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REGION I

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Location:	Wayne, PA 19087-0195
Dates:	November 18, 1997, through January 19, 1998
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EXECUTIVE SUMMARY
Limerick Generating Station, Units 1 & 2
NRC Inspection Report 50-352/97-10, 50-353/97-10

This integrated inspection included aspects of PECO Energy operations, engineering, maintenance, and plant support. The report covers a 9-week period of resident inspection.

<u>Operations</u>

- Control room supervisors at both units logged over 100 Technical Specification Limiting Conditions for Operations for primary containment isolation valves during the follow up inspection and testing event. Overall, log entries were adequately controlled. However, several issues involving the accuracy of the unified log were identified by the inspection. The most significant of these was the failure to log that two safety systems were inoperable resulting in a violation of controls stated in the Operation Manual for maintaining the unified log (Section O2.1).

<u>Maintenance</u>

- Overall, maintenance technicians completed the replacement activity of the 1D 125vdc safeguards battery well. However, there were several housekeeping and work practice issues which could have impacted battery operability (Section M1.3).

- The large number of similar hydraulic control unit (HCU) discrepancies identified during PECO's follow-up investigation to an individual control rod that fully inserted during a reactor protection system surveillance test indicated that inadequate maintenance had been performed during the recent on-line maintenance activities and during prior maintenance activities. The Nuclear Maintenance Division (NMD) appeared to have established adequate control and oversight of the on-line HCU work activities and NMD technicians demonstrated a good awareness and responsibility toward quality by stopping work to notify his supervision of a wiring discrepancy. However, PECO did not establish adequate measures to assure that the applicable design requirements were adequately maintained during HCU on-line maintenance resulting in a violation (Section M1.4).

- The licensee's response to the failure of the power monitor card in the Unit 1 RRCS was excellent. The licensee promptly established RRCS operability and corrected the problem. Adequate consideration was given to the method used to prevent an inadvertent plant trip during the maintenance repair, including use of the training simulation to heighten technician awareness (Section M1.5).

- In general, Limerick has adequate control of portable and/or temporary equipment in the reactor and turbine building such that it will not interact with equipment important to safety. Furthermore, this program is strengthened by periodic walkdowns and critiques with first line supervisors. However, while reviewing plant housekeeping, the inspector noted several discrepancies that were not identified by the PECO staff. Further, in the case of the deficient bolting associated with the monorail hoist it appeared the condition existed for a long period of time (Section M2.1).

- ST-6-076-360-1(2), Reactor Enclosure Secondary Containment Integrity Verification, overstated the requirements to meet Technical Specification 3.6.5, by equating the floor drain plugs with the components required to maintain secondary containment. Control of the configuration of these plugs remains necessary to prevent creating an opening in the secondary containment that would prevent the standby gas treatment system from maintaining secondary containment in the event of an accident. Inadequate control of the plugs demonstrated in October and the lack of timeliness for incorporating the proposed procedure revision have resulted in a violation (Section M8.1).

Engineering

- The PECO engineer demonstrated excellent awareness of component configuration by recognizing a mis-wired closing circuit for an Unit 1 reactor core isolation cooling steam isolation valve. Engineering promptly identified that the PCIVs were not adequately tested and implemented adequate measure to complete the required testing within the time allowed by technical specifications (Section E1.1).

- The engineering assessment and supporting safety evaluation to support operability of the HPCI exhaust valve was inadequate in that it did not address the valve closure time requirements. The plant operations review committee (PORC) approved the safety evaluation, but failed to challenge the engineering assessment discounting the requirement for the valve to close the first time to meet the closure time required by technical specifications in assessing operability. PORC accepted the degraded condition of the valve without having identified the root cause or evaluating the corrective actions to ensure future valve reliability and thereby the ability to meet the required closure time (Section E2.1).

  The use of a safety evaluation to accept the delay in further investigations and testing of the HPCI exhaust valve, until the next scheduled refueling outage, in effect inappropriately modified the technical specifications required closing time. The use of the safety evaluation in addressing operability was not necessary nor consistent with NRC guidance on operability provided in generic letter 91-18.

- The organization response to the D22 emergency diesel generator failing into the isochronous mode of operation was good, particularly since another EDG was inoperable for planned maintenance and was competing for the same personnel resources. The D22 EDG was returned to an operable status in about two and a half days after a thorough assessment of the overpower event which including a variety of followup inspections and measurements. The root cause analysis of this event was adequate; however, documentation weaknesses were noted including the as-found conditions not being documented in detail in the work order (Section E2.2).

- The licensee appropriately implemented the commitment change process for the main safety relief valve commitment change. Although the timing of NRC notification for the change was sooner than required by the process, the letter was misleading in that it implied that the change had been implemented as of the date of the letter, whereas three months later at the end of the inspection period the change had not been implemented. However, no violation of NRC requirements was identified. In addition the engineering evaluation to support the modification of the commitment was not comprehensive in that it did not correlate the performance data to specific changes in the thresholds values (Section E6.1).

Plant Support

- The radiation protection program controls for preventing internal exposures was effective. No significant personnel exposures were apparent. However, the whole body measurement capability appeared to lack sufficient rigor in assuring that all internally deposited radio nuclides, that the whole body counting instrument was expected to detect, were effectively identified and evaluated. It was not apparent that staff were cognizant of the inherent limitations of the equipment relative to discreet resolution of energy peaks to effect radio nuclide identification (Section R1.1).

- The respiratory protection program met regulatory requirements (Section R1.2).

- The air sample counting laboratory provided properly calibrated and reliable sample analysis services (Section R1.3).

- The inspector determined that the licensee's radiation protection instrument calibration program generally utilized sound principles and techniques. However, the process did not address or compensate for certain uncorrected calibration errors that could effect instrument accuracy. Notwithstanding, the instrument calibration process was determined to be effectively implemented. The TLD program oversight was very effective in enhancing the accuracy of vendor TLD processing results (Section R1.5).

- The bases upon which the licensee resolves exposure discrepancies between TLD and electronic dosimeter quarterly results was not apparent. The area will be further reviewed in a subsequent inspection (Section R4.1).

- Oversight of the radiation protection program consisted of independent and self-assessments that generally provided for effective insights and recommendations for program improvements (Section R7).

- The Radiation Protection (RP) training program was adequate. The licensee identified a weakness in the RP fundamentals training provided to RP technicians in the continuing training program, and has made some progress in addressing this concern (Section R5.1).

- The licensee has limited procedural controls over the advanced radiation worker program. Some survey and contamination area deposting activities have been performed by the advanced radiation workers that involved evaluation and judgement determinations without qualified RP technician supervision. Further investigation in the advanced radiation worker training and performance are needed to determine whether a violation of TS 6.3.1 has occurred (Section R5.2).

- An unqualified person had been assigned to perform tasks which require formal qualification. Generally, there was evidence of direct supervision for the more critical tasks performed by unqualified individual such as the performance and evaluation of whole body counts. However, for administrative tasks, generally there was no recorded evidence of direct supervision as required by the licensees training and qualification procedures. Although, the practice of using unqualified and unsupervised personnel is inconsistent with the licensee's procedure, this was determined not to be a violation of regulatory requirements since the position or job functions are not specifically addressed through the technical specification requirements for the training of plant staff. However, the failure of the licensee to appropriately control the use of unqualified personnel is of concern since the same procedure control are used to address positions which have specific training requirements (Section R5.3).

- Although, the licensee was not in full compliance with Procedure ERP-600-1, Health Physics Team, they were proactive in identifying the issues and their corrective actions are adequate for preventing recurrence. The inspector also noted that these issues were not identified in previous exercises or drills because the licensee had typically conducted their exercises during working hours in which HP technicians were onsite and available for immediate response (Section P4).

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Report Details

Summary of Plant Status

Unit 1 began this inspection period operating at 100% power. The unit remained at full power throughout the inspection period with exceptions for testing, rod pattern adjustments, and the following plant events.

- December 6 — Operators entered a 12-hour shutdown limiting condition for operation (LCO) in accordance with Technical Specification (TS) 3.6.3 after declaring several primary containment isolation valves inoperable for not meeting all requirements. The valves were properly tested and declared operable. Operators exited the LCO prior to the end of the 12-hours.

- December 13 — Operators reduced power to 65% to perform on-line maintenance on 52 hydraulic control units (HCUs). Maintenance activities were completed and the unit returned to full power on December 17.

- December 28 — Operators reduced power to 70% to remove the 1A condensate pump from service after operator noted degraded discharge pressure and significant vibration conditions with the pump's performance. Unit power was increased to 77% power during the period that maintenance technicians replaced the pump. Operators restored the unit to full power on January 4, 1998.

Unit 2 began this inspection period operating at 100% power. The unit remained at full power throughout the inspection period with exceptions for testing, rod pattern adjustments, and the following plant event.

- December 6 — Operators entered a 12-hour shutdown LCO in accordance with TS 3.6.3 after declaring several primary containment isolation valves inoperable for not meeting all surveillance requirements. The valves were properly tested and declared operable. Operators exited the LCO prior to the end of the 12-hours.

I. Operations

O1 Conduct of Operations[1]

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, PECO Energy's conduct of operations was professional and focused on safety principles.

O2 Operational Status of Facilities and Equipment

O2.1 Primary Containment Isolation Valve Configuration Control

a. Inspection Scope

On December 5, an engineer inspecting the breaker cubical for the Unit 1 reactor core isolation cooling (RCIC) inboard steam isolation valve (HV-49-1F007) identified a mis-wired closing circuit (see Section E1.1). During the subsequent investigation the engineering staff identified a testing deficiency that potentially affected the operability of numerous other primary containment isolation valves (PCIVs). Between December 5 and 7 (about 48 hours), the operations staff maintained control of safety-related system operability per TS during the plant-wide follow-up investigation and testing of the affected motor operated valves at both units. The inspector reviewed the unified control room log and the LCO log to ensure the appropriate LCO entries were made for PCIV and safety system inoperability. The inspector discussed his findings with representatives of the Operations Department staff. A Performance Enhancement Program (PEP I0007700) evaluation was initiated to address the inspectors concerns.

b. Observations and Findings

Shift management entered over 100 TS LCO entries into the control room's unified log in a 48-hour period. The unified log is a computer based log that records narrative log entries from both unit reactor operators, the chief operator, control room supervisor (CRS) and Shift Manager. The log also tracks TS LCO entries. During the 48-hour period, a unit supervisor was assigned to each unit to assist the CRS in maintaining control of the large number of TS LCO entries at both units for the inspection and testing of the effected valve motor operator circuits. Overall, the activities were generally performed well, with the exceptions noted below.

Station engineers determined that the closing circuits for the PCIVs had not been adequately tested. Therefore, shift management entered TS 4.0.3, allowing 24-hours to satisfy the missed surveillance testing prior to having to implement the required action as per the TS LCO. Shift management entered this TS at 10:30 a.m. of December 5. The

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1 Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

consequences for not completing the required surveillance testing within the 24-hour period would be to shutdown both units within 12 hours. While technicians tested each PCIV, the unit supervisor also entered the four-hour action statement for TS 3.6.3 (primary containment isolation valves) as appropriate.

The inspector identified several problems from the unified log review. Two safety-related systems were made inoperable during the valve testing and no TS LCO log entry for the system's inoperability was made in the unified log. The Unit 2 suppression pool spray mode of residual heat removal (RHR) system (TS 3.6.2.2) and high pressure coolant injection (HPCI) system (TS 3.5.1.c.2) were made inoperable (separately) for about two hours. The inspector verified that all alternate and low pressure coolant injection systems were operable during the time both systems were unavailable as required in each of the associated TS action statements. Therefore, the technical specifications for these two cases were technically met. However, Operations Manual OM-L-12.1, Regulatory Action, step 4.4, requires a narrative log entry in the unified log for the safety system inoperability

LCO numbers were not unique. Several factors caused this problem including:

- Duplicate LCO numbers were created when log entries were made within the 10-minute time period between system updates.
- LCO entries were inadvertently edited, changing the LCO from the original entry.
- The same LCO number was repeated several time throughout the year for different TS LCO entries.

The above mentioned variations were apparently caused by the computer software. For example, if the operator intended to initiate an LCO entry, the computer displayed the next chronological TS LCO entry number. This number, however, appeared on every computer terminal that allowed more than one supervisor to be entering differing TS LCOs with the same number. Operations management stated that the computer system software was unable to keep pace with the large number of entries made from multiple terminals.

Further, the inspector identified several significant typographical errors. These included a TS LCO closure at midnight, about six hours prior to the time logged initiating the LCO, and a core spray (system 52) valve that was typed as an RHR valve (system 51). The inspector raised concerns regarding the frequency and quality of log reviews performed by operations supervision. Operations management assured the inspector that the unified log is the official record of plant activities and that it was crucial that the log be complete and accurate.

Operations management agreed with the discrepancies noted, but stated that they were administrative in nature and in no case did they result in the inappropriate control of equipment operability or in non-compliance with TS. Management also stated that no narrative log entry was made for making the Unit 2 HPCI and suppression pool spray systems inoperable, however, a narrative log entry would be reconstructed and a late log entry made. Notwithstanding management's intent, the inspector presented his findings to the operations management three days following the event and no edits had been made in the log up to that point. Therefore, the inspector concluded that the log did not accurately reflect conditions as they occurred in the plant. OM-L-8.2, Narrative Logs / Scope of Entry, states that items are to be entered into the log pertaining to system operability or affecting the station. This action was not performed on two occasions. This is a violation. **(NOV 97-10-01)**

c. Conclusion

Control room supervisors at both units logged over 100 Technical Specification Limiting Conditions for Operations for primary containment isolation valves during the follow up inspection and testing event. Overall, log entries were adequately controlled. However, several issues involving the accuracy of the unified log were identified by the inspection. The most significant of these was the failure to log that two safety systems were inoperable resulting in a violation of controls stated in the Operations Manual for maintaining the unified log.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance Activities (62707)

The inspectors observed selected maintenance activities to determine whether approved procedures were in use, details were adequate, technical specifications were satisfied, maintenance was performed by knowledgeable personnel, and post-maintenance testing was appropriately completed.

The inspectors observed portions of the following work activities:

- Unit 1 Division 4 125vdc Safeguards Battery Replacement - November 18 - 21;
- Unit 1 High Pressure Coolant Injection Inboard Steam Valve Backseating - November 19;
- Unit 1 HCU on-line maintenance - replacement of SSPVs, - December 15;
- Unit 2 D23 18-month Inspection, December 8 - 12;

Observed maintenance activities were conducted well using approved procedures, and were completed with satisfactory results. Communications between the various work and support groups were good, and supervisor oversight was good.

Overview of Raising M/G Set Stops per SP-147

The inspector observed the adjustment of the 1B reactor recirculation motor generator scoop tube stops. The reactor operator and control room crew had effectively minimized distractions during this adjustment. Contingency procedures were opened and ready if required during the adjustment. The maintenance and engineering personnel involved in the physical adjustment at the scoop tube positioner were aware of the potential reactivity effect associated with working on this equipment. The supervisor in the field was aware of the requirement for a senior reactor operator (SRO) to control this evolution. The adjustment was completed satisfactorily.

M1.2 General Comments on Surveillance Activities (61726)

The inspectors observed selected surveillance tests to determine whether approved procedures were in use, details were adequate, test instrumentation was properly calibrated and used, technical specifications were satisfied, testing was performed by knowledgeable personnel, and test results satisfied acceptance criteria or were properly dispositioned.

The inspectors observed portions of the following surveillance activities:

- Unit 2 - D23 24-hour Endurance Test and Hot Restart - December 15;
- Unit 2 - Inservice Inspection Functional Pressure Test of HPCI Pump Discharge and Turbine Exhaust Piping - December 17;
- Unit 2 - HPCI Quarterly Surveillance Test - December 17;
- Unit 2 - D21 Weekly Surveillance Test, December 31;

Observed surveillance tests were conducted well using approved procedures, and were completed with satisfactory results. Communications between the various work and support groups were good, and supervisor oversight was good.

M1.3 Division 4 Safeguards Battery Replacement - Unit 1

a. Inspection Scope

During the week of November 17, maintenance electricians and I&C technicians replaced completely the Division 4 125vdc safeguards battery. The work activities included replacing the 60 battery cells and the inter-connecting hardware, and inspecting and cleaning of the battery rack. The inspector observed portions of the activity and discussed the observations with several maintenance representatives. The inspector reviewed the operations log for appropriate TS LCO entries.

b. Observations and Findings

The technicians completed the activity over a four day period, replacing 15 cells per day without making the battery inoperable. The battery was maintained operable throughout the evolution by jumpering the 15 cells to be replaced with a temporary safeguards battery. The temporary battery is maintained in the same condition as the inservice battery, is mounted in a seismically qualified cart, and meets the requirements of technical specifications.

The inspector noted several deficiencies in housekeeping and maintenance practices during the first day's activities. A temporary battery charger was left unattended without being properly secured. Battery cables from the temporary battery were routed through and/or tied to structural supports without using a softening material to protect the cables from chaffing from the sharp edges of the support. One cable was routed under a florescent lamp fixture. The inspector raised concern that the cable may have affected the seismic class II evaluation for the lamp fixture over the seismic class I component. Tools were not stored properly and an atmosphere monitoring device was left on a panel overtop of the temporary battery. Further, a battery lead, disconnected from the removed battery cells, was routed through the battery support rack to keep it out of the way, presented a potential electrical hazard. The inspector discussed these observations with the maintenance foreman.

Revisiting the area the next day, the inspector observed general improvement in the condition of the battery room. The temporary battery cables were routed through an industrial cable guard on the floor, tools and other materials were properly stored, and the disconnected battery cable properly isolated. A seismic engineer evaluated the overhead lighting and determined that the cable running under the lamp did not present a concern. The seismic class II over class I concern deals with the S-hooks used to suspend the lamp from the ceiling. The lamp could possibly be jarred out of the S-hooks if the hook was not closed or sealed properly. In this case, the S-hooks were closed and sealed and therefore did not create a problem.

The inspector noted that this was the first battery replacement performed by the maintenance electricians. The task had been the responsibility of the I&C technicians and was now being turned over to the electricians.

c. Conclusion

Overall, maintenance technicians completed the replacement activity of the 1D 125vdc safeguards battery well. However, there were several housekeeping and work practice issues which could have impacted battery operability.

M1.4 Hydraulic Control Unit Maintenance Activities

a. Inspection Scope (62707)

Several maintenance related activities involving HCU's at both units occurred during the inspection period. PECO Energy's Nuclear Maintenance Division (NMD) performed an on-line maintenance outage on selected Unit 1 HCUs beginning on December 12. On December 26, at Unit 2, a single control rod fully inserted without operator action during the performance of a reactor protection system (RPS) surveillance test. The inspector observed portions of the on-line maintenance activities performed at the HCUs. Further, the inspector reviewed the Unit 2 event, the PEP evaluation, and discussed the event with several PECO representatives.

b. Observations and Findings

Hydraulic Control Unit On-line Maintenance - Unit 1

The HCU maintenance focused on replacing the remaining scram solenoid pilot valve (SSPV) assemblies which utilized diaphragms made of BUNA-N material. Maintenance technicians used maintenance procedure M-047-027, Preventive Maintenance for HCUs, throughout the activities. NMD technicians replaced 52 SSPV assemblies over a five day period.

Work activities were planned, coordinated, and executed well between the Operations and NMD Departments, and the reactor engineering staff. Operators and reactor engineers performed a large number of control rod manipulations without error. Further, NMD personnel performed clearance and tagging responsibilities, maintenance activities, and HCU restoration without error. Following HCU restoration, operators performed scram time testing to verify the control rod's operability.

A technician identified a wiring discrepancy at HCU 38-43. The wiring for the SSPVs (V-117 and V-118) was found reversed. The technicians found the V-117 wired to the terminals supplied by 'B' reactor protection system (RPS) and V-118 wired to the terminals supplied by 'A' RPS. The technician immediately stopped work and notified his supervisor. Technicians checked all other HCUs to determine the scope of the problem. No other discrepancies were noted.

The system manager issued Non-Conformance Report (NCR) 97-03427 to address the issue. The NCR determined that the HCU would have performed its scram function regardless of which RPS bus the SSPVs were wired to. Further, the configuration problem did not present a single failure concern or have an impact on channel separation, and therefore was operable. The inspector found NCR's determinations to be acceptable. Maintenance was last performed on the HCU during an overhaul in 1993.

As a result of this discrepancy, NMD revised procedure M-047-027 to include several procedural enhancements. A 'Note' to enhance the po[illegible] identification of the V-117/118 SSPVs was added to the section for the SSPV replacement, as well as improved wire identification, and the wiring termination locations. The inspector determined that the safety consequences of this discrepancy were minor, in that, the scram function of the HCU was not effected by the wiring configuration. Further, the procedures changes appeared to enhance HCU wiring configuration control.

Single Control Rod Scram During Reactor Protection System Surveillance - Unit 2

On December 26, a single control rod fully inserted without operator action during the performance of a reactor protection system (RPS) surveillance test. An I&C technician was performing ST-2-042-645-2, RPS and NSSS Steam Dome Pressure, Channel A Functional, when the event occurred. The control room staff notified the NRC per the requirements of 10 CFR 50.72(b)(2)(ii), but later retracted the notification.

Control room operators immediately entered off-normal procedure, ON-104, Control Rod Problems, verified that control rod 10-47 was fully inserted, and verified all thermal limits were normal. The shift manager declared the control rod inoperable and directed that the HCU be hydraulically isolated. Nuclear Maintenance, I&C personnel, and reactor engineers initiated troubleshooting activities under troubleshooting control form, TCF 97-0965.

The plant staff's investigation at the HCU revealed loose terminal block connections on the load side leads to SSPV supplied from the 'B' RPS channel. The terminal block screws were found to be backed-off about three to four turns. The plant personnel at the HCU observed that the SSPV de-energized intermittently when the I&C technician attempted to tighten the screws. The reactor engineer hypothesized that the SSPV supplied from B RPS channel de-energized due to the loose connection prior to or when the surveillance test initiated the A RPS half-scram signal. The I&C technician tightened the connections and reactor engineers performed a partial scram timing test to prove the operability of the control rod.

Technicians performed an inspection of all Unit 2 HCUs for similar problems and found 22 other HCUs with variations of the same discrepancy. These findings were documented in PEP I0007742. Several other maintenance discrepancies were also identified and corrected during this inspection. A terminal lug was improperly landed at HCU 38-07. The lug was held in place by the screw head "pinching" down on the outside of the lug because the screw did not fit through the eyelet of the lug. At HCU 34-27, the technicians found a loose screw that had backed out to its last two threads. The inspection at Unit 1 identified 10 HCU discrepancies. One HCU was found with a cross-threaded terminal screw.

The inspector reviewing the maintenance history of the affected HCUs determined that HCU 10-47 was last worked in January 1996 as were 13 other of the 32 HCU identified at both units during this event. 11 HCUs were worked during the recent on-line maintenance activities in November and December 1997, one in March 1997, two in July 1995, and four HCUs were worked in December 1994. The inspector determined that the above examples demonstrated inadequate maintenance of the in-field changes performed during these previous on-line maintenance activities.

The inspectors concluded that the safety consequence of the event was minimal, but was concerned with the large number of examples of poor quality craftsmanship and design control during on-line maintenance. Appendix B, Criterion III, of 10 CFR 50 states, in part, that measures shall be provided for verifying or checking the adequacy of design changes performed during maintenance and repair, and that design changes, including field changes shall be subject to design control measures commensurate with those applied to the original design. Contrary to the above, PECO did not establish adequate measures to assure that the applicable design requirements were adequately maintained during HCU on-line maintenance. This was a violation of 10 CFR 50, Appendix B, Criterion III. **(NOV 97-10-02)**

PECO retracted the four-hour notification of the event based on the guidance of NUREG-1022, (Event Reporting Guidelines 10 CFR 50.72 and 50.73). The single control rod insertion was not considered an ESF actuation by itself and was not the result of an actuation of the RPS system. Further review by reactor engineering determined that the capability of the RPS and scram function of all control rods was not adversely impacted by the identified loose screws.

c. Conclusion

The large number of similar hydraulic control unit (HCU) discrepancies identified during PECO's follow-up investigation to an individual control rod that fully inserted during a reactor protection system surveillance test indicated that inadequate maintenance had been performed during the recent on-line maintenance activities and during prior maintenance activities. The Nuclear Maintenance Division (NMD) appeared to have established adequate control and oversight of the on-line HCU work activities and NMD technicians demonstrated a good awareness and responsibility toward quality by stopping work to notify his supervision of a wiring discrepancy. However, PECO did not establish adequate measures to assure that the applicable design requirements were adequately maintained during HCU on-line maintenance as required per 10 CFR 50, Appendix B, Criterion III, Design Control.

M1.5 Redundant Reactivity Control System Corrective Maintenance

a. Inspection Scope (71707)

A review of the licensee corrective action response to the failure of a power monitor card in the Unit 1 Redundant Reactivity Control System (RRCS) Division I was performed. The inspector reviewed the logs and discussed the failure with operators and the RRCS system manager.

b. Observations and Findings

On October 18, Unit 1 received a Division I RRCS Out-of-Service annunciator and an equipment operator was sent to investigate. The equipment operator reported that a "1B1 310 PWR MON TST/PWR SUPPLY FAILURE" error was displayed on Division I RRCS. The RRCS would not reset and an equipment trouble tag was written to document the failure. Subsequent troubleshooting identified that the power supplies were functioning and that the power monitor card had failed indicating that all RRCS functions were still operable. On October 20, the licensee successfully replaced the faulty power monitor card with the RRCS energized returning the RRCS a to fully operational condition.

Prior to implementing repairs, the licensee consulted the RRCS vendor, to determine if the power monitor card could be replaced with the system energized without causing an inadvertent trip. The vendor indicated that a trip should not occur but could not guarantee this assessment. The licensee also verified with the vendor that an updated power monitor card stocked in their supply system was completely compatible with the earlier model power monitor card that was malfunctioning. To provide further assurance, the licensee simulated the power monitor card replacement on a RRCS training simulator on loan from another plant and determined that the card replacement would not cause an inadvertent trip.

The licensee system manager indicated that, although there is no regulatory time constraints involved, repair of the RRCS is treated as an immediate concern since a faulty RRCS can cause an inadvertent plant trip.

c. Conclusions

The licensee's response to the failure of the power monitor card in the Unit 1 RRCS was excellent. The licensee promptly established RRCS operability and corrected the problem. Adequate consideration was given to the method used to prevent an inadvertent plant trip during the maintenance repair, including use of the training simulator to heighten technician awareness.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Plant Material Condition Reviews:

a. Inspection Scope

Plant walkdowns specifically focused on equipment important to safety were conducted to overview the plant material condition. This inspection also reviewed procedure A-C-030, Plant Material Condition and Housekeeping Controls which describes the licensee's controls for material condition.

b. Observations and Findings

The general plant areas in the reactor and turbine buildings were free of clutter. Emergency lighting necessary for plant shutdown under some postulated conditions appeared to be aimed at appropriate equipment and showed an acceptable battery charge. In general, material storage was away from equipment important to safety and properly anchored. Sensitive equipment that could initiate a plant transient was clearly labeled to caution personnel. The inspector noted that periodic walkdowns and critiques of housekeeping areas are performed by peer first line supervisors. However, several deficient conditions were identified by the inspector and are described below.

A large structural steel support for a monorail hoist that penetrates the Unit 1 primary containment access door had five of eight nuts not engaging the embedment plate. In several cases the nuts were backed off as much as an inch and appeared to have been in this condition for some time since the exposed threads had been painted. In response to the discrepancy the licensee performed a field walkdown and removed the hoist from service. The initial engineering review determined that this was a non-conforming condition, but the lateral supports would ensure that seismic loads would not damage the containment door. The licensee plans to complete the engineering evaluation and ultimately corrected the condition.

The inspector identified the instrument line to the Unit 2 Pressure Transmitter, PT-001-207, was vibrating. This is a small diameter line that provides the high pressure turbine exhaust signal to the electro-hydraulic control (EHC) system. This signal is utilized to provide a turbine trip if a load imbalance is sensed between the generator output and the turbine power. Engineering reviewed the configuration of this line and initiated equipment trouble tag (ETT) to provide better support and reduce the vibration of the instrument line. Walkdown of the surrounding area did not identify any safety related equipment that could be impacted by the failure of this small diameter steam line. The Unit 1 instrument line was configured differently and had no observed vibration.

The inspector identified two spare cubicles in 250 Volt DC MCC 1DB-1 that were open to the reactor building atmosphere since no breaker was installed. Engineering determined that there was no environmental qualification (EQ) concern because this area was not subjected to high humidity following accident. However, the engineering staff also determined that the opening should be covered to prevent foreign material from entering the cubicle. The system manager has initiated corrective actions to provide foreign material barriers to cover these openings.

The inspector identified a minor issue, in which an unsecured cart was found next to a safety-related 480 vac motor control center (MCC D114-R-G1). The cart was promptly removed after being brought to the attention of the control room staff.

c. <u>Conclusions</u>

In general, Limerick has adequate control of portable and/or temporary equipment in the reactor and turbine building such that it will not interact with equipment important to safety. Furthermore, this program is strengthened by periodic walkdowns and critiques with first line supervisors. However, while reviewing plant housekeeping, the inspector noted several discrepancies that were not identified by the PECO staff. Further, in the case of the deficient bolting associated with the monorail hoist it appeared the condition existed for a long period of time.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 <u>(Closed) URI 97-03-01, Performance of Reactor Enclosure Secondary Containment Integrity Verification.</u>

a. <u>Inspection Scope</u>

The inspector raised concerns with the Operation Department's methodology to verify the condition of plugged floor drains during the performance of ST-6-076-360-1(2), Reactor Enclosure Secondary Containment Integrity Verification. The concern focused on whether an operator reviewing the locked valve log only, to determine that the floor drain plug's condition, met the requirements of the TS. Operations management enhanced the procedure to include the review of the Barrier Breach Log (A-C-134), and the LCO and potential LCO logs. The inspector left the item unresolved pending the determination of whether any violations using the original methodology had occurred. During the current inspection period, the inspector discussed the issue with several engineering representatives.

b. Observations and Findings

PECO regulatory engineers defined the word "verify", as it is used in the Technical Specifications (TS 4.6.5.1.1.B.), to clarify confusion that resulted from discussions within various plant organizations over compliance with ST-6-076-360-1(2). They determined "verify" was to prove to be true by demonstration; to confirm or substantiate by investigation, comparison with a standard, or reference to the facts. Regulatory concluded that the intent of "verify" was to physically check the required configuration as much as practical, and then refer to the next best alternative that provided relative assurance that the configuration was correct based on the last known change to the configuration.

The engineering staff does not consider the floor drain plugs to be a Technical Specification penetration required to be closed during an accident condition. This is based upon establishing and maintaining secondary containment (a 0.25 inch of vacuum water gage) with the standby gas treatment system (SGTS) and by the satisfactory completion of the required TS surveillance which limits the scope of penetrations requiring surveillance to doors, hatches, dampers, and valves. The SGTS is able to maintain the negative pressure with a design leak tightness of 2500 cfm or less. PECO conservatively had included the floor drain plugs in the monthly surveillance test although they were not explicitly required by the TS definition for secondary containment. An engineering analysis indicated that the removal of a small number drain plugs does not impede SGTS ability to maintain secondary containment, but the removal of a significant number of drain plugs would. The engineers therefore stated that tight configuration controls for the removal of drain plugs would continue to be required and that an engineering evaluation would be performed to determine the amount of air inleakage presented by the opening when several drain plugs were removed to ensure the TS inleakage limit was not exceeded.

The inspector noted a licensee identified event that occurred on October 7, 1997, in which a floor drain plug at Unit 2 was unlocked and removed from drain FD-74 without proper configuration controls as stated in A-C-8, Control of Locked-Valves and Devices. The equipment operator (EO), performing GP-7, Plant Winterization, contacted and discussed opening the floor drain at Unit 1 with a licensed operator because he could not contact the flex supervisor or the control room supervisor. Subsequently, the EO proceeded to Unit 2 to perform the same task. The EO, however, did not contact the control room prior to opening the Unit 2 floor drain because he believed that his previous conversation covered both units. The following day, another EO found the Unit 2 drain opened and that it had been opened for about 26 hours.

The inspector determined that this activity did not meet PECO's configuration controls as stated requirements of A-C-8. A-C-8, steps 7.2.2 and 7.2.3 states, in part, that the individual requesting permission for the manipulation (of the locked device) should enter the valve or device information in the Locked Valve Log and obtain permission from the Shift Management. Shift Management shall then indicate authorization for the manipulation by initialing and dating the Log entry. The EO did not properly fill out the Lock Valve Log nor was Shift Management approval granted prior to removing the floor drain. The inspector determined that this activity was a violation. **(NOV 97-10-03)**

The ST currently reflects the floor drains as a TS required component. PECO intends to revise the ST to remove the asterisk delineating the component as a TS requirement. The floor drains will continue to be checked as stated in the ST. The difference being that they will not have to be "verified" as required by TS. The inspector agreed that floor drains are not defined penetrations as per TS, and drain plugs should not be equated with components required to maintain secondary containment integrity, as was discerned in the ST. However, the ST was the only document delineating what components were specifically required to meet the TS, configuration of the floor drains was not adequately controlled through the normal vehicle (A-C-8), and the proposed revision to the ST has not, to date, been performed.

c. Conclusion

ST-6-076-360-1(2), Reactor Enclosure Secondary Containment Integrity Verification, overstated the requirements to meet Technical Specification 3.6.5, by equating the floor drain plugs with the components required to maintain secondary containment. Control of the configuration of these plugs remains necessary to prevent creating an opening in the secondary containment that would prevent the standby gas treatment system from maintaining secondary containment in the event of an accident. Inadequate control of the plugs demonstrated in October and the lack of timeliness for incorporating the proposed procedure revision have resulted in a violation.

III. Engineering

E1 Conduct of Engineering

E1.1 Primary Containment Isolation Valve Configuration Error and Inadequate Testing

a. Inspection Scope

On December 5, an engineer inspecting a breaker cubicle identified a mis-wired closing circuit for the Unit 1 reactor core isolation cooling (RCIC) inboard steam isolation valve (HV-49-1F007). A contact that bypasses the closed limit switch and thermal overload protection had been incorrectly terminated. During the subsequent investigation, the engineering staff also identified a testing deficiency.

b. Observations and Findings

The engineer recognized that circuit in the AC cubicle was wired in the configuration normal for a DC breaker. Normally in the AC cubicle, the 42-C contact is terminated at terminal block 5&6 and is terminated at terminals 21&22 for the DC.

The circuit, as wired, would permit the closed limit switch instead of the torque switch to stop valve motion during an automatic isolation. Consequently, the valve may not close fully into the seat, creating the potential for leakage past this primary containment isolation valve (PCIV). The licensee declared the RCIC inboard steam isolation valve inoperable and isolated the penetration to comply with technical specifications.

The licensee identified that the computerized wire termination data base was consistent with the mis-wired RCIC circuit. The licensee evaluated the data base and determined that a number of PCIVs had the same or similar type closing circuits. Further review found three additional database descriptions that appeared to be discrepant. Field inspections of these three discrepancies revealed only one additional valve, the Unit 1 RCIC exhaust line vacuum breaker, with the same mis-wiring. The licensee also identified that the PCIVs were not adequately tested. Specifically, the control circuit in question contains two parallel paths; one for manual operation with thermal overload protection and the other for automatic isolation with the thermal overload protection bypassed. Both these paths are energized during automatic valve isolation. The licensee identified that a failure of the bypass contact could be masked by the proper operation of the valve via the thermally protected portion of the circuit. Therefore, the test did not verify that a containment isolation signal would fully close the valve with the thermal overload protection bypassed, as required by technical specifications. The licensee implemented the appropriate technical specification requirements and subsequently tested the bypass contact for all affected valves. All but one PCIV functioned correctly when properly tested and the licensee addressed this malfunction.

The valve mis-wiring problem was identified by an engineer during a breaker cubicle inspection to evaluate the use of some non-quality parts. The licensee also determined that the problem was introduced during a construction modification to add a closed limit switch contact to address another issue with torque switch re-closure following valve isolation.

The mis-wired valve circuit and associated drawing issues are unresolved **(URI 97-10-04)** pending NRC review of the licensee's identification of the root cause and implementation of corrective actions. The inadequate testing issue is also unresolved **(URI 97-10-05)** pending NRC review of the licensee's identification of the root cause and implementation of corrective actions.

c. Conclusion

The PECO engineer demonstrated excellent awareness of component configuration by recognizing the terminal mis-wiring. Engineering promptly identified that the PCIVs were not adequately tested and implemented adequate measure to complete the required testing within the time allowed by technical specifications.

E2 Engineering Support of Facilities and Equipment

E2.1 (Closed) LER 1-97-011 Unit 1 High Pressure Coolant Injection (HPCI) Turbine Exhaust Valve Failure

a. Inspection Scope

On January 8, 1998, the HPCI Turbine Exhaust Valve failed to stroke fully closed on the first attempt during a routine valve stroke time surveillance test. The inspector reviewed the engineering evaluations and corrective actions performed to address the surveillance test failure.

b. Observations and Findings

During stroke time testing of the HPCI turbine exhaust valve (HV-055-1F072), a loud grinding noise was heard at the valve and the valve operator torque switch actuated, stopping valve movement. The normally open valve stopped at approximately twenty five percent closed during the close stroke. The valve was then re-opened and during a subsequent attempt the valve closed without incident. This valve is a remote manual containment isolation valve that is required, by technical specifications, to close within 120 seconds. Although the valve does not have an automatic isolation function, it is necessary to isolate the HPCI system considered to be an extension of the containment boundary, in the event of a HPCI system leakage. The valve was declared inoperable and closed to comply with the primary containment isolation technical specifications.

The failure of a primary containment isolation valve and the associated isolation of HPCI which caused the loss of the high pressure injection safety function was reviewed for reportability and appropriately found to be not reportable. Although the valve condition resulted in the isolation of HPCI to comply with technical specifications, the PCIV deficiency, by itself, would not have resulted in a loss of the a safety function prior to identification and resultant actions taken by the operators. The inspector noted that the licensee had reported the previous valve failure and considered this a conservative report. Although the licensee's reportability determination for the most recent failure was ultimately correct the inspector noted some inconsistencies with the licensee's bases and the NRC guidance (NUREG 1022) on reportabilty. The licensee acknowledged the inconsistency and plans to review and revise their reportability procedures as necessary.

The HPCI turbine exhaust valve is required to be tested quarterly; however, the valve was being tested at a monthly periodicity as a result of previous stroke failures, consistent with the in-service test (IST) program requirements. The inspector found that valve HV-055-1F072 had four similar failures in the last four years. Following each of these failures, the valve was successfully closed on the second attempt after re-opening the valve. Diagnostic testing on the three most recent failures verified that there was no observable valve damage and that subsequent diagnostic tests did not indicate a degradation in valve performance. During the most recent failure, the licensee identified mechanical interaction of valve internal components while performing diagnostic evaluations during the first attempt to close the valve. The failures and associated corrective actions for valve HV-055-1F072 are as follows:

March 1994	During routine HPCI system restoration the valve failed to fully close. The root cause was identified to be lack of lubrication on the valve stem. The stem was lubricated and the valve subsequently stroked successfully.
December 1994	During a scheduled HPCI system work window this valve experienced mechanical binding near the full open position when stroked by hand. The failure was attributed to thermal effects (binding). The valve was placed on increased frequency IST testing (30 day intervals).
May 1995	The torque switch setting was increased to overcome the frictional forces of internal valve binding exhibited in the December 1994 event. The valve was successfully stroked numerous times during increased frequency IST testing (Dec.1994 to May 1995), subsequent quarterly testing and HPCI system scheduled maintenance.
September 1997	During application of a HPCI system clearance for a planned outage window this valve failed to fully close. Diagnostic testing did not identify a root cause and the valve was again placed on IST increased frequency stroke time testing (30 day intervals).
October 1997	During the monthly increased frequency valve stroke time testing the valve failed to fully close on the first attempt. The valve failed in the same manor as the September failure. Investigation of this failure did not identify a root cause. The valve actuator output force was increased, by adjusting the torque switch, as a precautionary measure to improve valve performance. Motor control center (MCC) components were reviewed to ensure that the additional load would not adversely effect other equipment. The valve remained on increased frequency IST valve stroke time testing.
December 1997	Diagnostic testing identified that the valve operator motor was degraded, but operable. Based on the test data the licensee concluded that this may have been a contributor but was not the root cause of the incomplete valve strokes.
January 1998	During increased frequency IST valve stroke time testing this valve failed to closed. The valve was reopened and successfully stroked closed. Diagnostic testing performed during the failed stroke attempt indicated internal valve binding. Subsequent diagnostic testing verified there was no internal valve damage which was consistent with past testing.

The inspector observed the site engineering interdisciplinary review and disposition of the valve performance and associated operability issues. This interdisciplinary review team consisted of the system manager, component experts, engineering supervision, onsite and off site licensing. The review was thorough with good candid discussions on the required safety functions, current licensing basis and technical issues associated with this valve.

Although the review was appropriately focussed on plant safety, the interdisciplining team did not adequately consider the compliance with the stoke time specified in technical specifications. The review concluded that this valve is of minor safety significance, there was no evidence of valve damage during the past failed attempts to close the valve and it could be closed successfully on the second attempt. Therefore, engineering was confident that the valve would close and if it did not close on the first attempt then the applicable TS would be entered. However, engineering did not recommend any additional or different measures to improve the reliability of this valve to close on the first attempt necessary to ensure the required closure time would be met.

The plant operations review committee (PORC) review of this issue considered operational impacts, including, current operator workarounds, accident progression, and operator abilities and concluded that these additional operator actions would not adversely burden a reactor operator. However, the inspectors observed that PORC failed to challenge the engineering recommendation and did not fully address compliance with the required stroke time in light of the repeated failures of the valve to close on the first attempt. In this review, PORC discussed increasing the valve stroke testing to more frequent interval than the 30 days specified by the IST program, but concluded that it was not necessary. The overall recommendations were similar to the site engineering recommendations discussed in the above paragraph.

The inspector determined that the licensee did not establish an adequate bases for operability and failed to fully address the required closing time specified in technical specifications. The inspector questioned the ability of the valve to consistently meet the required closure time in light of the valves performance history coupled with the lack of a definitive root cause. The concern was discussed with the plant manager.

As a result of the NRC concern, additional engineering evaluations were performed and the PORC members reconvened to further address the bases for operability. The subsequent engineering assessment concluded that stroking this valve more frequently than 30 days would not damage the valve but also that the valve was fully operable in the current condition. At the conclusion of the management meeting the PORC members determined that stroking of the HPCI exhaust more frequently than a 30 day interval was acceptable. Ultimately the licensee determined that stroking the valve at a more frequent interval would be prudent and provide the necessary assurance of valve operability. The valve was declared operable following three successful stroke tests and placed on an increased test frequency of seven days to ensure reliability of this valve.

The inspector determined that the purpose of the safety evaluation was to review the impact of delaying further investigation and repairs to the HPCI exhaust valve until the next scheduled refueling outage and the review of procedure changes being implemented to address a failure of the valve to close on the first attempt. However, this was not an appropriate vehicle to address the degraded condition of the HPCI exhaust valve since a there was a technical specification requirement for valve stroke time which was being impacted by the valves performance. Although the engineering assessment and supporting safety evaluation provided a strong safety bases for removal of the stroke time requirement from the technical specifications, the requirements cannot be modified directly or indirectly using the 10 CFR 50.59 process.

The HPCI turbine exhaust valve (HV-055-1F072) had failed five times in the last four years. Three of the five failures have occurred in the last five months. The inspector was concerned regarding the adequacy of the corrective actions implemented during these failures. This issue is unresolved **(URI 97-10-06)** pending the identification of the root cause of the valve failures to close on the initial attempt and the subsequent corrective actions.

c. Conclusions

The engineering assessment and supporting safety evaluation to support operability of the HPCI exhaust valve was inadequate in that it did not address the valve closure time requirements. The plant operations review committee (PORC) approved the safety evaluation, but failed to challenge the engineering assessment discounting the requirement for the valve to close the first time to meet the closure time required by technical specifications in assessing operability. PORC accepted the degraded condition of the valve without having identified the root cause or evaluating the corrective actions to ensure future valve reliability and thereby the ability to meet the required closure time.

The use of a safety evaluation to accept the delay in further investigations and testing of the HPCI exhaust valve, until the next scheduled refueling outage, in effect inappropriately modified the technical specifications required closing time. The use of the safety evaluation in addressing operability was not necessary nor consistent with NRC guidance on operability provided in generic letter 91-18.

E2.2 Emergency Diesel D22 Loss of Control During Monthly Load Test

a. Inspection Scope

On January 7, during the monthly load test of D22 EDG the control room operator was notified by I&C personnel who noticed a change in pitch of the engine as well as the diesel load at 3700 KW. The control room operator found the D22 EDG running at 2800 KW and started to lower the load to 2750 KW. The engine load instantly increased to 3700 KW and the operator could not restore the load to normal. The operator secured the EDG and declared it inoperable. The inspectors reviewed the root cause, corrective actions, and operability determination for the EDG.

b. Observations and Findings

The cross current control relay (CCCR) was found in the de-energized condition and its contacts had high resistance. When energized the CCCR allows the EDG droop circuit to control the loading of the diesel. When the CCCR is de-energized the droop circuit feedback is removed and the EDG will operate in the isochronous mode (will attempt to carry all the loads on the bus). The de-energization of this relay resulted in the EDG loading as it would during an accident. The EDG attempted to carry all the loads on the bus which was in parallel with the grid but was limited by the fuel rack stops.

The root cause of the CCCR relay failure was found to be a high resistance on the relay pin to socket connections. Oxidation was found at the base of the pins, on the portion not coated with solder, which caused intermittent contact and allowed the relay to de-energize. Since another EDG was inoperable for planned maintenance the CCCR from that EDG was also sent out for analysis, which found similar but less severe oxidation. The CCCR relays for both EDGs were replaced. At the end of the inspection period the licensee was still evaluating the cause of the oxidation and possible corrective measures. The inspector determined that although the root cause for this event appears to have been adequately identified, there was no cause and effect analysis documented and the as-found conditions were not documented in detail in the work order.

PECO inspected areas that could have been over-stressed during the overpower event including the upper and lower piston rings, the connecting rod bearings, and thrust measurements of both turbo chargers. No excessive wear or damage was identified and the turbo charger tolerances were within specifications. The review of the generator performance during this overpower condition concluded that the generator sizing was adequate to support the increase load without degradation.

The inspector also reviewed the maintenance rule (MR) failure analysis. The required function of the EDG is to supply AC power to the appropriate safeguards bus in the event of a loss of offsite power with and without a coincident loss of coolant accident. For these conditions the EDG starts with the governor control in isochronous mode in which case the CCCR relay is not energized. Since the CCCR relay is not required to energize for the safety related function of the EDG, this failure would not have prevented EDG from starting and loading as required by plant analysis. The licensee correctly evaluated this failure to not be a maintenance rule functional failure.

c. <u>Conclusions</u>

The organization response to this event was good particularly since another EDG was also inoperable for planned maintenance and was competing for the same personnel resources. The D22 EDG was returned to an operable status in about two and a half days following a thorough assessment of the overpower event which including a variety of followup inspections and measurements. The root cause analysis of this event was adequate; however, documentation weaknesses were noted including the as-found conditions not being documented in detail in the work order.

E6 Engineering Organization and Administration

E6.1 Main Safety Relief Valve Commitment Change

a. <u>Inspection Scope (71707)</u>

The inspector reviewed a commitment change regarding the threshold for the licensees actions related to leaking main safety relief valves (MSRV). In two letters dated October 6, 1995 and March 1, 1996, between PECO Energy and the NRC, the licensee committed to an action plan to address main steam safety relief valve leakage. This commitment was a result of an inadvertent opening of an MSRV as a result of degradation from prolonged pilot valve leakage (see Resident Inspection Report 50-352/353 95-81).

b. Observations and Findings

On October 15, 1997, the licensee forwarded a letter to the NRC which stated, "the purpose of this letter is to inform the NRC of a change to the commitment for MSRV leakage action plan only." The letter discussed the revision of the temperature and leakage parameter values for monitoring and performance of an operability assessment including the bases for these changes. An overview of the revised action plan was provided as an attachment.

PECO submitted the commitment change per the process as described in procedure LR-C-1, exhibit 4. The inspector determined that the specific commitment change documentation identified that implementation of the change was acceptable and that a 10 CFR 50.59 safety evaluation was not required. Normally, the NRC would be notified of the revised commitment during the next annual summary report; however, PECO elected to notify the NRC prior to the annual summary report due to the previous sensitivity of the issue.

The inspector questioned when the revised commitment would be implemented. An MSRV having an elevated tailpipe temperature already existed at Unit 2. The inspector noted that PECO's actions were as addressed using the originally committed strategy. Three months later, at the end of the inspection period, the inspector noted that PECO had not implemented the revised methodology to address MSRV leakage. The licensee explained that the advance letter was to notify the NRC of the upcoming change, and was not intended to reflect that the change had occurred. PECO plans to implement the MSRV commitment change in the near future and consequently will not revise the letter. In addition, PECO plans to review the procedures and make changes as necessary to ensure that written communications clearly identify the dates by which commitments are expected to become effective if not already implemented.

Although the revised methodology appeared technically sound, based on interviews and a review of the available data, the technical evaluation to support the modification of the commitment was not well documented. Specifically, the evaluation to support revised monitoring and operability strategy was not comprehensive in that it did not correlate the performance data to specific changes in the thresholds values. For example, the operational data used as the bases for the threshold for performing an operability assessment were not delineated. The failure to appropriately detail engineering evaluations creates a vulnerability to subsequent reviews such as plant operations review committee assessments. Based on discussion with engineering management the documentation did not meet their expectations and would be enhanced in the future for similar evaluations. The licensee plans to ensure complete and comprehensive evaluation of a change to an NRC commitment is documented in a single change package.

c. Conclusions

The licensee appropriately implemented the commitment change process for the main safety relief valve commitment change. Although the timing of NRC notification for the change was sooner than required by the process, the letter was misleading in that it implied that the change had been implemented as of the date of the letter, whereas three months later at the end of the inspection period the change had not been implemented. However, no violation of NRC requirements was identified. In addition the engineering evaluation to support the modification of the commitment was not comprehensive in that it did not correlate the performance data to specific changes in the thresholds values.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Internal Exposure Assessment

a. Inspection Scope (83750)

The inspector reviewed the licensee's internal exposure assessment program through a review of positive whole body count measurements and resulting licensee assessments and exposure record documentation. Calibration of whole body counters and measurement capability were also reviewed.

b. Observations and Findings

The inspector determined from a review of approximately 20 positive whole body counts over the previous 18 month period, that approximately 2/3 of these whole body counts had significant unidentified peaks with low error associated with them. It was not apparent that whole body counts indicating unidentified peaks were effectively resolved and dispositioned by the staff, though all were reviewed.

For example, a June 22, 1996 whole body count determined an internal dose of 3.4 mrem, however, the whole body count had an unknown peak that represented 23% of the total counts above background (not including natural radioactivity). This peak may have been cobalt-60 and if it had been properly dispositioned, would have added 12.5 mrem to the internal dose assessment for a total of [illegible] mrem instead of 3.4 mrem.

Whole body counter Quality Control (QC) checks were performed prior to instrument use each day. Cesium-137 and cobalt-60 sources were utilized and the detector performance and trending data were not printed out or otherwise documented. The software program provides notification to the whole body counter operator if the QC check falls outside of three standard deviations of the decay-corrected source activity.

The licensee had appropriate calibrations performed for both Sodium-Iodide (NaI) detector whole body counters in October 1997 that utilized appropriate phantom geometry with National Institute of Standards Technology (NIST) traceable sources. Upon review of the licensee's 10 CFR 61 waste stream analysis results, the inspector compared the principal gamma emitter photon energy peaks for each radio nuclide with the whole body counter peak resolution calibration. Both whole body counters exhibit photon peak resolution of approximately 61 keV in the 800 keV range. The inspector noted that cobalt-58 and manganese-54 have principal gamma photons separated by 24 keV, and that according to the calibration results reviewed, the whole body counters would not be able to accurately determine these two common radio nuclides. The licensee conducted two separate tests with medium and high activity smears from the plant that contained significant quantities of both cobalt-58 and manganese-54. The whole body counter (Accuscan bed counter) identified manganese-54, but failed to identify any cobalt-58 from either test. Other gamma emitters that were identified in the test samples by the chemistry counting laboratory, were also not detected by the whole body counter (zinc-65, chromium-51, and iron-59). Approximately 64% total activity of the gamma emitters was not identified by the whole body counter.

To demonstrate the potential effect, the inspector weighted the percentages of each gamma-emitting isotope by their Annual Limit for Intake for inhalation and determined that the whole body counter identified approximately 85% of the hypothetical internal exposure from the gamma-emitters. Approximately 15% of the internal exposure was not represented. Therefore, if the smears taken by the licensee were indicative of the plant airborne inhalation hazard, the licensee's dose assessments, if based solely on whole body count measurements, may be approximately 15% low.

The inspector reviewed approximately 20 whole body counts that indicated activity above background (and natural radioactivity) and noted the same phenomenon. In addition, from the review of a personnel contamination incident that occurred on August 2, 1991 (documented in Section R8.1 of this report), the radio nuclide Cr-51 was the prominent isotope found in urine samples collected, was detected in nasal smears, and in contaminated clothing samples, however, none of the whole body counts identified this radio nuclide. For that case, the licensee utilized the urine bioassay data to calculate the exposure due to the Cr-51 and added it to the whole body count derived exposure.

The inspector determined that the licensee's program for use of the NaI whole body counters at Limerick did not appear to have sufficient rigor relative to the disposition and assessment of unidentified peaks. Further, it was not apparent that the staff was cognizant of the equipment limitations posed by Sodium-Iodide detectors relative to the effective resolution and identification of all of the detectable radio nuclides that may be common to the plant.

Notwithstanding this weakness, the effectiveness of the contamination control program at Limerick has made it unnecessary for the licensee to document internal exposures of workers. Consequently, weakness in this particular area does not currently effect personnel exposure assessments. The licensee committed to perform further review of this area to ascertain the adequacy of the equipment, procedures, and personnel training in this area.

The licensee utilizes personnel contamination monitors located at the egress from the RCA and from the station protected area, for detecting the presence of internally deposited radio nuclides. The use of these monitors has replaced the use of routine entrance, exit and annual whole body counting of station personnel. The Eberline PM-7 monitors are gamma sensitive plastic scintillator detectors that appears to have, based on currently identified station radio nuclides, the ability to detect approximately 4% of the annual limit of intake (ALI) based on the most restrictive radio nuclide within 7 hours of the intake. This corresponds to an internal exposure screening level of approximately 200 mrem. By procedure, following a PM-7 alarm, contamination frisking and, if necessary, investigative whole body counting is performed in order to quantify internal exposures. Regulations require internal exposure determinations at 10% of an ALI (500 mrem for Limerick Station). No discrepancies were noted.

c. Conclusion

The radiation protection program controls for preventing internal exposures was effective. No significant personnel exposures were apparent. However, the whole body measurement capability appeared to lack sufficient rigor in assuring that all internally deposited radio nuclides, that the whole body counting instrument was expected to detect, were effectively identified and evaluated. It was not apparent that staff were cognizant of the innerent limitations of the equipment relative to discrete resolution of energy peaks to effect radio nuclide identification. The licensee acknowledged the inspection finding and stated their intent to procure a higher resolution whole body counter detector before the next refueling outage.

R1.2 Respiratory Protection

a. Inspection Scope (83750)

The respiratory protection equipment storage and issue controls were reviewed.

b. Observations and Findings

The licensee's respirator processing is provided by a vendor service. The licensee has conducted a vendor QA audit upon initial contracting for this service in 1997. The radiation protection manager (RPM) indicated that periodic audits of this service would be conducted by the RP group to ensure calibrated leak testing of respiratory protection equipment is conducted as required. Proper onsite storage and control of respirators and breathing air bottles was verified. Issuance of respiratory protection is controlled through computer verification of qualifications, which was verified by the inspector. The station service air and Eagle air compressor (utilized for filling air bottles) air quality had been tested quarterly and met Grade E quality standards (as defined by the Compressed Gas Association). All inspected areas of the respiratory protection program met regulatory requirements.

c. Conclusions

The respiratory protection program met regulatory requirements.

R1.3 Counting Laboratory Calibrations

a. Inspection Scope (83750)

The inspector reviewed the licensee's air sample counting laboratory instrument calibration and QC response check program with respect to regulatory requirements and industry standards. This review consisted of laboratory counting geometry observations, review of calibration and detector response check documentation, and interviews with applicable licensee staff.

b. Observations and Findings

The inspector reviewed the calibration data for two gas flow proportional counters and four germanium detectors that are utilized for counting air samples as well as chemistry samples. The calibration data indicated that appropriate voltage plateaus and counting efficiencies had been determined utilizing NIST traceable sources following correct methods. QC response checks for all the above counting instruments were kept up to date and provided the appropriate trending data of detector performance.

The inspector reviewed the most recent 10 CFR 61 radio nuclide analysis results for the dry active waste-stream, which represents average plant contamination, and therefore, airborne contamination. Using this information, the inspector determined that approximately 6.3% of the total activity consisted of non-gamma emitting radio nuclides that were not measurable by the germanium detector counting equipment with respect to average plant contamination. By reference to 10 CFR 20, Appendix B, the missing activity would account for approximately 2.7% of Derived Air Concentration (DAC) measurements. Although this is a relatively low amount, the licensee does not have a criteria for including non-gamma radio nuclides into DAC evaluations. The RPM indicated that this issue would be reviewed.

c. Conclusion

The air sample counting laboratory provided properly calibrated and reliable sample analysis services.

R1.4 Release of Material from Turbine Building Roof

The licensee began replacing the turbine building and control structure roofs in August 1997 and work was in progress at the time of this inspection. The licensee had taken numerous core samples and found trace contamination in three samples of the outer rock layer from the control structure roof centerline while all other samples did not indicate any measurable activity. Approximately 8 drums of rocks were collected from the control structure roof to be shipped to a radwaste processing vendor. All other roof material was free released and disposed of in a conventional local landfill. The inspector reviewed the licensee's sampling plans that included 188 core and rock samples and determined that a good systematic sample plan had been conducted. The inspector reviewed the sensitivity of sample counting. The licensee utilized the Offsite Dose Calculation Manual (ODCM) to establish the counting sensitivity at the environmental lower limits of detection (LLDs).

Selected roof sample analysis results were reviewed and the inspector verified that for the roof materials released for unrestricted use, no radioactivity was detected in those samples and they were adequately counted to the environmental LLD sensitivity level as specified in table 13.4-3 of the OCDM.

R1.5 Instrumentation Calibration

a. Inspection Scope (83750)

The inspector reviewed the licensee's portable radiation detection instrumentation and dosimetry calibration program through a review of plant radiation characterization; source selection and instrument calibration; and instrument calibration methodology and instrument calibration records. This review included calibration laboratory observations, instrument calibration record review, and interviews with plant staff.

b. Observations and Findings

Through a review of April 1997 in situ gamma scans of plant piping and a review of the most recent waste stream characterization data, the inspector determined the average gamma and beta energies at Limerick Station to be 1.2 MeV and 100 keV, respectively. The inspector reviewed the instrument calibration sources and determined that the Tc-99 beta source was appropriate for the beta spectrum in the plant, however, the Cs-137 source, at 662 keV, was a calibration source that was almost half of the average gamma energy found in the plant. The inspector determined that the licensee's calibration methodology did not correct for this difference in gamma energy. By reference to instrument vendor information for two of the most common portable radiation detection instruments utilized at Limerick (Eberline RO-2, E-530), the response in the field would be expected to be 2-5% higher than actual. Though this is a minor error in the conservative direction, the inspector noted that the licensee's process compensated for other errors, such as temperature and pressure, that had a more minor effect on instrument accuracy.

Other minor discrepancies included:

- Source-to-instrument distances needed for calibration were not determined prior to source calibration. Consequently, during instrument calibration, dose rate values needed to be interpolated between values, which may introduce a minor, but unnecessary, calibration error.

- The vendor software program that provides decay corrected source calibration tables of dose rate versus distance for each source attenuator was not inputted with the current NIST traceable source calibration values. Accordingly, a minor error may be included into the instrument calibration target values.

The RPM indicated that these source calibration discrepancies would be reviewed and action taken as necessary to assure the accuracy of the RP instrument calibration program.

The inspector reviewed calibration documentation records for selected RP instruments that were available for issue and determined that all were properly calibrated within the required time period. The inspector also verified proper locked storage of calibration sources and that the source calibrator interlocks were in proper operating condition to prevent inadvertent exposure to personnel.

A review of the Rados Rad-51 electronic dosimeter calibrations and National Voluntary Laboratory Accreditation Program (NVLAP) testing results indicated appropriate calibration techniques and calibration frequencies were met and that the electronic dosimeter demonstrates a positive 11% bias in the normal gamma energy range of the plant and a positive 8% bias for high energy gammas associated with N-16 decay that might be encountered during personnel entries at power. The positive bias is desirable to ensure conservatism in the exposure control program relative to later TLD processing and record exposure determinations No discrepancies were noted.

A review of TLD processing quality controls were found to be excellent and well managed. After changing to (ICN), as a new TLD processing vendor in early 1997, quality control badge processing results indicated combined bias and standard deviation values approaching NVLAP limits. Both Peach Bottom and Limerick RP staffs actively pursued the issue with ICN, which resulted in new Thermoluminescent Dosimeter (TLD) calibration factors for each TLD and resulting improved performance. Each calendar quarter Limerick and Peach Bottom alternate sending spiked quality control TLD badges for testing of the vendor's TLD processing capability. The TLD vendor maintains current NVLAP accreditation as required.

c. Conclusion

The inspector determined that the licensee's radiation protection calibration program utilized sound principles, however, minor discrepancies in the instrument calibration process had the potential to introduce unnecessary errors. Notwithstanding, the instrument calibration area was determined to be adequate. The TLD program oversight was very effective in enhancing the accuracy of vendor TLD processing results.

R2 Status of RP&C facilities and Equipment

During this inspection, the inspector conducted numerous tours of the plant during operating conditions. The licensee made frequent use of radiation dose rate postings and posting sources of radiation postings in applicable areas, which were excellent practices. All radiological postings and locked areas met regulatory requirements and areas were generally clear of unnecessary equipment, well illuminated and generally free of safety hazards. One exception, an abandoned-in-place post-accident sample skid, was noted in Unit 1 Reactor Building, Room 501.

R4 Staff Knowledge and Performance in RP&C

R4.1 Exposure Discrepancy Reports

a. Inspection Scope (83750)

The inspector reviewed the disposition of external exposure discrepancies for the third quarter of 1997 that indicated exposure differences between individual's quarterly TLD results and quarterly electronic dosimeter (ED) exposure results.

b. Observations and Findings

The inspector observed that there were a large proportion of exposure discrepancies derived from individuals making roof repairs from the turbine and control structure roofs. Upon review of several of the roofers exposure discrepancy reports, the inspector observed that all of them showed higher ED results, i.e., between 27% and 111% greater than TLD results for the same time period. In all cases, the personnel exposures were well below regulatory limits.

All of the subject exposure discrepancy reports assigned the lower TLD results rather than the more conservative decision to assign the electronic dosimeter results in the personnel exposure records. The reasons stated in the individuals' personnel exposure records were nonspecific, but indicated that degraded N-16 gamma photons and electro-magnetic field (EMF) interference could have caused the discrepancy and that surveys of the roof confirmed the TLD results.

Expecting that EMF radiation may be responsible, the licensee conducted a EMF survey but did not detect any EMF fields. The inspector's review of the ED calibration testing indicated a relatively accurate response in the N-16 gamma energy range. At the time of the inspection, the licensee was still attempting to test the EDs response to cellular phone broadcast interference, but no evidence had been uncovered that would explain the exposure discrepancy results for the roofers.

The inspector identified that this area will be followed to ensure the adequacy of the licensee's process for evaluating personnel dosimeter result discrepancies.
(IFI 50-352,353/97-10-07).

c. Conclusion

Several exposure discrepancies between TLD and electronic dosimeter quarterly results were resolved but the adequacy of their disposition requires further review.

R5 Staff Training and Qualification in RP&C

R5.1 RP Technician Training and Qualifications

a. Inspection Scope (83750)

The inspector reviewed the RP technician training program, reviewed selected RP technician qualifications with respect to TS 6.3 requirements, and reviewed the control of RP work task assignments to only qualified individuals.

b. Observations and Findings

During 1997 there were three individuals that completed the initial qualifications for Level II (senior) RP technician. Currently all RP technicians at Limerick Station are fully qualified Level II RP technicians. The inspector reviewed the initial RP technician training program and determined that it was comprehensive including sufficient classroom study and job performance evaluations prior to qualifications.

The inspector determined that the licensee had an adequate process for reviewing staff qualification signoffs prior to assigning staff duties. At the principal radiological controlled area (RCA) access point (41-line), an RP technician qualification matrix is printed out weekly and made available for first line supervisor use in assigning only qualified staff to perform tasks. By licensee procedure (TQ-C-7), it is the supervisor's responsibility to ensure staff are not assigned to perform work they are not qualified to perform.

The Radiation Protection (RP) technician continuing training was found to be adequately implemented. In February 1997 the licensee administered an RP fundamentals examination to 36 permanent RP technicians. The results were poor. The licensee provided remedial training and testing and the results improved to an adequate level. The licensee is aware of the RP fundamentals training weakness and is working to improve the level of RP technician knowledge in this area.

c. Conclusions

The RP training program was adequate. The licensee has self-identified a weakness in the RP fundamentals training provided to RP technicians in the continuing training program, and has made some progress in addressing this concern. Currently, all RP technicians are fully qualified senior technicians and an active continuing training program and qualification tracking program is in place.

R5.2 Advanced Radiation Worker Program

a. Inspection Scope (83750)

The inspector reviewed the licensee's advanced radiation worker procedure and selected survey results with respect to Technical Specification 6.3.1 requirements.

b. Observations and Findings

The licensee has established a program to qualify experienced radiation workers on certain selected RP tasks traditionally performed by RP technicians. Procedure HP-C-111 requires the advanced radiation worker (ARW) candidates to complete an 8-hour class and successfully pass a job performance measure evaluation to qualify for task-specific RP technician duties. The procedure limits the radiological conditions to less than high radiation areas and less than 50,000 dpm/100cm^2 contamination levels. The procedure indicates that specific task qualifications can only be added with the approval of the RPM.

The inspector reviewed recent surveys completed by several advanced radiation workers and observed that several radwaste technicians that were appropriately qualified ARWs, had surveyed contamination areas after decontamination and based on their surveys, removed postings and released the areas as clean areas without any RP technician supervision or verification. It was not apparent to the inspector, whether the ARWs were within the limited specific task qualification or whether they were exercising broader RP technician skills of judgement as to when an area of the plant should be deposted. Further review of the ARW program is needed to properly evaluate whether a violation of staff qualification requirements has occurred. This is an unresolved item **(URI 97-10-08)**.

c. Conclusions

The licensee has limited procedural controls over the scope of the advanced radiation worker program. Some survey and contamination area control activities have been performed by the advanced radiation workers that involved evaluation and judgement determinations without qualified RP technician supervision. Further investigation in the advanced radiation worker training and performance is needed to determine whether a violation of TS 6.3.1 has occurred.

R5.3 Health Physics Personnel Qualification

a. Inspection Scope

The inspector reviewed the process and controls associated with personnel qualifications with a specific focus on the dosimetry clerk position. In addition, the implementation of controls for the use of unqualified personnel were evaluated.

b. Observations and Findings

The individual selected for review was found to be fully qualified for the position of dosimetry clerk. However, during the review the inspector identified that the individual had performed the duties of dosimetry clerk prior to completing qualification for all tasks. The job functions are typically broken down to a task or series of tasks for the purpose of implementing qualifications. Qualification includes a classroom training session and a subsequent demonstration of task competency during completion of a job performance measure (JPMs). The individual in question had completed all required classroom training but had not performed the required JPMs prior to performing the duties of a dosimetry clerk.

Technical specifications requires that the unit staff training meet or exceed the standards of ANSI/ANS 3.1-1978. The dosimetry clerk position is not specifically addressed in this standard. However, the licensee had established a training and qualification program for this positions that is accredited by the Institute of Nuclear Power. The licensee's program has provisions for the use of unqualified personnel to perform tasks provided that they are appropriately supervised. Procedure TQ-C-7, "On-The-Job Training and Qualification," requires that unqualified personnel may only perform tasks under the direct, continual observation of either a qualified worker or line supervision.

The primary functions of the dosimetry clerk include operation of the whole body counters and an evaluation of the results, performance of respirator fit tests, and issuance of dosimetry. The inspector reviewed the logs and documentation of whole body scans performed by the individual that was not qualified and found that generally there was evidence that a qualified person performed the whole body count reviews or provided supervision to the unqualified individual. Specifically for the sample of documentation reviewed, either a qualified individual counter signed the log sheets or signed the whole body scan results as the reviewer. However, in the case of the other dosimetry clerk tasks there was no evidence of direct supervision. For example, the dosimetry issue log does not contain countersigned entries indicating that the task was supervised.

The most significance of the dosimetry clerk tasks is reviewing of the whole body scans for anomalies. The rest of the tasks were generally found to be of low complexity and low consequence if improperly performed. For example, operation of the respirator fit equipment involve operation of a computer driven test routine which automatically prompts the actions required by the person being tested. An incorrectly performed operational check or test routine would result in a test failure. In the case of issuing dosimetry, this task is administrative in nature and provisions are in place which would likely identify if dosimetry issued was not recorded correctly.

During the records review the inspector identified one instance in which there was no signature for reviewing the results of a whole body count. Following discussion with the inspector the licensee plans to perform a more comprehensive sample of personnel records to determine if a more wide spread problem exists. The licensee plans to sample a minimum of 100 files containing whole body counts to confirm the required reviews were performed and determine any other administrative errors exist.

c. <u>Conclusions</u>

An unqualified person had been assigned to perform tasks which require formal qualification. Generally, there was evidence of direct supervision for the more critical tasks performed by unqualified individual such as the performance and evaluation of whole body counts. However, for administrative tasks, generally there was no recorded evidence of direct supervision as required by the licensees training and qualification procedures. Although, the practice of using unqualified and unsupervised personnel is inconsistent with the licensee's procedure, this was determined not to be a violation of regulatory requirements since the position and job functions are not specifically addressed through the technical specif... ..on requirements for the training of plant staff. However, the failure of the licensee to appropriately control the use of unqualified personnel is of concern since the same procedure control are used to address positions which have specific training requirements.

R7 Quality Assurance in RP&C Activities

a. Inspection Scope (83750)

The inspector reviewed the licensee's quality assurance oversite of the RP program consisting of a review of licensee documents of a recent QA audit, recent QC surveillance, and RP self-assessments.

b. Observations and Findings

The inspector reviewed the report of a Quality Division audit of the RP program that was conducted in March of 1996. The report was detailed and comprehensive. One minor radiation work permit (RWP) discrepancy and some additional training was needed for outage contractors was reported. The inspector noted that Limerick and Peach Bottom Stations provide technical specialists to evaluate each other, but no outside PECO Energy technical specialists were utilized in the independent program reviews.

Since March 1997, there have 16 QC surveillance of the RP program areas that indicated a wide scope of program review and oversight.

The RP Section provides its own self-assessment reviews and the inspector reviewed the September 30, 1997, "Annual Self-Assessment of the RP Section," and found it to represent all of the radiation protection functional areas ... the Station and included many recommendations. This appeared to be a valuable program review.

Other RP Section program reports were also reviewed by the inspector included the Radiation Protection Integrated Program Review and the Limerick Unit 2 fourth Refueling Outage Report.

c. Conclusions

Oversight of the RP program consisted of independent and self-assessments that generally provided for effective insights and recommendations for program improvements, notwithstanding the minor weaknesses in the instrument calibration and bioassay measurement programs that were noted by the inspector.

R8 Miscellaneous RP&C Issues

R8.1 Dose Assessment Review of an August 2, 1991 Contamination Incident

The inspector reviewed a spent RWCU resin personnel contamination incident that occurred on August 2, 1991, where three individuals were contaminated. After repeated decontamination, persistent skin contamination remained on the extremities of the individuals. Multiple whole body counts and urine samples were taken and outside consultants were involved to provide a comprehensive review of bioassay data and to assess the radiation exposures to the affected individuals. Bioassay measurements continued until August 9, 1991, when the contamination levels dropped to below threshold values for all affected individuals. The highest exposed individual was calculated to have received 150 mrem to the skin of the right forearm due to the event. Based on radiochemical laboratory analysis of several urine samples, 3.5 MPC-hours was calculated due to internal exposure. The inspector reviewed the licensee's exposure records and verified that for each of the three individuals, the additional skin of the extremities exposure was recorded, however, no internal exposures were recorded because they were all below procedural and regulatory recording requirements. In 1991, the regulatory limits for the skin of the extremities was 18750 mrem per quarter and the internal exposure recording requirements were greater than 40 MPC-hours per seven consecutive days and a limit of 520 MPC-hours per quarter. Based on the inspector's' review, the licensee provided a comprehensive dose study related to the August 2, 1991 incident; accurately represented the personnel exposures; and was appropriately documented in the individuals' exposure records.

P4 Staff Knowledge and Performance in EP

a. Inspection Scope (82701)

Following an Alert emergency notification or above, the licensee's Emergency Response Procedure (ERP) 600-1, Health Physics Team, Step 3.1, states that six Health Physics (HP) technicians must be onsite within a half-hour and six more within 60 minutes. Following the October 9, 1997 Alert incident, the HP Team Leader identified that he had difficulty in locating 12 qualified HP Technicians and the timeliness of their response was not acceptable. The inspector assessed the licensee's review of these concerns to determine the adequacy of their self assessment and corrective actions.

b. Observations and Findings

The licensee identified three concerns regarding HP emergency response staffing: (1) untimely emergency notification to the HP staff; (2) not staffing the required HP Technician positions in a timely manner; and (3) unavailability of qualified technicians.

With the exception of the HP Team Leader, the HP technicians are not included in the emergency automated dialer callout system and are called by the on-shift technicians following direction from the Team Leader. The first available individual was not contacted until 12:08 a.m., approximately 38 minutes after the ERO was notified by pagers. The inspector reviewed all the sign-in logs and determined that the licensee did not meet the commitments made in ERP-600-1 as stated above. The licensee stated that as a result of this issue they are planning to add the HP technicians to the automated dialer callout system to ensure an immediate and timely response.

HP technician availability was diminished because the licensee had several technicians working at Peach Bottom to assist in their refuel outage. The Radiation Protection Manager is currently working with Peach Bottom management to revise the refuel outage policy to ensure that there will always be an adequate number of HPs available to meet their emergency response commitments. Also, a tracking system is being developed to track all HPs as to where they can be located during off-hours.

The inspector reviewed Procedures, NSC-1.2, HP Technician II Training; LEPP-9500, Emergency Preparedness Training Plan and training records of the individuals that responded to the Alert event and determined that their EP training was current. However, the inspector noted that three of the HP II Technicians did not appear to have completed all the Job Performance Measures (JPMs) tasks as required by HP Procedure NSC-1.2, Section 7.2.2, which states "Emergency Preparedness Training is developed and conducted by the Site Emergency preparedness organization and is provided upon completion of HP Technician II Qualifications." After further review of additional training procedures, the licensee was able to adequately demonstrate that the pertinent JPMs related to emergency response had been completed by the three individuals. However, the licensee recognized that Procedure NSC-1.2 was ambiguously written and clarity and consistency was needed between HP training qualification procedures and the Emergency Preparedness training and qualifications plan.

c. Conclusion

Although, the licensee was not in full compliance with Procedure ERP-600-1, Health Physics Team, they were proactive in identifying the issues and their corrective actions are adequate for preventing recurrence. The inspector also noted that these issues were not identified in previous exercises or drills because the licensee had typically conducted their exercises during working hours in which HP technicians were onsite and available for immediate response. This non-repetitive, licensee identified and corrected violation is being treated as a Non-Cited Violation **(NCV 50-352,353/97-10-09)**, consistent with Section VII.B.1 of the NRC Enforcement Policy.

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of plant management at the conclusion of the inspection on January 28, 1998. The plant manager acknowledged the inspectors' findings. The inspectors asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X2 Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 83750:	Occupational Radiation Exposure
IP 90712:	In-office Review of Written Reports
IP 90713:	Review of Periodic and Special Reports
IP 92904:	Followup - Plant Support
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

NOV 50-352,353/97-10-01	Operations Log Did Not Accurately Reflect Conditions in the Plant. (Section O2.1)
NOV 50-352,353/97-10-02	Adequate Measures Not Established to Assure Design Requirements were Adequately Maintained During HCU On-line Maintenance. (Section M1.4)
NOV 50-352,353/97-10-03	Inadequate Implementation of Locked-Valve Controls. (Section M8.1)
URI 50-352,353/97-10-04	Mis-wired Valve Breaker Circuit and Associated Drawing Issues. (Section E1.1)
URI 50-352,353/97-10-05	Inadequate Testing of Valve Breakers. (Section E1.1)
URI 50-352,353/97-10-06	Unit 1 High Pressure Coolant Injection (HPCI) Turbine Exhaust Valve Failure. (Section E2.1)
IFI 50-352,353/97-10-07	Resolution of non-conservative exposure determinations between TLD and electronic dosimeter results. (Section R4.1)
URI 50-352,353/97-10-08	Determine whether advanced radiation workers that survey and release contamination areas should be qualified RP technicians. (Section R5.2)
NCV 50-352,353/97-10-09	Difficulty in Locating 12 Qualified HP Technicians and the Timeliness of Their response During the October 9, 1997 Alert incident. (Section P4)

<u>Closed</u>

LER 1-97-011	Unit One High Pressure Coolant Injection (HPCI) Turbine Exhaust Valve Failure (E2.1)
URI 97-03-01	Performance of Reactor Enclosure Secondary Containment Integrity Verification. (Section M8.1)

<u>Discussed</u>

None

LIST OF ACRONYMS USED

ALARA	As low as is reasonably achievable
AR	Action Request
AWR	Advanced Radiation Worker
CFR	Code of Federal Regulations
CRS	Control Room Supervisor
DAC	Derived Air Concentration
ED	Electronic dosimeter
EDG	Emergency Diesel Generator
ERO	Emergency Response Organization
ERP	Emergency Response Procedure
EO	Equipment Operator
ESF	Engineered Safety Feature
FIT	Focused Improvement Team
FP	Fire Protection
HCU	Hydraulic Control Units
HEPA	High Efficiency Particulate
HPCI	High Pressure Coolant Injection
IFI	Inspection Follow-up Item
IR	Inspection Report
LCO	Limiting Condition For Operation
LER	Licensee Event Report
LGS	Limerick Generating Station
NaI	Sodium-Iodide
NCR	Non-Conformance Report
NCV	Non-Cited Violation
NED	Nuclear Engineering Department
NIST	National Institute of Standards Technology
NMD	Nuclear Maintenance Division
NRB	Nuclear Review Board
NRC	Nuclear Regulatory Commission
NUPIC	Nuclear Procurement Issues Committee
NVLAP	National Voluntary Laboratory Accreditation Program
ODCM	Offsite Dose Calculation Manual
PCIV	Primary Containment Isolation Valves
PDR	Public Docket Room
PECO	PECO Energy
PEP	Performance Enhancement Process
PORC	Plant Operations Review Committee
QA	Quality Assurance
QC	Quality Control
RCA	Radiological controlled area
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RP&C	Radiological Protection and Chemistry
RP	Radiation Protection

RPM	Radiation Protection Manager
RPS	Reactor Protection System
RWCU	Reactor Water Clean-up
RWP	Radiation Work Permit
SGTS	Standby Gas Treatment System
SSPV	Scram Solenoid Pilot Valve
ST	Surveillance Test
TLD	Thermoluminescent dosimeter
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation