UNITED STATES NUCLEAR REGULATORY COMMISSION

REGION I

Inspection Report Nos.	50-387/94-20; 50-388/94-21
License Nos.	NPF-14; NPF-22
Licensee:	Pennsylvania Power and Light Company 2 North Ninth Street Allentown, Pennsylvania 18101
Facility Name:	Susquehanna Steam Electric Station
Inspection At:	Salem Township, Pennsylvania
Inspection Conducted:	September 5, 1994 - October 10, 1994
Inspectors:	M. Banerjee, Senior Resident Inspector, SSES D. J. Mannai, Resident Inspector, SSES
Approved By:	John White 11/4/94 J. White, Chief Date Reactor Projects Section No. 2A,

Scope: Resident Inspector safety inspections were performed in the areas of plant operations; maintenance and surveillance; engineering; and plant support. Initiatives selected for inspection were security access control, Nuclear System Engineering System Review Meeting, and residual heat removal fuel pool cooling assist mode valve testing.

Findings: Performance during this inspection period is summarized in the Executive Summary. Details are provided in the full inspection report.

EXECUTIVE SUMMARY

Susquehanna Inspection Reports

50-387/94-20; 50-388/94-21

September 6, 1994 - October 10, 1994

Operations

Operations personnel operated the facility safely during routine and non-routine activities. The operators reacted well to a rod position indication system (RPIS) failure and followed the Technical Specification requirements. Section 2.2 pertains. During routine surveillance testing one of the redundant automatic depressurization system (ADS) malfunctioned. Plant personnel responded well to correct the condition. Section 2.3 pertains.

Maintenance/Surveillance

The inspector observed strong maintenance practices during Standby Gas Treatment System post-modification testing and Reactor Core Isolation Cooling Breaker preventive maintenance. Sections 3.1.1 and 3.1.2 pertain. Additionally, station personnel performed Residual Heat Removal (RHR) Fuel Pool Cooling Assist Mode Valve testing successfully. The inspector observed that involved personnel displayed a strong safety perspective during test performance. Section 3.2.2 pertains.

An unplanned half scram was received in the control room when an I&C technician shorted out a fuse in a reactor protection system (RPS) cabinet while performing a fire detector test. PP&L management recognized the event indicated a performance weakness since the fire protection surveillance procedure was not considered an intrusive evolution and the potential impact to RPS was not considered, and less than adequate work practices were utilized. Plant management responded well to the event by emphasizing the weakness and articulating their expectations to the staff. Section 3.2.1 pertains.

Engineering/Technical Support

The licensee evaluated an Engineering Deficiency Report (EDR) initiated in 1990 and concluded that viable failure modes of certain dampers in the standby gas treatment system and reactor building recirculation system will result in higher post-accident offsite doses, although within the regulatory limits. The licensee's interim corrective actions were appropriate. Long-term corrective actions are under consideration.

The licensee identified ten 250V DC Motor Control Center (MCC) breakers where the magnetic trip setpoint ranges were outside the specified setpoint. The GE type TEC Mag-Break circuit breakers are associated with the Units 1 and 2 RCIC, HPCI and RHR systems. The licensee's analysis concluded that, except for the RCIC outboard steam supply isolation valve, the installed breakers would perform their design function. The licensee's periodic system review meetings are considered a strength. Section 4.1 pertains.

Plant Support

The security card reader replacement was an improvement to access control and was implemented well without any perturbation to the Security Data Management System. Section 5.3 pertains.

Licensee management conservatively decided to terminate all liquid effluent releases following a calculated increase in offsite dose. An investigation revealed an error in the calculation. Section 5.4 pertains.

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1. SUMMARY OF FACILITY ACTIVITIES

Susquehanna Unit 1 Summary

Throughout the inspection period, Unit 1 operated at essentially 100% of rated thermal power with the exception of minor power reductions for surveillance testing and reduced electrical demand due to cool weather, and two planned power reductions. The planned downpower to 40% on September 9, was for control rod scram time testing, reactor recirculation motor generator set maintenance, control rod sequence exchange and feedwater heater maintenance. On September 12, the licensee determined a condition outside the design basis of the plant existed when newly postulated single failure modes of the Standby Gas Treatment System (SGTS) were discovered. The licensee made the required NRC notification. Section 4.3 pertains. On September 14 at 7:24 p.m., the Rod Position Indication System (RPIS) failure alarm was received. An investigation showed rod position indication for nine rods was lost. Rod position indication was restored following a multiplexer card replacement. Section 2.2 pertains.

On September 21, the emergency notification system sirens were declared inoperable. The required one hour NRC notification was made. The licensee determined the cause was telephone line problems. The licensee identified the telephone line problem during system testing. The testing frequency was increased previously, as an interim corrective measure, due to recurring phone line problems. The licensee has initiated a corrective action plan to improve system reliability including hardware upgrades. NRC inspection Report 50-387/94-19 reviewed the previous occurrences.

Susquehanna Unit 2 Summary

Unit 2 operated at full power throughout the inspection period with the exception of one planned downpower and minor power reductions for surveillance testing and reduced electrical demand. A downpower to 80% was accomplished to support hydraulic control unit maintenance, associated scram time testing and a control rod sequence exchange. On October 8, during the control rod sequence exchange, operators mispositioned control rod 18-31. Reactor engineering determined no thermal or preconditioning limits were exceeded. The licensee documented the event on Significant Operating Occurrence Report (SOOR) 94-536 and formed an Event Review Team (ERT) to perform a comprehensive evaluation of the event. Section 2.2 pertains.

2. PLANT OPERATIONS (71707, 92901, 93702, 40500)

2.1 Plant Operations Review

The inspectors observed the conduct of plant operations and independently verified that the licensee operated the plant safely and according to station procedures and regulatory requirements. The inspectors conducted regular tours of the following plant areas:

- Control Room
- Control Structure
- Unit 1 and 2 Reactor Buildings
- Unit 1 and 2 Turbine Buildings
- Engineered Safeguards Service Water Pump House
- Emergency Diesel Generator Bays
- Protected Area Perimeter
- Security Facilities

Control room indications and instrumentation were independently observed by NRC inspectors to verify plant conditions were in compliance with station operating procedures and Technical Specifications. Alarms received in the control room were reviewed and discussed with operators. The operators were cognizant of control board and plant conditions. Control room and shift manning were in accordance with Technical Specification requirements.

During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, blocking permits, and bypass logs. The inspector observed plant housekeeping controls including control and storage of flammable material and other potential safety hazards. Posting and control of radiation, high radiation, and contamination areas were appropriate. Workers complied with radiation work permits and appropriately used required personnel monitoring devices.

The inspectors performed 11.5 hours of deep backshift inspections during the period. These deep backshift inspections covered licensee activities between 10:00 p.m. and 6:00 a.m. on weekdays, and weekends and holidays.

2.2 Rod Position Indication Failure

At 7:24 p.m. on September 14, 1994, the Unit 1 control room operators received a "rod position indication system (RPIS) inoperable" alarm. The plant computer showed indeterminate position indication for 9 rods in row 55. The operators followed the appropriate alarm response and off-normal procedures and contacted instrumentation and controls (I&C) department and reactor engineering. The operators entered plant Technical Specification 3.1.3.7. The action statement required the accomplishment of alternative actions, or otherwise required the plant be in hot shutdown within 12 hours.

I&C troubleshooting identified a failed probe multiplexer card in the RPIS logic that was generating false address for a group of other cards. I&C further determined that the condition affected the rod position indication for 137 out of a total of 185 control rods. The defective card was replaced and the RPIS failure alarm was cleared. The rod position indications were declared operable at 11:00 p.m. The licensee documented the event on SOOR 94-510.

The licensee tested the faulty card on the reactor manual control system (RMCS) simulator and determined that the card failed due to a random failure of a logic chip. The licensee considered this particular failure mode as isolated.

The inspector noted during the event, that the operators lost the ability to maneuver the control rods. However, the ability to scram the reactor was not affected by the loss of RPIS. The rod block monitor was unaffected. Also, if the event occurred with reactor power level was below the low power setpoint (LPSP), the rod worth minimizer (RWM) and the rod sequence control system (RSCS) would have generated rod withdrawal and insert blocks, thus preventing control rod motion without position indication. Therefore, the inspector concluded that the impact to actual plant safety was low.

The inspector concluded operators responded well to the event and the licensee's immediate corrective actions were appropriate. At the end of the inspection period the licensee's resolution of the SOOR was ongoing. The inspector will review the SOOR resolution as part of ongoing assessment of reactivity control/inadvertent rod movement issues reported previously in NRC Inspection Report, 50-387/93-19.

2.3 Automatic Depressurization System (ADS) Timer

During the performance of a monthly Unit 2 surveillance test on September 26, the Division 2 ADS timer did not start when the logic was initiated. Each of the two redundant ADS trip logic channels is provided with a 102 second timer. The licensee's troubleshooting indicated that the relay coil associated with the timer did not energize. With the timer inoperable at 8:10 p.m., the licensee entered Technical Specifications 3.3.3 and 3.5.1. After the timer reset switch was cycled, the timer and the associated alarm worked as designed. The surveillance steps involving the timer were reperformed, the ADS declared operable and the licensee exited the LCO at 9:10 p.m. SOOR 94-522 was written to document the problem.

The licensee concluded that the ADS timer reset switch (GE2940) caused the timer malfunction by not making contact. The reset switch is not cycled during the performance of the monthly surveillance, and the cause of its not making contact was not determined. The licensee considered the switch failure as an isolated event. The licensee concluded that although one division of ADS automatic initiation circuit was affected during the event, the other division of ADS actuation circuit was operable. Also, the manual initiation of ADS function was unaffected. Actual impact to plant safety was small. At the end of the inspection period the licensee was continuing the investigation and review. The SOOR resolution remains to be completed.

The inspector agreed with the licensee's safety assessment, but considered the event potentially significant because of the loss of redundancy of the automatic ADS actuation. The inspector concluded that the licensee's immediate response was appropriate, and will assess the licensee's SOOR resolution as part of the routine resident inspection program.

3. MAINTENANCE AND SURVEILLANCE (62703, 61726, 92902, 40500)

3.1 Maintenance Observations

The inspector observed and reviewed selected maintenance activities to determine that the work was conducted in accordance with approved procedures, Technical Specifications, and industry codes or standards. The following items were considered, as applicable, during this review: Limiting Conditions for Operation were met while components or systems were removed from service; required administrative approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and quality control hold points were established where required; functional testing was performed prior to declaring the involved component(s) operable; activities were accomplished by qualified personnel; radiological controls were implemented; fire protection controls were implemented; and the equipment was verified to be properly returned to service.

Maintenance observations and/or reviews included:

-- WA 40742, Emergency Diesel Generator Fuel Oil Storage Tank Ultrasonic Level Switch Cable Replacement, dated September 12, 1994.

-- WA 43238, Standby Gas Treatment System (SGTS) Drain Line Modification, dated September 20.

-- WA 36158, Motor Control Center Breaker Cubicle Inspection for Reactor Core Isolation Cooling RCIC Steam Supply Isolation Valve, dated September 22, 1994.

-- WA 40045, Rewire '1B' Residual Heat Removal Service Water (RHRSW) Pump Breaker Drawout Switch to Separate Class '1E' Electrical Circuit Contracts from Non-1E Electrical Circuit Contacts (Heart Breaker Mod), dated October 1.

-- WA 04390, Install Supply and Return Piping for New Containment Radiation Monitor (CRM) Panel 1C291B, dated October 6.

-- WA 42258, RHRSW Pump Breaker Three Year Switchgear Maintenance, dated October 7.

3.1.1 Standby Gas Treatment System Modification Retest

On September 20, the inspector observed the retest following a portion of the standby gas treatment system drain line modification. The retest was done in accordance with Operating Procedure OP-020-001 Section 3.4, Emergency Initiation of Standby Gas Treatment System. The inspector witnessed operators performing the procedure with the system engineer. The inspector found operators correctly performed the procedure, and that the systems responded per design. The system operability was successfully demonstrated. Good communications and coordination between operators was observed.

3.1.2 Reactor Core Isolation Cooling (RCIC) Breaker PMs

The inspector observed portions of Unit 1 reactor core isolation cooling (RCIC) breaker preventive maintenance for the RCIC steam supply outboard isolation valve and the RCIC steam supply to the turbine valve. Electrical maintenance personnel were observed checking magnetic trip setpoints for the aforementioned breakers. The setpoints were checked in accordance with procedural requirements. The required magnetic trip setpoint for the RCIC outboard steam supply isolation valve could not be obtained. Electrical maintenance brought this deficiency to the attention of the Nuclear Technology Department. Section 4.2 pertains.

The inspector observed strong field supervisory oversight. The supervisor questioned why the breaker for the RCIC steam supply to turbine valve breaker setpoint was not adjusted to the "preferred" setting since the "as found" setpoint was close to the low end of the tolerance band. As a result of the supervisory interaction, the electrician adjusted the setpoint to the "preferred" setting.

3.2 Surveillance Observations

The inspector observed and/or reviewed the following surveillance tests to determine that the following criteria, if applicable to the specific test, were met: the test conformed to Technical Specification requirements; administrative approvals and tagouts were obtained before initiating the surveillance; testing was accomplished by qualified personnel in accordance with an approved procedure; test instrumentation was calibrated; Limiting Conditions for Operations were met; test data was accurate and complete; removal and restoration of the affected components was properly accomplished; test results met Technical Specification and procedural requirements; deficiencies noted were reviewed and appropriately resolved; and the surveillance was completed at the required frequency.

3.2.1 Inadvertent Half Scram Generated from Fire Protection Surveillance

On September 20, 1994, technicians were performing a semi-annual functional check of the fire protection heat detectors located in the reactor protection system (RPS) cabinets in the upper relay room. While the technicians were heating a detector in panel 1C609, the base of the metal enclosed heat gun came in contact with a shorting link for the Unit 1 'A2' manual scram channel, and resulted in a half scram.

The affected fuse was replaced, and the manual scram channel was functionally tested and restored. The metal heat gun was replaced with an insulated heating unit to perform further testing. First line supervision conducted a briefing with the technicians to heighten their sensitivity to the potential plant impact before testing resumed. SOOR 94-516 was generated.

The licensee management determined that the plant staff considered the surveillance as a no-risk, non-intrusive evolution since it affected only fire protection equipment. Accordingly, more formal controls and work practices were not utilized. The licensee also identified that an ongoing surveillance for the high drywell pressure instrumentation also generated planned half scrams within short time intervals of the inadvertent half scram signal, which would have resulted in a plant scram under different circumstances. The licensee established a group, including the involved I&C technicians and a human factor engineer, to review the performance deficiencies revealed by the event. At the plant morning status meeting the next day, the inspector observed that management clearly articulated their expectations regarding the performance weaknesses.

The group reviewed 37 SOORs involving human performance deficiencies generated since 1990. Their purpose was to determine the underlying causes of the continued human performance errors, a majority of which involved less than adequate work practices. The group concluded that root causes for these errors were not often accurately identified, the corrective actions were less than adequate, and no method for measuring the effectiveness of the corrective actions were implemented. Based on this review, I&C management developed a broad based corrective action plan.

The plan includes better utilization of a process, similar to the employee safety process, that implements the attributes of the licensee's STAR (stop, think, act, review) program and an exchange of good work practice information to avoid human error. A system is being developed that is expected to provide the technicians "lessons learned" from previous performances of the activity, in addition to information that increases awareness of potential risk. The licensee intends to field observation and job performance trending will be used to assess the effectiveness of the program, and expects the results will be discussed with the I&C technicians during monthly meetings.

Increased availability of special tools like insulated heat guns and various human factor enhancements (e.g., labels, color coding) are being implemented. I&C is expected to evaluate the surveillance scheduling process to ensure other similar potentially high risk surveillances are not scheduled within the same time frame. The licensee plans to implement the corrective actions by the first quarter of 1995.

The inspector reviewed the surveillance procedure SI-113-253, "Semiannual Functional Test of Heat and Ionization Detectors for Upper Relay Room Halon System (fire zone 0-27E)", interviewed the I&C technicians and supervision involved with the testing, and inspected the 1C609 RPS cabinet. The inspector noted that the procedure had no precautions or instructions regarding the potential impact to the reactor protection system, but did require using an insulated heat gun. The inspector determined that the event indicated an apparent lack of sensitivity to the potential impact to the safety systems associated with performance of the procedure, but that the immediate actions taken by the licensee were appropriate, and management responded well by vigorously communicating their expectations to plant personnel regarding safe work practices. Management demonstrated a strong safety perspective by iterating their expectation of properly coordinating multiple activities that can generate half scrams. The inspector noted that the licensee did not propose any procedure changes to address the human performance issue, but rather is developing a real time process of input and feedback to the workers to improve performance. The inspector concluded the licensee's corrective actions were comprehensive and appeared to address the human performance weakness identified by the event.

3.2.2 Residual Heat Removal (RHR) Fuel Pool Cooling Assist Mode Valve Testing

The inspector directly observed the residual heat removal (RHR) fuel pool cooling assist skimmer surge tank (SST) suction and fuel pool discharge valve testing on Unit 2. The licensee performed the testing in accordance with TP-235-013. The procedure was performed to assure proper operation of the main suction and discharge valves used during the RHR fuel pool cooling assist mode of operation.

A Nuclear System Engineering (NSE) system engineer was the test director for TP-235-013. The test director conducted a pre-evolution brief in the control room with all personnel involved with the test. The test director discussed prerequisites, precautions, communications, radiological conditions and overview of the test itself. The subject valves were manually stroked successfully with no difficulties.

The inspector determined the procedure was satisfactorily completed. The pre-evolution brief was thorough, comprehensive and well attended. One particular strength of the pre-evolution brief was a discussion by the test director concerning expected system response during valve stroking and a course of action if the system response was not as expected. The inspector considered the communications during the performance of the procedure were good. Personnel involved with the test demonstrated sound knowledge of their roles and responsibilities during the test. The inspector considered the communications during test performance were good. The test procedure was well written, however the inspector noted the procedure did not specifically require monitoring of spent fuel pool temperature during performance of the test. The inspector had no further questions.

4. ENGINEERING (71707, 37551, 92903, 40500)

4.1 Weekly Nuclear System Review Meeting

The inspectors continued to observe the weekly system review meetings. The October 4, 1994 meeting on Containment Instrument Gas (CIG) System was well attended by corporate engineering and site management. The presentation made by the system engineer addressed the pertinent areas of system performance, problem areas, deficiencies and required corrective actions. Active management participation and involvement was observed. The inspector considered the weekly nuclear system review meeting a strength.

4.2 250V Magnetic Trip Breakers

On June 27, 1994 electrical maintenance originated EDR 94-041 which identified three 250V DC MCC breakers with magnetic trip setpoint ranges outside trip setpoints specified by controlled plant drawings. This was discovered during planned breaker preventive maintenance (PM).

The EDR was assigned to Systems Analysis at PP&L Corporate office. Systems Analysis identified ten 250V DC MCCs breakers total in RCIC, HPCI and RHR systems for both units with installed setpoint ranges outside required trip setpoints. The affected breakers are General Electric type TEC Mag-Break circuit breakers which have a fixed range of magnetic trip setpoints. The licensee performed an operability determination which concluded that for each breaker the installed trip setpoint permits passage of inrush current and full load current for the duration of the motor operation, and that it clears an unrestricted fault at the motor terminals.

However, during the Unit 1 RCIC outboard steam supply isolation valve breaker PM on September 22, the breaker was found to have an actual setpoint of 33.8/32.4 amps, while the specified setpoint was 50 amps ± 25%. Of the affected breakers this was the most limiting case. The licensee postulated a reversal during mid-stroke could result in an inrush current of twice the measured value of 17.8 amps. Consequently the breaker could trip in mid travel. Systems Analysis concluded since the valve is normally open, the failure mode is only possible if a valve closure signal (e.g., containment isolation signal) came in during valve stroking while in surveillance testing. Given the normally open valve position, the short valve stroking time of approximately 13.9 seconds, and the vulnerability exists during quarterly surveillance testing, the licensee determined valve was operable. For breakers with trip setpoint ranges above the specified setpoints, Systems Analysis concluded the breaker would trip well below the calculated maximum unrestricted fault current. For one particular valve breaker the continuous rating was below the full load current for the valve motor. The licensee indicated this would not affect operability since the short valve stroke time would not result in a breaker trip. The licensee also indicated the breaker trip set points for all breakers were acceptable for dynamic loads under accident conditions. Nuclear System Engineering (NSE) updated the operability determination on September 22. The licensee is developing a schedule to perform a field walkdown to obtain installed breaker trip setpoint range data for all 250V DC MCCs for comparison with controlled drawing trip setpoint requirements.

The NRC resident and regional specialist inspectors agreed with the licensee's EDR operability determination. The inspector concluded actual safety significance was minimal since the affected breakers were operable, thus capable of performing their intended safety function. Although the EDR was initiated by the maintenance organization in June 1994, the EDR disposition was not communicated to appropriate site personnel by Systems Analysis. (NSE) was not aware of EDR 94-041 nor was electrical maintenance aware of the EDR disposition before the PM was done on September 22 for the RCIC outboard steam supply isolation valve breaker. At the time no action plan to resolve the design issue had been developed nor communicated to site personnel. Thus the licensee's communication and coordination for resolution of this engineering design issue was not fully effective. PP&L management expects the Continuous Process Improvement Program (CPIP) team on corrective actions will encompass this issue.

4.3 Single Failure in SGTS or the HVAC Recirculation System

EDR G00153 issued on August 29, 1990, identified a nonconforming condition related to the standby gas treatment system (SGTS) design basis. It referenced a 1986 non-conformance report on potential failure modes of the outside makeup air dampers and the recirculation discharge dampers that could affect the secondary containment drawdown time and consequently the ability to meet the 10 CFR Part 100 LOCA dose requirements. In September 1991 and May 1993, operability evaluations were performed for the identified potential failure modes of the dampers. The evaluations concluded that a quantitive analysis was required to determine if there are viable failure modes of these dampers and if the actual draw down time would exceed the FSAR assumption of three minutes, thus affecting offsite doses. On May 4, 1993 a responsible engineer was assigned and a closure date of May 20, 1994 was assigned to this EDR.

On September 12, 1994, SOOR 94-505 was initiated following an operability evaluation of the EDR. In this evaluation the outside air makeup damper was postulated to prematurely open during SGTS initiation due to its controller or associated loop failure. A hot short in the control cable of the recirculation fan damper was postulated that could result in the damper failing open without its associated fan running. The licensee consulted the industry failure data in Nuclear Plant Reliability Data System (NPRDS) and concluded that, because there were no alarms in the control room to alert the operator of such failures, these failures could remain undetected for some time period.

The licensee analyzed the effects of these postulated single failures during a LOCA, and concluded that a premature opening of the outside air makeup damper during initiation of SGTS could increase the SGTS drawdown time from 3 to 18 minutes. The opening of the recirculation for damper in the idle line, could decrease reactor building air mixing volume from 50% to 6.5%. The licensee determined the postulated single failures constituted a condition outside the design basis of the plant and was reportable. A one hour NRC notification was made on September 12, 1994. The licensee's offsite dose calculation using the Regulatory Guide 1.3 source terms projected the LOCA 2 hour site boundary thyroid dose of 192 rems for the postulated single failure of makeup air damper controller, and 231 rems for the postulated single failure of the recirculation fan damper. These offsite dose values were an increase in the values reported in the FSAR, but still within the 10 CFR Part 100 limit of 300 rem. The whole body doses for the two failures were similarly higher than the FSAR values, but still within the 10 CFR Part 100 limits.

As an interim corrective measure, the control room operators were instructed to check the control boards, once per shift, to verify that the recirculation fan dampers are not open and that the outside makeup air damper controller did not fail with a high output signal. The monthly surveillance test procedure for SGTS was revised to require the operators to verify that the outside makeup damper opens with a time delay after fan start. PP&L is evaluating permanent corrective actions.

The inspector concluded the licensee's interim corrective actions were appropriate, and verified that they were implemented. Identification of the failure modes is considered a strength. However, the resolution of the EDR was not timely and the licensee's EDR disposition schedule was not commensurate with its potential safety significance. The CPIP team for Corrective Actions is expected to address this issue.

5. PLANT SUPPORT (71750, 71707, 92904, 40500)

5.1 Radiological and Chemistry Controls

During routine tours of both units, the inspectors observed the implementation of selected portions of PP&L's radiological controls program to ensure workers' awareness of radiological conditions, and the utilization of and compliance with radiological work permits (RWPs). The inspectors observed adequate controls of access to various radiologically controlled areas and use of personnel monitors and frisking methods upon exit from these areas. Posting and control of radiation contamination areas, contaminated areas and hot spots, and labelling and control of containers holding radioactive materials were verified to be in accordance with PP&L procedures. Health Physics technician's control and monitoring of these activities was satisfactory. Overall, the inspector observed an acceptable level of performance and implementation of the radiological controls program.

5.2 Security

Implementation of the physical security plan was routinely observed in various plant areas with regard to the following: protected area and vital area barriers were well maintained and not compromised; isolation zones were clear; personnel and vehicles entering and packages being delivered to the protected area were properly searched and access control was in accordance with approved licensee procedures; security access controls to vital areas were maintained and persons in vital areas were authorized; security posts were adequately staffed and equipped, security personnel were alert and knowledgeable regarding position requirements, and written procedures were available; and adequate illumination was maintained. Licensee personnel were observed to be properly implementing and following the physical security plan.

5.3 Security Card Reader Changeover

The security card readers were replaced over the weekend of September 16, 1994. The existing card readers became obsolete and maintenance was increasingly difficult. The modification, done under the replacement item evaluation (RIE) process, added numeric keypads at the plant protected area entrances. The new keypads require input of a four digit code number for entry. The system compares this number with the individual's personnel file in the Security Data Management System, before entry is granted. This enhanced the security controls for preventing unauthorized access. Also replaced were the key cards with individual's picture maintained in a computer file to facilitate generation of replacement cards. The new system is state of the art and is expected to improve system maintenance.

The inspector reviewed the RIE package, which included a 10 CFR 50.59 safety evaluation and concluded that the change improved access control. The inspector noted that the change was implemented without difficulty. The security personnel provided assistance and monitored the use of the new system at the protected area entrances. The inspector concluded the installation of the new card readers was handled well by security and maintenance personnel.

5.4 Liquid Effluent Release

On September 27, 1994, the licensee initiated SOOR 94-523 after the August 1994 composite sample analysis for radioactive liquid releases showed a considerably higher calculated organ dose compared to past results. The increase was attributed to phosphorous P-32. Although the quarterly dose of 5.5 mrem was still well below the Technical Specification 3.11.1.2 limit (5 mrem per reactor unit), it exceeded the licensee's more conservative administrative limit of 5 mrem. Consequently, the licensee stopped all liquid releases until the cause of the increase was understood. The licensee formed an Event Review Team (ERT) to determine root cause and recommend corrective actions.

The licensee requested the laboratory (Teledyne) to analyze the remaining samples for P-32. The reanalysis indicated that actual value of P-32 was below the lower limit of detection (LLD), and that a Teledyne made an error in their initial analysis when they reported a value of P-32 above the LLD. Additionally, the licensee used a bio-accumulation factor (BF) of 100,000 for P-32 from Regulatory Guide 1.109 (1977), while a more recent NRC publication, NUREG CR 1336, issued in 1980, revised the BF to 3,000.

After the determination, the licensee resumed liquid release with continued monitoring for gamma. The offsite dose calculation manual (ODCM) was being revised to reflect a BF of 3,000 for P-32. Additional samples from various process points were sent to Teledyne for P-32 analysis to determine if the liquid effluent sample really needs to be analyzed for P-32. At the end of the inspection period the ERT was finalizing its recommendations regarding corrective actions.

The inspector concluded PP&L management showed a strong safety perspective since they decided to terminate all liquid releases until the cause of the apparent increase in P-32 was understood. The formation of an ERT was considered a strength. The corrective actions taken by the licensee were also appropriate and timely.

6. MANAGEMENT AND EXIT MEETINGS (30702)

6.1 Resident Exit and Periodic Meetings

The inspector discussed the findings of this inspection with PP&L station management throughout the inspection period to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting summarizing the preliminary findings of this inspection. Based on NRC Region I review of this report and discussions held with licensee representatives, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

6.2 Other NRC Activities

On September 7 and 8, 1994, an NRC Region I Emergency Preparedness Specialist conducted an Emergency Preparedness Inspection. Inspection results will be documented in NRC Inspection Report 50-387/94-19; 50-388/94-20.

On September 12 - 16, an NRC Region I Radiation Specialist conducted a new 10 CFR 20 implementation inspection. Inspection results will be documented in NRC Inspection Report 50-387/94-21; 50-388/94-22.

On October 3 - 13, NRC Region I conducted an MOV Inspection. Results will be documented in NRC Inspection Report 50-387/94-14; 50-388/94-15.