L. Daltroff, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company, the Applicant herein; that he has read the foregoing response to Inspection Report No. 50-277/82-06 & 50-278/82-06 and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

S L Daltroff

Subscribed and sworn to before me this 20th day of May, 1982.

Patricia A Jones

Notary Public

PATRICIA A. JONES
Notary Public, Media Boro, Delaware Co.
My Commission Expires Sept. 13, 1982