U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Docket Nos.	50-352 50-353
License Nos.	NPF-39 NPF-85
Licensee:	Philadelphia Electric Company Correspondence Control Desk P.O. Box 195 Wayne, Pa 19087-0195
Facility Name:	Limerick Generating Station, Units 1 and 2
Inspection Period:	March 15, 1992 - April 25, 1992
Inspectors:	T. J. Kenny, Senior Resident Inspector L. L. Scholl, Resident Inspector H. J. Gregg, Senior Reactor Engineer

Approved by: ______________________ 5/21/92
Jeffrey J. Lyash, Acting Chief — Date
Reactor Projects Section No. 2B

Inspection Summary: This inspection report documents routine and reactive inspections during day and backshift hours of station activities including: plant operations; radiation protection; surveillance and maintenance; and safety assessment/quality verification.

EXECUTIVE SUMMARY
Limerick Generating Station
Report No. 92-11 & 92-11

Plant Operations

Plant operations continued to be good. Diligent monitoring of plant conditions resulted in the Unit 1 Reactor Operator promptly observing a loss of level in the reactor cavity due to a maintenance error. The prompt action limited the loss of water from the reactor cavity. (Section 1.0)

Surveillance and Maintenance

A maintenance technician caused a loss of inventory in the Unit 1 reactor cavity due to an error during local leak rate testing of containment isolation valves. A violation was cited for failure to follow the current procedure (Section 3.2).

Engineering and Technical Support

Design packages for plant modifications reviewed by the inspector were found to be very good, enabling their installation to be completed with minimal field changes. Several of these modifications resolve conditions that had prompted LER's in the past. (Section 4.0)

Radiological Protection

Several instances were identified where the ALARA impact of scaffolding construction was not adequately assessed by Philadelphia Electric Company (PECo). (Section 5.0)

Safety Assessment and Quality Verification

Maintenance errors resulting from inattention to detail and failure to adhere to station procedures continued to occur during this inspection period. Corrective actions for previous similar problems have not been fully effective. (Section 6.0)

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DETAILS

1.0 PLANT OPERATIONS (71707)[1]

The inspectors conducted routine entries into the protected areas of the plant, including the control room, reactor enclosure, fuel floor, and drywell (when access was possible). During the inspections, discussions were held with operators, health physics (HP) and instrument and control (I&C) technicians, mechanics, security personnel, supervisors and plant management. The inspections were conducted in accordance with NRC Inspection Procedure 71707 and evaluated the licensee's compliance with 10 CFR, Technical Specifications, License Conditions and Administrative Procedures.

1.1 Operational Overview

Unit 1

Unit 1 began this report period at 100 percent power. The unit was shutdown on March 20, 1992, to commence the fourth refueling outage.

Unit 2

Unit 2 operated at or near 100 percent power throughout this report period.

1.2 Reportable Events

Unit 1

Blowout Panel Opening

On March 21, 1992, at 7:34 p.m., the Unit 1 reactor enclosure (RE) heating, ventilation, and air conditioning (HVAC) system was turned off as per procedures SP-S-006, "De-energizing D13 Safeguards Bus," and S76.2.B, "Shutdown of Reactor Enclosure HVAC." As required by S76.2.B, there was an operator stationed at electrical panel 10C206. When the exhaust fans were turned off, the operator at panel 10C206 recognized that one supply fan did not trip. The operator was unable to restart the exhaust fan because the RE high delta pressure logic had tripped. As a result of the 'C' supply fan failing to trip, the RE experienced a positive pressure and caused the following.

- The blowout panel in the Unit 1 safeguards system access area blew out to relieve the pressure (blowout pressure is .25 pounds per square inch differential (psid)). This opened a vent path to atmosphere.

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[1]The NRC Inspection Procedures used as guidance are listed parenthetically throughout this report.

- At the time of the event the reactor seals were deflated and the shield plugs were removed as part of the reactor disassembly. The air flow went from the RE to the refuel floor and agitated the contamination in the reactor well, causing refuel floor airborne activities to reach 300-500k counts per second.

- After the operator tripped the 'C' supply fan the air flow reversed and contaminated elevation 313 in the RE.

Due to the reactor being in cold shutdown the operators deemed that an NRC notification was not necessary. The licensee replaced the blowout panel, and performed sampling that showed there was no release of contaminated material to the atmosphere. The licensee reevaluated the incident on March 23 and reported under 10 CFR 50.72(b)(2)(iii)(c), after concluding that it constituted a condition that could have prevented the RE from controling the release of radioactive material.

Fifteen workers on the refuel floor were slightly contaminated, none greater than regulatory limits. Minimum efforts were required to decontaminate the workers. Most were cleaned by removing the anti-contamination clothing, others required a shower. Overall, a total estimated skin dose of 2 mrem was received by four individuals with skin contamination. Followup whole body counts did not show internal contamination to any individual.

The supply fan did not trip because the associated breaker trip coil had failed. A team made up of representatives of system engineering, maintenance, operations, corporate engineering and a contractor specializing in breaker testing was formed to investigate the failure coil. The scope of their review included the following:

- The trip control and indication circuitry was reviewed, inspected and found to meet the design.

- Resistance readings of the trip control circuit logic were taken and found to meet specifications.

- The design of the supply fan trip control logic was reviewed and determined to be acceptable.

- Field testing of the trip control and indication circuit wiring was performed verifying that the breaker is correctly wired.

- Voltage readings were taken to investigate the possibility of a voltage dip at the trip coil, normal readings were found.

- The '1C' RE HVAC supply fan power supply breaker was sent to the manufacturer for failure mode analysis and investigation for generic concerns. The trip coil adjustment was verified to be correct, and after the failed trip coil was replaced, the suspect breaker was tripped approximately 20 times successfully. Inspection and testing did not reveal the specific cause of the failure nor any generic concerns.

- A transient analysis recorder was installed on the trip control circuit logic with the breaker racked in the test position. The breaker was tripped via the trip control logic and voltage and trip coil current draw were recorded. At the close of the inspection period this data was being analyzed.

The '1C' RE HVAC supply air fan power supply breaker was repaired by replacing the trip coil. The breaker was satisfactorily tested on April 6, 1992, and remains in an "Emergency Use Only" status to support continued troubleshooting. To prevent the possibility of RE overpressurization following failure of a supply fan breaker to trip, the following corrective actions were being taken:

- Procedure S76.2.B was temporarily revised on March 23, 1992, and is expected to be permanently revised by May 15, 1992. The revision includes addressing the concerns identified as a result of this event, and will remain in effect until the cause of the failure of the RE supply fans to trip has been determined and corrected.

- A modification to increase blowout panel setpoints had been planned prior to this event. The modification is being implemented during the current refueling outage for Unit 1, and will be implemented during the next refueling outage for Unit 2. See section 4 for further details.

To prevent future RE HVAC events from causing contamination of the refuel floor through the cavity seals, the following corrective actions were taken:

- The reactor well seals were inflated on March 22, 1991.

- Procedures S76.1.A, "Startup of Refuel Floor HVAC," S76.1.B, "Startup of Reactor Enclosure HVAC," S76.2.A, "Shutdown of Refuel Floor HVAC," and S76.2.B were revised to add an additional precaution to operators performing startup and shutdown of RE and refuel floor HVAC systems when the reactor shield plugs are removed. The cautions highlight the potential for contamination due to air flow via the deflated reactor well seals.

- PECo has committed to temporarily change the appropriate maintenance and operations procedures for reactor disassembly and reassembly prior to Unit 1 reactor reassembly, and to permanently change them prior to the next refueling outage. The revisions will add a step to ensure that the reactor well seals are inflated when the shield plugs are removed.

- A letter outlining this event has been issued to all personnel responsible for evaluating reportable events. The letter discussed the reporting requirements related to conditions that alone could have prevented the fulfillment of the safety functions of a system, even if found when the plant is in an operating condition where the system is not required. The letter will be included in the Licensed Operator Requalification training program by June 1, 1992.

The inspector noted that the Health Physics Department handling of the event was expeditious and thorough and that the area contaminated by the event was cleaned up within two shifts.

Containment Group 6A Isolation

On March 21, 1992, at 4:55 p.m., a group '6A' isolation was caused when a chemistry technician entered an improper high radiation set point into the Radiation and Meteorological Monitoring System (RMMS) computer. The setpoint entered was lower than the existing reading and caused the isolation. The technician was completing ST-5-057-810-0, "North Stack Containment Purge Sampling and Analysis," and was misled by an example in the procedure showing a negative exponent. The technician entered the negative exponent instead of the correct positive one. The following corrective steps have been taken by PECo:

- The technician was counseled regarding attention to detail and procedural requirements.

- Through memorandums and discussions all chemistry technicians were reminded of the need for diligent attention to detail while performing their duties.

- The procedure was revised to clarify the examples, and to incorporate cautions about the exponents.

- All station training programs were updated to emphasize the importance of procedural adherence and the lessons learned by the incident.

The inspector concluded that the licensee's corrective actions were adequate.

Retraction of an Engineered Safety Feature Report

On March 31, 1992, at 2:51 p.m., PECo notified the NRC via the Emergency Notification System (ENS) that an Engineered Safety Feature (ESF) actuation had taken place when the D13 safeguard bus feeder breaker from the 201 safeguard transformer tripped open on an undervoltage signal. The independent safeguard power feeder from the 101 safeguard transformer closed in automatically, as designed, to reenergize the D13 safeguard bus. The licensee traced the problem to a faulty undervoltage relay that was later replaced.

After further analysis, on April 6, 1992, PECo retracted the notification. The operation of the diesel generators (standby AC Power System) is the only ESF associated with the Class IE power system. Operation of the feeder breaker automatic transfer is a design feature for the reliability of the Class IE power system and not part of the ESF. The inspector reviewed the applicable portions of the Updated Final Safety Analysis report (UFSAR) and concluded that PECo was correct in the final analysis.

Retraction of an Inoperable Radiation Monitor Report

On April 12, 1992, at [illegible]:00 a.m., PECo notified the NRC that the residual heat removal service water (RHRSW) radiation monitors were inoperable because of higher than required setpoints. The setpoints were changed in error by a technician who inadvertently installed 'A' loop set points into the 'B' loop. The setpoints are based on backround radiation levels and the 'A' loop has the higher backround. After further analysis PECo determined that although the higher numbers were entered into 'B' loop, the loop would have tripped at a level that would have controlled any release below the applicable limitations.

The inspector reviewed UFSAR Section 11.5 and the setpoint data recorded on ST-5-026-880-1, "Evaluation of the Hi and Hi-Hi alarm setpoint for the RHR/SW Rad Monitors." The inspector confirmed that even with the higher set points they were still conservative with regard to release limits, and would not have prevented the isolation of RHRSW heat exchanger and/or trip of the RHRSW pump. Therefore, the RHRSW radiation monitors remained capable of performing their safety functions to control the release of radioactive material.

The technician was counseled concerning the need for better attention to detail. The inspector has no further questions concerning this event.

Unit 2

Toxic Gas Monitor Alarm

On April 17, 1992, at 4:34 a.m., a main control room toxic gas alarm was received due to an indicated high ethylene oxide concentration on the a 'B' channel of the toxic gas detection system. The control room operators followed procedure SE-2, "Toxic Gas," and manually isolated the control room and donned Scott Air Packs.

Subsequent chemistry sampling did not identify the presence of toxic gas. The NRC received reports of the above event via the ENS. The inspectors determined that the licensee's initial response and corrective actions were appropriate. The root cause analysis and the need for additional/long-term corrective action will be reviewed upon issuance of the Licensee Event Reports as part of the routine inspection program.

2.0 SURVEILLANCE/SPECIAL TEST OBSERVATIONS (61726)

During this inspection period, the inspector reviewed in-progress surveillance testing and completed surveillance packages. The inspector verified that surveillances were done according to PECo approved procedures and plant Technical Specification requirements. The inspector also verified that the instruments used were within calibration tolerance and that qualified technicians did the surveillances. The activities observed by the inspectors were acceptable.

3.0 MAINTENANCE OBSERVATIONS (62703)

The inspector reviewed the following safety-related maintenance activities to verify that repairs were made according to approved procedures and in compliance with NRC regulations and recognized codes and standards. The inspector also verified that the replacement parts and quality control used on the repairs were in compliance with PECo's Quality Assurance (QA) program.

3.1 Steam Separator Removal

On March 27, 1992, problems were encountered during the removal of the steam separator. During the lift of the separator the main hoist motion stopped with the separator approximately 20 feet above its normal seated position. At this point the separator was positioned over the reactor vessel and still engaged with the guide rods. The control room Shift Supervisor was informed and all work stopped until the situation was reviewed by maintenance supervision. PECo decided to perform an inspection of the crane and if no problems were obvious the separator would be lowered back into its normal location. No problems were evident with the crane hoist and the separator was reseated. The inspector observed this operation.

The inspector attended a followup meeting where a troubleshooting plan was developed. The results of the troubleshooting did not identify any obvious equipment problems and it was concluded that the likely cause was the trip of an undervoltage relay due to electrical component heatup during prolonged lifting at very slow speeds. The moisture separator was subsequently removed and placed in the equipment storage pit without any additional problems.

The inspector reviewed the following procedures used during this evolution:

M-041-011	Maintenance Procedure for Reactor Vessel Disassembly
M-041-049	Steam Separator Unlatching
M-041-022	Steam Separator Removal
M-098-010	Procedure for Rigging and Handling Heavy Loads
M-098-003	Operation of Reactor Enclosure Crane
A-90	Procedure for the Control of Heavy Loads

During his review of these procedures the inspector found no directions for the use of the main hoist load cell indication or load limit switch. Should a component not be fully unlatched, or hang-up during the lift, the load cell could be used to prevent overstressing and damaging the crane or vessel components. Section 9.1.5.2 of the UFSAR describes the load sensing system and states that it functions to provide load indication to the crane operator, and to limit the hoist to 100 to 115 percent of the rated load by means of a load actuated limit switch.

During the troubleshooting effort PECo determined that the load indicator and limit switch had not been operable since May, 1991. The maintenance staff determined that the crane was operable, although the load cell was not functioning. No written evaluation was performed to document the basis for this determination. However, prior to NRC discussion, the Maintenance Superintendent had recognized that an evaluation had not been done and directed the load cell to be repaired prior to using the crane for any additional lifts. The Maintenance Superintendent agreed that an evaluation was needed. PECo has taken additional corrective actions to include the periodic calibration of the load cell instrumentation and the checkout of the load cell and weight converter as part of the periodic crane test. Also, directions will be provided to the crane operators for proper use of the load cell indication prior to reactor reassembly.

Based on a review of the actions taken and the additional actions planned, the inspector had no further questions regarding this event. The inspector also noted that the oversight for not performing an evaluation of the inoperable load cell appears to be an isolated event.

3.2 Reactor Cavity Drain During Local Leak Rate Testing

On March 23, 1992, a maintenance team commenced local leak rate tests (LLRTs) of the containment isolation valves in the 'B' loop of the reactor feedwater system. The valves were tested according to procedure ST-4-LLR-092-1, "Feedwater." The leak tests were completed on March 26, 1992, at which time the system restoration portion of the procedure began. Step 7.1 directs the test personnel to remove the LLRT tags and return the valves to the "as found" position documented in the Tag Accountability Log. The "as found" positions were recorded at the start of the test. When the maintenance technician began restoring the system valves, he used a copy of the log that did not contain the "as found" data. During the restoration of reactor water cleanup (RWCU) valve 44-1029, the technician returned the valve to its normal (plant operating) position of "locked open" rather than the required "as found" position of "closed." Opening this valve aligned a flowpath from the RWCU system to the turbine enclosure drains via the feedwater system. At the time of the event the vessel head was removed and the reactor cavity filled to the level just below the overflow weirs. No fuel movement was in progress. The RWCU recirculation pumps were operating at the time and the valve misposition resulted in a loss of six inches of level within the reactor cavity. The Unit 1 Reactor Operator quickly noted the loss of level in the cavity and notified the Shift Supervisor who then contacted the maintenance foreman to ascertain the status of the LLRT. Meanwhile the maintenance technician, who heard flow through the valve, realized his mistake and closed the valve. The valve was open for approximately eight minutes. The inspector noted that the technician failed to heed several test procedure cautions stating that improper operation of valve 44-1029 could result in inadvertent draining from the reactor vessel.

During this time approximately 13,000 gallons of water were drained. If this event occurred prior to flooding up the refueling cavity it would have resulted in the reactor vessel level decreasing by 65 inches. Such an uncontrolled loss of cavity or vessel inventory has potentially severe consequences due to the associated loss of shielding and decay heat removal capability.

The failure to follow step 7.1, to return the valves to the "as found" position, is a violation of plant Technical Specification (TS) 6.8.1.d that requires written procedures be implemented for surveillance and test activities of safety-related equipment (50-352/92-22-01).

4.0 ENGINEERING AND TECHNICAL SUPPORT (37700)

4.1 Modifications

4.1.1 Blowout Panel Modification

This modification was developed because of several events involving RE ventilation supply fans failing to trip, when the exhaust fans were turned off, causing the RE to become pressurized and opening blowout panels. These panels are designed to protect the RE from overpressurization following a high energy steam line break (HESLB) accident.

The blowout pressure setting on the panels is 0.25 psid, which is a lower pressure than the supply fans can produce, thus, the blowout panels open if the supply fan continues to run without the exhaust fan running. PECo conducted an engineering evaluation and determined the following:

- The set points of the panels could be raised to 0.50 psid and still protect the RE against HESLB accidents. The higher setting is above the pressurizing capacity of the RE supply fans.

- Tornado depressurization requirements would not be adversely affected.

- Calculations for compartment temperature transients were previously assumed at the pressure of 0.5 psid.

- The potential for RE overpressurization due to ventilation system problems can only occur during non-accident conditions since the RE heating ventilation and air conditioning system is isolated from the secondary containment by a loss of coolant accident (LOCA) signal or by a high radiation signal. Secondary containment isolation valves can close under full air flow, and the standby gas treatment system (SGTS) will not be effected even if the RE supply or exhaust fans continue to run during isolation conditions.

The engineering evaluations supported increasing the set point of the blowout panels. There are four blowout panels that will be increased from 0.25 psid to 0.50 psid. The panels are:

- Steam venting tunnel to atmosphere - Elevation 241
- Steam venting tunnel to stack - Elevation 307
- Main steam tunnel to condenser area - Elevation 253
- Main Steam vent stack - lower to upper positions Elevation 332

The setpoints will be increased by adding additional explosive release fasteners or replacing existing fasteners with similar ones of a higher release value.

The inspector reviewed design change document M6183-1, "Blowout Panel Fastener Upgrades," and the associated documentation necessary to implement the modification. The inspector concluded that the NRC requirements for performing a plant design change were met including completion of a 10 CFR 50.59 review.

The inspector also reviewed a non conformance report (NCR) that questioned the use of sealant around the panel after installation, and its impact on blowout pressure. PECo evaluated testing using a control test box, with a blowout panel installed, and found the sealant and gasket materials acceptable. The products used for gasket and sealant material were specifically referred to in the design package and are being incorporated into the repair procedure. The inspector observed work in progress and concluded that:

- Materials requiring certification were certified, in writing.
- Procedures were being followed by the workers on the job.
- Personnel were knowledgeable of procedures and requirements and implemented them according to approved administrative procedures.

The modification is currently being installed on Unit 1, and will be done on Unit 2 prior to the startup following refueling outage number two.

The inspector concluded that this design change should improve the overall secondary containment integrity, and does not impact any post-accident safety-related engineered safety features described in the UFSAR, Section 6.5.1.

4.1.2 Reactor Protection System Inverter Replacement

Modification 6112-1, "Replace RPS/UPS Inverters," was installed to replace the two reactor protection system (RPS) static inverters. The inverters convert direct current power from a Class IE battery source to 120 volt, 60 cycle alternating current power for the RPS. The inverters were replaced with state-of-the-art design inverters because the existing inverters were no longer in production, and had a high failure rate.

The inspector observed portions of the installation and reviewed the modification package documentation. The new inverters are direct replacements for the existing inverters and do not required any significant wiring changes. The inspector observed that the installation work was of a high quality.

The inspector also reviewed the following new system operating procedures and found them to provide adequate direction for the operation of the new inverters:

S94.1.5 (COL)	Equipment alignment for placing the 1A RPS UPS static inverter 1AD160 in service
S94.1.5	Placing the 1A RPS UPS static inverter in service
S94.2.5	Bypassing and removing the 1A RPS UPS static inverter from service
S94.9.5	Routine inspection of the 1A RPS UPS static inverter

PECo verified that the new procedures worked properly during Modification Acceptance Test.

The decision to upgrade the inverters is a good initiative by the station and corporate engineering personnel. Fewer inadvertent safety system isolations and reactor half-scrams should occur because of this modification. The inspector also noted close involvement of the station system engineers in the modification process, including the witnessing of successful performance tests at the vendor's factory.

5.0 RADIOLOGICAL PROTECTION (71707)

During the report period, the inspector examined work in progress in both units including health physics procedures and controls, ALARA implementation, dosimetry and badging, protective clothing use, adherence to radiation work permit (RWP) requirements, radiation surveys, radiation protection instrument use, and handling of potentially contaminated equipment and materials.

During plant tours the inspectors observed scaffold building for various work activities. The inspector questioned the use of scaffolding in areas where radiation levels were in the 5 to 30 mrem per hour range. The scaffolding was constructed to work on snubber removal and safety valve maintenance that could easily have been reached standing at ground level or on a small ladder. In some cases, this practice is not in the best interest of the ALARA program as more exposure is being obtained by the erection of scaffolding than is being obtained by actual job activities. This practice was discussed with Limerick management.

The inspector observed individuals frisking according to HP procedures. A sampling of high radiation area doors was verified to be locked as required. Compliance with RWP requirements was reviewed during plant tours. RWP line entries were reviewed to verify that personnel provided the required information and people working in RWP areas were observed as meeting the applicable requirements. The activities observed by the inspectors were acceptable.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

As exemplified by the reactor cavity draining event described in Section 3.2, maintenance errors continue to persist as a result of inattention to detail and the failure to adhere to station procedures. PECo corrective actions taken as a result of root cause analyses of these events have not been fully effective. The inspector also found that an analysis was not performed on the use of a main crane hoist after the loss of the installed load cell. These specific deficiencies on the part of maintenance personnel were corrected when identified. However, additional efforts by maintenance personnel could have prevented their happening.

Conservative reporting of two events are discussed in the Section 1.2. The inspector reviewed the results of PECo's investigations into these events and agrees they were not reportable. however, in one case a technician error caused the incident. Two additional technician errors resulted in reportable events during this report period. The licensee has taken corrective actions to address these incidents.

7.0 REVIEW OF LICENSEE EVENT REPORTS (LERs), ROUTINE AND SPECIAL REPORTS (90712, 92700)

7.1 Licensee Event Reports (LERs)

LERs are 30 day reports submitted to the NRC, by PECo, as required by 10 CFR 50.73. These reports document: the major occurrences present during an event, including all component or system failures; a clear, specific, narrative description of what occurred; plant operating conditions before the event; status of contributors to the event; dates and approximate times of the factors; the causes and failure modes; personnel errors if applicable; procedural deficiencies if applicable and the short-term and long-term corrective actions taken to prevent recurrence. The Resident Inspector routinely reviews these documents and performs followup to PECo's actions regarding the disposition of corrective initiatives. In his review, the inspector validates the above and determines whether events are described accurately and whether corrective and compensatory actions have been properly addressed. Unless otherwise delineated below, the following LERs meet all the requirements discussed above.

LER 1-92-002, Event Date: March 14,1992, Report Date: April 7,1992
This LER discusses a failed inverter for the 'D' low pressure injection system, 'B' core spray subsystem, and high pressure coolant injection (HPCI) system. The LER also discusses entry into TS 3.0.3 to repair the inverter. The NRC granted a temporary waiver of the TS to repair the inverter. The discussion of this event is recorded in Combined Inspection Report 50-352/92-11 and 50-353/92-11, the inspector has no further questions.

LER 1-92-003, Event Date: March 21, 1992, Report Date: April 15, 1992
This event involved an ESF actuation due to a chemistry technician entering incorrect settings into the radiation and meteorological monitoring system computer. This was discussed in Section 1.2 of this report.

LER 1-92-004, Event Date: March 21, 1992, Report Date: April 16, 1992
The licensee experienced a loss of reactor enclosure secondary containment integrity due to the openir of a blowout panel. This was discussed in Section 1.2 of this report.

LER 2-92-004, Event Date: February 21, 24 and 26, 1992, Report Date: March 19, 1992
The HPCI system was made inoperable due to blown fuses in the inverter power supply. This event was discussed in Combined Inspection Report 50-352/92-11 and 50-353/92-11. The inspector has no further questions.

7.2 Routine and Special Reports

Routine and special reports are submitted by PECo to inform the NRC of routine operating conditions and other noteworthy occurrences that are reportable due to requirements in 10 CFR 20, TS and other regulatory documents. The inspector reviews these reports for information and confirms the accuracy of the reports. The Monthly Operating Report for March 1992, dated April 10, 1992, was reviewed and satisfied the requirements for which it was reported.

8.0 FOLLOWUP OF PREVIOUS INSPECTION FINDINGS (92702)

(Closed) Unresolved Item (50-352/88-07-01). This item is related to Inservice Testing (IST) program control and the need to 1) improve administrative procedure control; 2) improve administrative procedure A-80 to define IST responsibilities; 3) assign valve maximum allowable stroke times; and 4) assign rapid acting valve stroke times that are in agreement with NRC guidance. The inspector reviewed the evaluations and actions taken by the licensee in response to the above issues.

The IST program document ML-008 issued December 22, 1988, for the first ten year interval was reviewed by the inspector. The original program has been updated to Revision 2 and was verified to be a fully controlled document. The inspector reviewed several of the Revision 2 changes, specifically those made to incorporate NRC safety evaluation positions, and determined that the program was well prepared and meets ASME requirements and NRC

Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," guidance. Program revisions were noted to include a descriptive index record of the changes, and the program pages contain right hand column bars at change locations.

The inspector determined that Administrative Procedure A-80, "Inservice Inspection," was augmented by the newly developed and released Procedure A-80.7 "Inservice Testing." This procedure specifically addresses controls, responsibilities and implementation of the IST program in response to the NRC identified item. Review of A-80.7 verified that individual responsibilities are fully defined in Section 5.0 of the procedure.

The valve stroke time issues were fully reevaluated by the engineering staff and a rationale was developed that resulted in appropriate and conservative stroke times. The assigned reference stroke times were based on vender data and the prior stroke history of each valve. the ASME Section XI 1986 Code, ASME Standard OM-10 "Inservice Testing of Valves in Light Water Reactor Plants," GL 89-04 guidance, UFSAR requirements, operating procedures, and TS formed the bases for the alert and action stroke times. The most restrictive of these was used as the maximum allowable stroke time.

The alert and action (maximum allowable) valve stroke time criteria are:

- For electric motor operated valves:

  a) With reference stroke times > 10 seconds:
     Alert stroke time = ± 1.10 (reference stroke).
     Maximum allowable stroke time = 1.15 (reference stroke).

  b) With reference stroke > 2 and ≤ 10 seconds:
     Alert stroke time = ± 1.20 (reference stroke) ± 1 second.[2]
     Maximum allowable stroke = 1.25 (reference stroke) + 1 second.

  c) With reference stroke ≤ 2 seconds:
     Alert stroke time = not applicable.
     Maximum allowable stroke = 2 seconds.

- For other power operated valves:

  a) With reference stroke > 10 seconds:
     Alert stroke time = 1.25 (previous test stroke) per Section XI.
     Maximum allowable stroke = 1.25 (reference stroke).

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[2]The ± 1 second is in process of being eliminated.

b) With reference stroke > 2 and ≤ 10 seconds:
Alert stroke time = 1.50 (previous test stroke) per Section XI.
Maximum allowable stroke = 1.50 (reference stroke) + 1 second.

c) With reference stroke ≤ 2 seconds:
Alert stroke time = not applicable.
Maximum allowable stroke = 2 seconds.

The inspector verified that the stoke test records of several recent surveillance tests (ST-6-049-200-2, "RCIC Valve Test," ST-6-057-200-1, "CAC Valve Test," and ST-6-011-203-1, "A Loop ESW Valve Test") followed the newly established stroke time criteria. The inspector noted that the new acceptance criteria were appropriate and conservative.

The last issue of the unresolved item involved the stroke time limits for rapid acting valves, those that stroke in 2 seconds or less. As noted in the preceding paragraphs PECo has modified the rapid acting valve stroke time acceptance criteria according to NRC/PECo IST program meeting agreements and GL 89-04 positions. The inspector reviewed procedures ST-6-061-200, Revision 10, and ST-6-011-203-1, Revision 4 and verified that the new rapid acting valve stroke times were appropriate.

The inspector reviewed PECo's recently implemented graphic trending program for safety-related pumps and valves. The inspector determined that the preparation for the trending program involved the indexing of all prior and current surveillance test results into an electronic data base. Graphic trend charts that display the prior and current test result data are used by the IST coordinator and the system engineers to evaluate component performance. Currently the need for the continued charting of stroke times for electrical motor operated valves is being reviewed because of the stroke time repetitiveness of these valves.

Discussions were held with cognizant engineers who pointed out several of the benefits derived from the program. One example reviewed by the inspector was trending that projected the time that diesel generator fuel oil transfer pump 1AP 514 would reach the required action range. This information enabled engineering and maintenance to plan purchase of parts and the scheduling of the repair. The inspector concluded that the trending program has provided beneficial results.

The inspector concluded that the IST program document ML-008 is effective and well controlled. The reformatting of the IST program improved the document and the thorough valve stroke time evaluations resulted in conservative stroke acceptance criteria. This item is closed.

(Closed) Unresolved Item (50-352/90-09-001). Interim Radwaste Storage. This item was opened to track PECo's progress toward planning and constructing an interim storage facility for radwaste. Since this item was opened PECo has formulated plans for a facility and has initiated a plant modification for the design and construction of a facility. Based on these actions this item is closed.

(Closed) Unresolved Item (50-352/91-14-001). Blowout Panel Actuations. This item was open pending NRC review of PECo corrective actions to resolve the problem with spurious actuations of reactor enclosure blowout panels due to ventilation system failures. Based on the implementation of the Unit 1 modification discussed in Section 4.1.1 of this report, and the planned implementation of a similar modification on Unit 2, this item is closed.

(Closed) Unresolved Item (50-353/91-17-03). Additional corrective actions for LER 2-91-012. This item was opened to track the NRC review of additional PECo corrective actions to be taken in response to an event in which the removal of floor drain plugs negatively impacted the reactor enclosure secondary containment. The additional corrective actions, and clarifications of the event description and cause, were reported in Revision 2 to LER 2-91-012. Based on a review of the revised LER the inspector concluded that PECo took appropriate corrective actions. This item is closed.

9.0 MANAGEMENT MEETINGS

9.1 Exit Interviews

The NRC Resident Inspectors discussed the issues in this report with PECo representatives throughout the inspection period, and summarized the findings at an exit meeting with the Vice President, Limerick Generating Station, Mr. Graham Leitch, and Plant Manager, Mr. J. Doering, on April 27, 1992. No written inspection material was provided to licensee representatives during the inspection period.

9.2 Additional NRC Inspections this Period

The Resident Inspector also attended the following exit interviews during the report period:

Date	Inspector	Report	Subject
March 29, 1992	Sami Sherbini	50-352/92-13 50-353/92-13	Special inspection worker contamination.
April 24, 1992	Sami Sherbini	50-352/92-14	Health Physics, outage inspection.

March 25, 1992 — Routine visit by Charles W. Hehl, Director, Division of Reactor Projects. Mr. Hehl toured the facility and had discussions related to motor operated valves, service water systems, corrective maintenance backlog and recent maintenance events.