July 27, 2004

Mr. Bryce L. Shriver
Senior Vice President and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000387/2004003 AND 05000388/2004003

Dear Mr. Shriver:

On June 30, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Susquehanna Steam Electric Station Units 1 and 2. The enclosed integrated inspection report presents the results of that inspection, which was discussed with Mr. Britt McKinney, Vice President - Nuclear Site Operations and other members of your staff on July 8, 2004.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two self-revealing findings of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs), consistent with Section VI.A of the NRC Enforcement Policy. Additionally, two licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

If you have any questions please contact me at 610-337-5209.

Sincerely,
/RA by
Arthur L. Burritt
Acting for/
Mohamed Shanbaky, Chief
Projects Branch 4
Division of Reactor Projects

Docket Nos. 50-387; 50-388
License Nos. NPF-14, NPF-22

Enclosure: Inspection Report 05000387/2004003 and 05000388/2004003
Attachment: Supplemental Information

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REGION I

Docket Nos.:	05000387, 05000388
License Nos.:	NPF-14, NPF-22
Report No.:	05000387/2004003 and 05000388/2004003
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	769 Salem Boulevard Berwick, PA 18603
Dates:	April 1, 2004 to June 30, 2004
Inspectors:	S. Hansell, Senior Resident Inspector A. Blamey, Senior Resident Inspector J. Richmond, Resident Inspector F. Jaxheimer, Resident Inspector J. Furia, Sr. Health Physicist F. Arner, Senior Reactor Inspector R. Fuhrmeister, Senior Reactor Inspector B. Norris, Senior Reactor Inspector S. Chaudhary, Senior Reactor Inspector D. Werkheiser, Reactor Inspector
Approved by:	Mohamed M. Shanbaky, Chief Projects Branch 4 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000387/2004003, 05000388/2004003; 04/01/2004 - 06/30/2004; Susquehanna Steam Electric Station, Units 1 and 2; Personnel Performance During Non-Routine Plant Evolutions.

The report covered a 3 month period of inspection by resident inspectors and announced inspections by a regional health physicist, regional senior reactor inspectors and a regional reactor inspector. Two Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3 dated July 2000.

A. NRC Identified Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding was identified because PPL did not ensure that the contract workers cleaning the Unit 1 cooling tower maintained the required minimum distance from an energized electrical line. PPL's Safety Operations Safety Rule Book requires a minimum distance of 15 feet 8 inches from an energized 230 KV offsite electrical power line. Subsequently, the worker in a bucket lift contacted the 230 KV line which resulted in the loss of one of two offsite electrical power sources for Unit 2. This resulted in the loss of one of two station startup transformers, T-10. In addition, the loss of T-10 resulted in a loss of condensate transfer keepfill water supply for the Unit 2 "A" and "C" residual heat removal pumps. The pumps were rendered inoperable when the system keepfill pressure dropped below 50 pounds per square inch gauge, the minimum value for pump operability, as required by PPL procedure.

  This finding is more than minor because the deficiency affects the Initiating Events and Mitigating Systems cornerstone attributes related to equipment performance which reduced availability for the T-10 offsite power source and the "A" and "C" residual heat removal pumps. The error adversely affected the objective of the Initiating Events cornerstone to limit the likelihood of those events that upset plant stability such as loss of offsite power. The finding is of very low safety significance because Transformer T-10 was out-of-service for a short period of time (30 hours) and "B" and "D" residual heat removal pumps, as well as remaining containment venting and power conversion systems, were unaffected. Also, the error adversely affected the objective of the Mitigating Systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events to prevent reactor core damage. (Section 1R14.1)

Cornerstone: Mitigation Systems

- Green. A self-revealing finding was identified because PPL did not ensure that the contract workers cleaning the Unit 1 cooling tower maintained the required minimum distance from an energized electrical line as required by PPL's Safety Operations Safety Rule Book. Subsequently, the bucket lift contacted the 230 KV line which resulted in the loss of one of two offsite electrical power sources for Unit 1. Unit 1, shutdown for a refuel and maintenance outage, lost one of two alternate decay heat removal systems that provide cooling for the shutdown reactor fuel.

  This finding is more than minor because it affects the Mitigating Systems cornerstone attributes in that the human performance deficiency led to an actual loss of the Unit 1 fuel pool cooling system. The deficiency resulted in a loss of electrical power to an alternate decay heat removal system (spent fuel pool cooling) for the shutdown Unit 1 reactor. The error adversely affected the objective of the Mitigating Systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events to prevent reactor core damage. The finding is of very low safety significance because the Unit 1 reactor water temperature minimally increased approximately 2 degrees Fahrenheit. (Section 1R14.1)

B. Licensee Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 40A7 of this report.

REPORT DETAILS

Summary of Plant Status

On April 1, 2004, Susquehanna Steam Electric Station (SSES) Unit 1 was in a refueling and maintenance outage. On April 21 during the plant startup and prior to placing the generator online, excessive Main turbine vibrations developed due to a shaft “rub”. A manual scram was initiated at 14.5 percent reactor power and condenser vacuum broken to slow the turbine down. The startup was recommenced on April 22 and the generator brought online on April 23. Full power operation was achieved for Unit 1 on April 28. Unit 1 was operated at or near full power for the remainder of the inspection period, with exceptions for brief power reductions to support control rod pattern adjustments or to support transmission and distribution limitations (minimum generation alerts).

Unit 2 was operating at or near full power at the beginning of the inspection period. On April 28, reactor power was rapidly reduced to 80 percent power due to the loss of a plant equipment electrical transformer. The power loss impacted condenser air removal capability, causing a turbine back pressure increase. Alternate equipment was put in service and full power was restored within 3 hours of the downpower. Unit 2 power was reduced to 65 percent power on May 8, 2004 for a planned control rod sequence exchange and condenser maintenance. Power was restored to 100 percent on May 11, 2004. Unit 2 reactor power was again reduced to 70 percent on June 26 for control rod sequence exchange and condenser maintenance. Power was restored to 100 percent on June 27.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01- 1 Sample)

a. Inspection Scope

The inspectors reviewed PPL’s preparations for hot weather including the actions in the hot weather procedure and a review of open work on the service water system and the reactor building closed cooling water system. The scope included plant walkdowns for selected structures, systems, and components. The walkdowns and reviews were conducted to determine the adequacy of PPL's weather protection activities and system features for prolonged hot weather.

Procedures and Documents

- ON-000-005, Revision 8, “Hot Weather”
- OP-111-001, Revision 18, “Service Water System”
- SO-100-006, “Shiftly Surveillance Operating Log”
- PCWO # 583503, Maintenance Work Order to Remove Floor plugs and install Temporary Air conditioners.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (71111.02 - 20 samples)

a. Inspection Scope

The inspectors reviewed selected safety evaluations associated with the initiating event, mitigating system, and barrier integrity cornerstones to verify changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), were reviewed and documented in accordance with 10CFR50.59. The inspector also determined whether the safety issues pertinent to the changes were properly resolved or adequately addressed. The safety evaluations were completed during the past two years, and were selected based on the safety significance of the changes and the risk to structures, systems and components; the inspectors reviewed seven evaluations. The inspectors also reviewed selected screen-out evaluations for changes and tests for which the licensee determined that safety evaluations were not required; the inspectors reviewed 13 issues that were screened out. This review was performed to verify that the licensee's threshold for performing safety evaluations was consistent with 10CFR50.59. In addition, the inspectors reviewed the licensee's administrative procedures that control the screening, preparation, and issuance of the safety evaluations to ensure that the procedure adequately covered the requirements of 10CFR50.59. The inspectors also reviewed selected Condition Reports (CRs), engineering self-assessments reports, and nuclear oversight audits and surveillances reports associated with the 10CFR50.59 process. This inspection activity represented twenty samples. The documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments

1. Partial System Walkdowns (71111.04Q - 3 samples)

a. Inspection Scope

The inspectors performed partial system walkdowns to verify system and component alignment and to note any discrepancies that would impact system operability. The inspectors verified selected portions of redundant or backup systems or trains were available while certain system components were out of service. The inspectors reviewed selected valve positions, electrical power availability, and the general condition of major system components. This inspection activity represented 3 samples. The walkdowns included the following systems:

- "E", "B", "C" & "D" emergency diesel generators (EDGs) prior to Unit 1 Reactor startup.

- “A”, “B”, & “D” EDGs with “C” EDG inoperable due to a lube oil leak, May 10
- Unit 1 and Unit 2 control rod drive system walkdown, May 13

2. Complete System Walkdowns (71111.04S - 1 sample)

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of offsite power systems including the 230 KV and 500 KV substations and the connections to 13.8 KV station distribution. This inspection activity represented 1 sample. The inspectors used operator rounds and PPL procedures and other documents listed below to verify proper system alignment:

- OP-003-001, “Electrical Plant Lineup”
- OI-098-001, Revision 5, “230KV and 500KV Voltage Schedules”
- FSAR Chapter 8.2 Offsite Electric Power Systems

The inspectors also verified that the documented system configuration existed in the field including design limits such as minimum system voltage for operation (degraded grid setpoints), labeling, remote indication and alarms. The condition and alignment of substation control power supplies (DC protective power and AC control power) were examined and reviewed. Inspectors reviewed the surveillances and other monitoring that is performed on the major system components.

- Drawing E-4, Shts 1& 2, Single line Meter & Relay Diagram 13.8 KV
- Drawing E-1, Sht 1 & 1A, Single line Diagram Station
- AR 523779,
- Calculation EC-004-1019, Minimum System Voltage - Degraded Grid Setpoint
- CR 587628, Offsite power systems (switchyard and 13.8 KV) classified with all non-risk significant maintenance rule functions.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

1. Routine Plant Area Observations (71111.05Q - 6 Samples)

a. Inspection Scope

The inspectors reviewed PPL's fire protection program to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for selected areas. The inspectors walked down those areas to assess PPL’s control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures to assess PPL's fire protection program in those areas. The inspectors reviewed the respective pre-fire action plan procedures for the inspected areas. This inspection activity represented six samples. The inspected areas included:

- ‘E’ Diesel Generator Building, Fire zone 0-41E.
- Control Structure Standby Gas Treatment & Chiller Rooms, Fire Zone 0-30A
- Unit 2 Remote Shutdown Panel Area, Fire Zones 2-2A and 2-2C.
- Station Battery Rooms, FP-013-168 and FP-013-169.
- ESW Pumphouse, FP–013-200 and FP-013-201.
- Control Structure Heating & Ventilation Equipment Rooms, Fire Zones 0-29A to 0-29D

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

1. Internal Flood Protection (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed PPL's internal flooding evaluation, flood mitigation procedures, and design features for the Unit 2 HPCI and RCIC rooms, to verify whether they were consistent with SSES design requirements and industry standards. The inspectors walked down Unit 2 reactor building elevation 645 room flood detectors, watertight doors, sump pumps, and other associated flood protection design features to determine if they were adequate and operable. During the walkdowns, the inspectors also evaluated whether there were any unidentified or unanalyzed sources of flooding, including holes and unsealed penetrations in floors and walls, between flood areas, and between common drain systems and sumps. The inspectors reviewed PPL’s preventative maintenance tasks for room flood detectors, flood barriers, and watertight doors to evaluate whether component functionality was routinely verified. In addition the inspectors reviewed PPL’s corrective action program, including system health reports. This inspection activity represented one sample. The specific procedures and documents reviewed included:

- FSAR Section 9.3.3, "Equipment and Floor Drainage System"
- ON-269-002, "Flooding in the Reactor Building"
- EO-000-104, "Secondary Containment Control"
- Design Basis Document DBD-010, "HELB, MELB, and Internal Flooding"
- NPE-91-001, Section F4, "Individual Plant Examination - Internal Flooding"
- Maintenance Rule Basis Document for Plant Leak Detection System-76D

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

1. Routine Licensed Operator Requalification (71111.11Q -1 Sample)

a. Inspection Scope

On June 8th and 9th, 2004, the inspectors observed licensed operator classroom and simulator training on the oscillation power range monitor system. The inspectors compared the training material to the Technical Specifications revisions, and the modification design basis. The inspectors' evaluation focused on the operating crew's evaluation of plant conditions and the satisfactory completion of crew critical tasks, for the simulated plant conditions. The review included a comparison of the simulator's ability to model the actual plant system performance and the ability to demonstrate industry experience with reactor power oscillations. This inspection activity represented one sample. The following training was observed:

- TM-OP-078J-PG, "Oscillation Power Range Monitor" (Classroom and Simulator)

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

1. Routine Maintenance Effectiveness Observations (71111-EP - 2 Samples)

a. Inspection Scope

The inspectors evaluated PPL's work practices and follow-up corrective actions for selected system, structure, or component (SSC) issues to assess the effectiveness of PPL's maintenance activities. The inspectors reviewed the performance history of those SSCs and assessed PPL's extent of condition determinations for these issues with potential common cause or generic implications to evaluate the adequacy of PPL's corrective actions. The inspectors reviewed PPL's problem identification and resolution actions for these issues to evaluate whether PPL had appropriately monitored, evaluated, and dispositioned the issues in accordance with PPL procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and PPL's corrective actions that were taken or planned, to verify whether the actions were reasonable and appropriate. This inspection activity represented two samples. The following issues were reviewed:

Equipment Issues

- Unit Common off-site power source lost during cooling tower maintenance, CR 561358
- Unit 2 RHR swing bus automatic transfer switch failure, CR 451668

Procedures and Documents

- Maintenance rule bases documents and system health reports for Switchyard, 13.8 KV, 4.16 KV systems, and 480V motor control centers
- Technical Specifications and Bases 3.8.1, "AC Sources - Operating"
- Condition Reports 332609, 344473, 426082, 439521, 451668, 463490, 491482, and 561358

b. Findings

No significant observations or findings were identified.

1R13 Maintenance Risk Assessments & Emergent Work Evaluation (71111.13 - 5 Samples)

a. Inspection Scope

The inspectors reviewed scheduled and emergent work activities with licensed operators and work-coordination personnel to verify whether risk management action threshold levels were correctly identified. In addition, the inspectors compared the assessed risk configuration to the actual plant conditions and any in-progress evolutions or external events to evaluate whether the assessment was accurate, complete, and appropriate for the issue. The inspectors performed control room and field walkdowns to verify whether the compensatory measures identified by the risk assessments were appropriately performed. This inspection activity represented five samples. The selected maintenance activities included:

- Reactor Scram - High Turbine Vibration during plant startup (turbine roll).
- 'C' EDG Fuel injector pump base replacement due to a lube oil leak
- Replace the 13.8 KV Breaker # OA103-06 on Startup Bus 10, both units in 72 hour Tech Spec LCO.
- Troubleshooting and corrective maintenance following failure of 'D' EDG PMT on 6/17/04
- Unit 2, 'A' RHRSW pump and motor replacement, ERMP 511552, PM E0545-51

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14 - 3 Samples)

1. Loss of the 230 KV Offsite Electrical Supply to Startup Transformer T-10

a. Inspection Scope

The inspectors reviewed an unexpected de-energization of Unit 1 230 KV offsite power line on March 21, 2004. A man lift bucket came in contact with the 230 KV line during cooling tower maintenance. The 230 KV line was de-energized by a protective circuit designed to open electrical breakers to isolate the fault. The de-energized 230 KV line resulted in a loss of power to T-10, a Unit 1 startup transformer. Transformer T-10 supplies redundant electrical power for two Unit 1 and two Unit 2 safety related 4.16 KV electrical buses and Unit 1 non-safety related electrical buses. The T-10 startup transformer was out of service for approximately 30 hours while the 230 KV line was being repaired.

The inspectors reviewed the loss of T-10 and it's impact on Unit 1, shutdown for a refuel and maintenance outage, and Unit 2, operating at 100% power. The review focused on PPL's control of the contract workers and PPL electrical safety procedures related to working near the high voltage electrical lines. In addition, the inspectors reviewed operator actions, the plant response after the 230 KV power source was lost, and plant procedures related to the loss of electrical power to Units 1 and 2. This inspection activity represented one sample. The following documents were included in the review:

Procedures and Documents

- Technical Specification (TS) 3.8.1, “Electrical Power Systems; AC Sources-Operating”
- Safety Operations Safety Rule Book
- Technical Specification 3.9.7, “Residual Heat Removal-High Water Level”
- Units 1 and 2 Control Room Operator Logs
- ON-149-001, “Loss of RHR Shutdown Cooling Mode”
- Condition Reports 561358 and 562787

b. Findings

Loss of One Offsite Power Source to Unit 2 (Operating Unit)

Introduction. A Green self-revealing finding was identified because PPL did not ensure that the contract workers cleaning the Unit 1 cooling tower maintained the required minimum distance from an energized electrical line. PPL’s Safety Operations Safety Rule Book requires a minimum distance of 15 feet 8 inches from an energized 230 KV offsite electrical power line. Subsequently, the bucket lift contacted the 230 KV line which resulted in the loss of one of two offsite electrical power sources for Unit 2 and the loss of one of two station startup transformers (T-10). The loss of T-10 led to a loss of condensate transfer keepfill pumps for Units 1 and 2 emergency core cooling systems (ECCS). The Unit 2 “A” and “C” residual heat removal pumps were rendered inoperable when the system keepfill pressure dropped below 50 pounds per square inch gauge (psig), the minimum value for pump operability, as required by PPL procedure.

Description. The inspectors reviewed an unexpected de-energization of a Unit 1 230 KV offsite power line on March 21, 2004. PPL’s Safety Operations Safety Rule Book, Sections 8.1 and 8.2, require the person in charge of a job shall assure that employees maintain a minimum distance of 15 feet 8 inches from an energized 230 KV offsite electrical power line. A man lift bucket, with a contract worker inside, came in contact with an energized 230 KV line during cooling tower maintenance. The 230 KV line was de-energized by a protective circuit designed to open electrical breakers to isolate the fault. The de-energized 230 KV line resulted in a loss of power to T-10, a Unit 1 startup transformer. Transformer T-10 supplies redundant electrical power for two Unit 1 and two Unit 2 safety related 4.16 KV electrical buses and Unit 1 non-safety related electrical buses. The 4.16 KV electrical buses were immediately transferred to the Unit 2 T-20 startup transformer supply. The T-10 startup transformer was out of service for approximately 30 hours while the 230 KV line was being repaired.

Unit 2 was operating at 100% power when the loss of T-10 occurred. PPL entered Technical Specification (TS) 3.8.1, “Electrical Power Systems; AC Sources-Operating.” TS 3.8.1, Condition A, “one offsite circuit inoperable.” TS 3.8.1 required PPL to repair the damaged 230 KV line supplying transformer T-10 within 72 hours.

The loss of T-10 also led to a loss of condensate transfer keepfill pumps for Unit 2 ECCS resulting in a lower pressure than the normal 150 pounds per square inch gauge. The ECCS systems also have a passive keepfill water storage tank that is designed to provide a sufficient amount of water pressure in the system to ensure the pipes are maintained full of water. The Unit 2 “A” and “C” residual heat removal (RHR) keepfill water pressure dropped below 50 psig, the minimum procedure value for pump operability. Operations declared the Unit 2 “A” and “C” RHR pumps inoperable and disabled the pump start capability. The Unit 2 “A” and “C” RHR pumps were unavailable for 2 hours.

Analysis. The finding is a performance deficiency because PPL did not ensure that the contract workers cleaning the Unit 1 cooling tower maintained the minimum distance of 15 feet 8 inches from the energized 230 KV offsite electrical power line. For Unit 2, the deficiency resulted in a loss of one of two required offsite electrical power sources and contributed to the loss of the “A” and “C” RHR pumps. Traditional enforcement does not apply because the issue did not have any actual safety consequence, or potential for impacting the NRC’s regulatory function, and is not the result of any willful violation of NRC requirements.

This finding is more than minor because the deficiency affects the Initiating Events and Mitigating Systems cornerstone attributes related to equipment performance which reduced availability for the T-10 offsite power source and the “A” and “C” RHR pumps. The error adversely affected the objective of the Initiating Events cornerstone to limit the likelihood of those events that upset plant stability such as loss of offsite power. Also, the error adversely affected the objective of the Mitigating Systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events to prevent reactor core damage.

The inspectors conducted a significance determination process (SDP) Phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and determined that the finding degraded both the initiating event and mitigating systems cornerstones. Therefore, an SDP Phase 2 evaluation was required.

The inspectors conducted an SDP Phase 2 evaluation of the risk significance of the performance deficiency and determined that the finding was of very low safety significance (Green). The inspectors used the following assumptions in the Phase 2 evaluation.

- The T-10 startup transformer was out of service for 30 hours; therefore a fault exposure time of less than 3 days was used for the Initiating Event Likelihood.

- The error contributed to a loss of the “A” and “C” RHR pumps and resulted in 2 of 4 RHR pumps being unavailable to provide containment heat removal and low pressure water injection during an event. The remaining equipment for the containment venting and power conversion system was unaffected.

- Credit was given for the recovery of the “A” and “C” residual heat removal pumps.

The dominant accident sequence involved: 1) a loss of turbine building closed cooling water and failure of containment heat removal. The inspectors concluded that the increase in core damage frequency due to this finding was less than 1.0E-7 (Green).

Enforcement. No violation of NRC regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because the PPL Safety Rule Book is not a procedure that is referenced in NRC Regulatory Guide 1.33, Revision 2, February 1978, Appendix A. **(FIN 05000388/2004003-01)**

Loss of One Offsite Power Source to Unit 1 (Outage Unit)

Introduction. A Green self-revealing finding was identified because PPL did not ensure that the contract workers cleaning the Unit 1 cooling tower maintained the required minimum distance from an energized electrical line as required by PPL’s Safety Operations Safety Rule Book. Subsequently, the bucket lift contacted the 230 KV line which resulted in the loss of one of two offsite electrical power sources for Units 1 and 2. Unit 1, shutdown for a refuel and maintenance outage, lost one of two alternate decay heat removal systems that provide cooling for the shutdown reactor fuel.

Description. The inspectors reviewed an unexpected de-energization of a Unit 1 230 KV offsite power line on March 21, 2004. PPL’s Safety Operations Safety Rule Book, Sections 8.1 and 8.2, require the person in charge of a job shall assure that employees maintain a minimum distance of 15 feet 8 inches from an energized 230 KV offsite electrical power line. A man lift bucket, with a contract worker inside, came in contact with an energized 230 KV line during cooling tower maintenance. The 230 KV line was de-energized by a protective circuit designed to open electrical breakers to isolate the fault. The de-energized 230 KV line resulted in a loss of power to T-10, a Unit 1 startup transformer. Transformer T-10 supplies redundant electrical power for two Unit 1 and two Unit 2 safety related 4.16 KV electrical buses and Unit 1 non-safety related electrical buses. The 4.16 KV electrical buses were automatically transferred to the Unit 2 T-20 startup transformer supply. The T-10 startup transformer was out of service for approximately 30 hours while the 230 KV line was being repaired.

The loss of T-10 resulted in a loss of an alternate decay heat removal system on Unit 1. Unit 1 was shutdown for a refuel and maintenance outage with the reactor vessel and cavity flooded with water and connected to the Unit 1 and 2 spent fuel pools. At the time of the event, both loops of Unit 1 residual heat removal (RHR) shutdown cooling were out of service for planned maintenance. The loss of T-10 resulted in a momentary loss of Unit 1 fuel pool cooling system. The Unit 2 fuel pool cooling system remained in service for reactor fuel decay heat removal. The Unit 1 alternate decay heat removal systems were restored within two hours. The Unit 1 reactor water temperature minimally increased by approximately two degrees from 88.9 to 90.6 degrees Fahrenheit. The time to reach a reactor coolant temperature of 212 degrees Fahrenheit was approximately 40 hours.

Analysis. The finding is a performance deficiency because PPL did not ensure that contract workers cleaning the Unit 1 cooling tower maintained the minimum distance of 15 feet 8 inches from the energized 230 KV offsite electrical power line. For Unit 1, the deficiency contributed to a loss of electrical power to an alternate decay heat removal system (spent fuel pool cooling) for the shutdown reactor. Traditional enforcement does not apply because the issue did not have any actual safety consequence, or potential for impacting the NRC's regulatory function, and is not the result of any willful violation of NRC requirements. This finding is more than minor because it affects the Mitigating Systems cornerstone attributes and the human performance error led to an actual loss of the Unit 1 fuel pool cooling system. The error adversely affected the objective of the Mitigating Systems cornerstone to ensure the availability, reliability and capability of systems that respond to initiating events to prevent reactor core damage.

This finding is of very low safety significance (Green) using the NRC IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The event was evaluated against the Appendix G, Table1, "Loss of Control," criteria. The loss of control includes Loss of Thermal Margin (reactor coolant system temperature change that would cause boiling). The loss of thermal margin was not exceeded because the Unit 1 reactor water temperature increased two degrees, during the loss of power, from 88.9 to 90.6 degrees Fahrenheit.

Enforcement. No violation of NRC regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because the PPL Safety Rule Book is not a procedure that is referenced in NRC Regulatory Guide 1.33, Revision 2, February 1978, Appendix A. **(FIN 05000387/2004003-02)**

2. Operator Response - Reactor Scram and Breaking Vacuum

a. Inspection Scope

The inspectors reviewed a manual Scram in response to main turbine bearing high vibrations experienced during startup turbine testing on April 21st at 4:35 pm. The vibration increased to levels warranting a turbine trip as directed by operating procedures. The turbine was manually tripped. The Reactor Protection System was manually initiated and condenser vacuum broken to allow for a more rapid decrease in turbine speed and vibration levels.

All rods fully inserted into the core as designed. The reactor core isolation cooling system was manually initiated to assist with reactor level control and the lowest level reached was approximately 3 inches.

The inspectors reviewed the operator actions, plant response before and after the manual shutdown, and plant procedures related to the main turbine startup testing and RCIC operation. The review focused on the operator performance during the startup testing of the main turbine and the quality of the procedures used to perform the test. Inspectors reviewed the cause determination and the corrective actions that were developed to prevent recurrence. Inspectors observed the implementation of corrective actions that were implemented prior to plant restart. This inspection activity represented the second sample for 71111.14.

b. Findings

No findings of significance were identified.

3. Unusual Event - Unit 2 Reactor Building 13.8 KV Stepdown Transformer Fault Inside the Protected Area

a. Inspection Scope

The inspectors reviewed the plant transient and operator response to stabilize the plant and restore equipment following the explosion at a Load Center located in the Unit 2 Reactor Building on April 28. Reactor power was reduced due to the loss of main condenser vacuum capability. Inspectors also observed licensee personnel initial actions to control the area, preserve evidence, and the interactions between site personnel including operations personnel during the event response. Inspectors verified the event classification. All rods fully inserted into the core as designed. The reactor core isolation cooling system was manually initiated to assist with reactor level control and the lowest level reached was approximately 3 inches.

Inspectors reviewed the cause determination and the corrective actions that were initiated to prevent recurrence. This inspection activity represented the third sample for 71111.14.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111-EP - 6 Samples)

a. Inspection Scope

The inspectors reviewed operability determinations that were selected based on risk insights, to assess the adequacy of the evaluations, the use and control of compensatory measures, and compliance with the Technical Specifications. In addition, the inspectors reviewed the selected operability determinations to verify whether the determinations were performed in accordance with NDAP-QA-0703, "Operability

Assessments." The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report (FSAR), and associated Design Basis Documents as references during these reviews. This inspection activity represented six samples. The issues reviewed included:

- Unit 1 “B” loop RHR pressure increase from 155# to 290#, CR 574028
- “C” Emergency Diesel Generator with lube oil leak in the 1R fuel injector pump, CR 577583
- “E” Emergency Diesel Generator trip after monthly surveillance test due to an air leak, CR 577592
- New MCPR & LHGR Operating Limits needed due to fuel channel bow measurement data, CR 584400
- Common Mode operability determination following failure of “D” EDG to load greater than 3900 KW during post maintenance testing, CR 585913.
- Unit 1 “A” Drywell Floordrain Sump, CR 586544

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (IP 71111.17 - 13 samples)

a. Inspection Scope

The inspectors reviewed selected modification packages for risk significant systems and equipment. The modification packages were selected from the list of those completed within the prior two years. The modifications were distributed among the initiating event, mitigating systems, and barrier integrity cornerstones. The inspectors reviewed the design inputs, assumptions, and design calculations. The inspectors also reviewed changes that were issued during installation to confirm that problems identified during installation were adequately resolved. In addition inspectors reviewed the post-modification testing, functional testing and surveillance testing to determine readiness for operation. The inspectors also reviewed drawings, procedures, design basis documents and relevant sections of the UFSAR to verify that necessary changes had been made. The inspectors walked down accessible portions of the modifications to detect potential abnormal or unexpected installation conditions and to verify that the equipment was actually installed in the location and configuration as documented and analyzed. For modifications which affected plant transient response, the inspectors verified the simulator reflected what was installed in the plant and verified the response of the simulator with respect to the reviewed modifications

The inspectors reviewed selected CRs , self-assessments, and audits associated with the modification process. Specific documents reviewed are listed in the attachment to this report.

The purpose of the reviews was to verify that 1) the design bases, licensing bases, and performance capability of the risk significant structures, systems and components had not been degraded as a result of the modification; 2) modifications performed during increased risk configurations did not place the plant in an unsafe condition. This inspection activity represented thirteen samples.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111-ST - 4 Samples)

1. Routine Post Maintenance Testing Observations

a. Inspection Scope

The inspectors observed portions of post maintenance testing (PMT) activities in the field to determine whether the tests were performed in accordance with the approved procedures. The inspectors assessed the test's adequacy by comparing the test methodology to the scope of maintenance work performed. In addition, the inspectors evaluated the test acceptance criteria to verify whether the test demonstrated that the tested components satisfied the applicable design and licensing bases and the Technical Specification requirements. The inspectors reviewed the recorded test data to determine whether the acceptance criteria were satisfied. This inspection activity represented four samples. The post maintenance testing activities reviewed included:

- Scram time testing Unit 1 control rods after rod drive replacement, SR-155-004
- RCIC Flow Surveillance - PMT after steam leak repairs, 5/3/04
- Unit 1 HPCI Quarterly Flow surveillance after CST instrument calibration and HPCI suction swap from the suppression pool back to the CST, SO-152-002
- Diesel Driven Fire Pump surveillance following engine coolant leak repair, SO-013-001, 6/1/04

b. Findings

No findings of significance were identified.

1R20 Unit 1 Refueling Outage Activities (71111.20 - 1 Sample)

1. Control of Outage Activities

a. Inspection Scope

Configuration Management & Risk Management. The inspectors observed selected portions of maintenance activities, equipment and system operations and restoration, and reviewed selected test procedures. The inspectors monitored the availability of reactor coolant makeup water sources to evaluate whether PPL maintained a defense-in-depth commensurate with the outage risk management goals and in accordance with Technical Specification requirements. The inspectors evaluated selected work activities to ensure the component configuration management, test control, and post maintenance checks were performed in accordance with NRC requirements and approved PPL procedures. In addition, inspectors reviewed unexpected plant conditions, emergent work, and system configuration control during testing and maintenance activities to evaluate whether PPL appropriately identified, assessed, and managed plant risk during those activities. This inspection activity in conjunction with the activities documented in Inspection Report 2004-002 represented one sample for 71111.20. Outage activities were reviewed and documented in the previous inspection report. Additional outage activities that were examined during the reporting period including the following:

- Hydrostatic Testing
- Heatup, Startup and Turbine Testing
- N2J and N1B Recirculation Nozzle Weld Repairs
- Rippled Control Blade and Channel Bow Measurement activities
- Main Steam Isolation Valve Stem Scratches and Gouges due to misalignment.

Procedures and Documents

- PL-NF-02-007, Rev. 14 "Channel Management Plan"
- NDAP-QA-0507, "Conduct of Refuel Floor Operations"
- NDAP-QA-0480, “ASME Section XI System and Component Pressure Testing”
- SE-000-017, “ASME Leak Inspection for Class 1, 2 and 3 Piping”
- Condition Reports (CRs) 554598, 554839, 554957, 556923, 555449 and 558627.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111-ST - 4 Samples)

a. Inspection Scope

The inspectors observed portions of selected surveillance test activities in the control room and in the field and reviewed the test data results. The inspectors compared the test result to the established acceptance criteria and the applicable Technical Specification or Technical Requirements Manual operability and surveillance requirements to evaluate whether the systems were capable of performing their intended safety functions. This inspection activity represented four samples. The observed or reviewed surveillance tests included:

- SE-100-002, “ASME Class I Boundary System Leakage/Hydrostatic Pressure Testing” prior to Unit 1 Reactor Startup
- SE-104-203, “24 Month ESS Bus 1B, 93% Degraded Grid Voltage Timer Reselect Test,” 6/4/04
- SO-152-002, “HPCI Quarterly Flow Surveillance Test,” CR 573645
- SE-104-104, “24 Month ESS Bus 1D, 93% Degraded Grid Voltage Timer Reselect Test,” 6/7/04

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modification (71111.23 - 1 Sample)

a. Inspection Scope

The inspectors reviewed a temporary plant modification to determine whether the temporary change adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the Final Safety Analysis Report (FSAR), Technical Specifications, and assessed the adequacy of the safety determination screenings and evaluations. The inspectors also assessed configuration control of the temporary change by reviewing selected drawings and procedures to verify that appropriate updates had been made. The inspectors compared the actual installation of the temporary modification to determine that the implemented change was consistent with the approved documents. The inspectors reviewed selected post installation test results to verify that the actual impact of the temporary change had been adequately demonstrated by the test. This inspection activity represented one sample. The following temporary modification and documents were included in the review:

- Temporary Modification # 572902, “Gagging Closed PSV-24138.” (CR 572042)

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 1 Sample)

a. Inspection Scope

On May 18, 2004, the inspectors observed PPL’s nuclear emergency response organization (NERO) during an announced emergency preparedness training exercise to evaluate PPL’s NERO performance. The simulated emergency included the activation of the operations support center (OSC), technical support center (TSC), and emergency operations facility (EOF). The control room simulator was used for the exercise.

The inspectors observed the conduct of the exercise in the control room simulator and TSC. The inspectors assessed licenced operator and NERO adherence to emergency plan implementation procedures, and their response to simulated degraded plant conditions to identify weaknesses and deficiencies in classification, notification, and protective actions recommendations. The inspectors observed PPL’s facility critiques to evaluate PPL’s identification of weaknesses and deficiencies and also observed the drill critique presented to management on June 7th. The inspectors compared PPL’s identified findings against the inspectors’ observations to determine whether PPL adequately identified failures. This inspection activity represented one sample. The inspectors’ review included the following documents and procedures:

- Susquehanna Emergency Plan, revision 44
- EP-PS-126, "Control Room Communicator"

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 5 Samples)

a. Inspection Scope

The inspector discussed with the Radiation Protection Manager (RPM) high radiation area and very high radiation area (VHRA) controls and procedures. The inspector verified that any changes to procedures did not substantially reduce the effectiveness and level of worker protection. The controls implemented were compared to those required under plant technical specifications (TS 5.7) and 10 CFR 20, Subpart G, for control of access to high and locked high radiation areas.

The inspector verified adequate posting and locking of entrances to accessible locked high radiation areas and VHRA.

During job performance observations, the inspector observed radiation protection technician performance with respect to radiation protection work requirements. The inspector determined they were aware of the radiological conditions in their workplace and the radiation work permit (RWP) controls/limits, and determined that their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

During job performance observations, the inspector verified the adequacy of radiological controls, such as: required surveys (including system breach radiation, contamination, and airborne surveys), radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls.

For high radiation work areas with significant dose rate gradients (factor of 5 or more), the inspector reviewed the application of dosimetry to effectively monitor exposure to personnel, and verified that controls were adequate. This inspection activity represented five samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 3 Samples)

a. Inspection Scope

The inspector discussed with PPL personnel the 2004 station exposure goal (235 person-rem). The inspector also reviewed the 2004 Unit 1 refueling outage (U113RIO) exposures and goals. The outage exposure was 185.657 person-rem (goal 145 person-rem).

The inspector reviewed the assumptions and basis for the current annual collective exposure estimate. The inspector also reviewed applicable procedures to determine the methodology for estimating work activity-specific exposures and the intended dose outcome.

The inspector reviewed the method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work are encountered. This inspection activity represented three samples.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03 - 1 Sample)

a. Inspection Scope

The inspector reviewed the plant Final Safety Analysis Report (FSAR) to identify applicable radiation monitors associated with transient high and very high radiation areas including those used in remote emergency assessment. This inspection activity represented one sample.

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01 - 10 Samples)

a. Inspection Scope

The inspector reviewed the most current Radiological Effluent Release Report (Susquehanna Steam Electric Station Radioactive Effluent Release Report, January 1, 2003 - December 31, 2003) to verify that the program was implemented as described in Radiological Effluent Technical Specification (RETS)/ODCM; reviewed the report for significant changes to the Offsite Dose Calculation Manual (ODCM) revision dated January 28, 2004 and to radioactive waste system design and operation; determined whether the changes to the ODCM were made in accordance with Regulatory Guide 1.109 and NUREG-0133 and were technically justified and documented; determined whether the modifications made to radioactive waste system design and operation changed the dose consequence to the public; verified that technical and/or 10 CFR 50.59 reviews were performed when required; and, determined whether radioactive liquid and gaseous effluent radiation monitor setpoint calculation methodology changed since completion of the modifications. The inspector determined that anomalous results reported in the current Radiological Effluent Release Report were adequately resolved. The inspector reviewed RETS/ODCM to identify the effluent radiation monitoring systems and its flow measurement devices; reviewed effluent radiological occurrence performance indicator incidents for onsite follow-up; reviewed self assessments, audits, and licensee event reports that involved unanticipated offsite releases of radioactive material; and, reviewed the FSAR description of all radioactive waste systems.

The inspector walked-down the major components of the gaseous and liquid release systems (e.g., radiation and flow monitors, demineralizers and filters, tanks, and vessels) to observe current system configuration with respect to the description in the FSAR, ongoing activities, and equipment material condition.

The inspector observed the routine processing (including sample collection and analysis) and release of radioactive liquid waste to verify that appropriate treatment equipment is used and that radioactive liquid waste is processed and released in accordance with procedure requirements and observed the sampling and compositing of liquid effluent samples. The inspector reviewed several radioactive liquid waste release permits (permits issued for 4 batch releases in May 2004 [04-046 thru 04-049]), including the projected doses to members of the public. The inspector also observed the routine processing (including sample collection and analysis) and release of radioactive gaseous effluent to verify that appropriate treatment equipment is used and that the radioactive gaseous effluent is processed and released in accordance with RETS/ODCM requirements. The inspector reviewed several radioactive gaseous effluent release permits (May 2004 sampling for continuous airborne effluent releases), including the projected doses to members of the public.

The inspector reviewed the records of any abnormal releases or releases made with inoperable effluent radiation monitors and reviewed PPL’s actions for these releases to ensure an adequate defense-in-depth was maintained against an unmonitored, unanticipated release of radioactive material to the environment.

The inspector reviewed changes made to the ODCM as well as to the liquid or gaseous radioactive waste system design, procedures, or operation since the last inspection. For each system modification and each ODCM revision that impacted effluent monitoring or release controls, the inspector reviewed the technical justification and determine whether the changes affect PPL’s ability to maintain effluents ALARA and whether changes made to monitoring instrumentation resulted in a non-representative monitoring of effluents.

The inspector reviewed a selection of monthly, quarterly, and annual dose calculations to ensure that PPL had properly calculated the offsite dose from radiological effluent releases and to determine if any annual TS/ODCM (i.e., Appendix I to 10 CFR Part 50) values were exceeded and, if appropriate, issued a PI report if any quarterly values were exceeded.

The inspector reviewed air cleaning system surveillance test results and the specific methodology to ensure that the system is operating within the PPL's acceptance criteria. The inspector also reviewed surveillance test results and the methodology used to determine the stack and vent flow rates and verified that the flow rates are consistent with RETS/ODCM or FSAR values.

The inspector reviewed records of instrument calibrations performed since the last inspection for each point of discharge effluent radiation monitor and flow measurement device and reviewed any completed system modifications and the current effluent radiation monitor alarm setpoint value for agreement with RETS/ODCM requirements (see Table 1, in the List of Documents Reviewed section, for a listing of monitor and flow measurement devices reviewed). The inspector also reviewed calibration records of radiation measurement (i.e., counting room) instrumentation associated with effluent monitoring and release activities and reviewed quality control records for the radiation measurement instruments (HPGe systems 1-3; liquid scintillation counter 39-03 and 2200 CA).

The inspector reviewed the results of the interlaboratory comparison program to verify the quality of radioactive effluent sample analyses; reviewed the quality control evaluation of the interlaboratory comparison test and associated corrective actions for any deficiencies identified; and reviewed the results from quality assurance (QA) audits and determined that PPL met the requirements of the RETS/ODCM.

The inspector reviewed PPL's Licensee Event Reports, Special Reports, audits, and self assessments related to the RETS/ODCM program performed since the last inspection (QA Audit # 435308; self-assessments CHM-02-05, CHM-03-02, and CHM-03-05). The inspector determined that identified problems were entered into the corrective action program for resolution.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 6 Samples)

a. Inspection Scope

Reactor Safety Indicators

The inspectors reviewed PPL's performance indicator (PI) data, to verify whether the PI data was accurate and complete. The inspectors examined selected samples of PPL's PI data summary reports and compared PPL's PI data to plant records. The inspectors' plant record review included selected control room narrative logs, Technical Specification limiting condition for operation logs, licensee event reports, and condition reports. In addition, the inspectors interviewed the responsible system engineers. The inspectors compared the PI data against the guidance contained in NEI 99-02. This inspection activity represented six samples. The following indicators and PPL documents were included in this review:

Mitigating Systems Cornerstone Performance Indicators

- Units 1 & 2 High Pressure Injection System (HPCI) Unavailability (for the previous 3 quarters, from July 1, 2003 to March 31, 2004)
- Units 1 & 2 Heat Removal System (RCIC) Unavailability (for the previous 3 quarters, from July 1, 2003 to March 31, 2004)
- Units 1 & 2 Safety System Functional Failures (BWR) (for the previous 4 quarters, from April 1, 2003 to March 31, 2004)

PPL Documents

- Units 1 & 2 Control Room Logs
- NDAP-QA-0737, "Regulatory Performance Assessment"
- SO-100/200-006, "Shiftly Surveillance Operating Log"
- Susquehanna Licensee Event Reports for 2003 and 2004
- Nuclear Energy Institute (NEI) 99-02, revision 2, "Regulatory Assessment Performance Indicator Guideline"
- LI-00-018, "Preparation of Performance Indicator Data, NRC Submittals, and Cornerstone Assessment Reports"

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 1 Annual Sample, 1 Semi-annual Sample)

1. Annual Sample Review - Reactor Building Blow-Out Panels

a. Inspection Scope

The inspector reviewed PPL's evaluation and associated corrective actions for selected condition reports (CRs) related to the Reactor Building blowout panels. The issue was selected due to the potential for a generic operability concern based on the increase to blowout panel release pressures due to the historical application of an adhesive caulk. The inspector completed a detailed walkdown of the blowout panels associated with the Unit 1 and 2 Residual Heat Removal (RHR) pump and piping areas and the Unit 2 Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) areas. The inspector reviewed calculations, the Final Safety Analysis Report (FSAR), and Operations Training material to ensure that the elevated blowout panel release pressures had been considered. This inspection activity represented one annual sample. The following documents were included in the review:

Condition Reports, Calculations and Other documents

- Condition Report Numbers 550433, 264271 and Action Request 580714
- EC-012-3094, Rev. 0, "Use Of Caulking On Blowout Panels"
- EC-STRU-1957, Rev. 0, "Reactor Building Wall Evaluations for Pressure Loads"
- EC-STRU-1992, Rev. 0, "Evaluation of Rx. Building and Main Steam Tunnel Blowout Panel Caulking Strength Properties"
- EC-EQQL-0695, Rev. 3, "Determination of Room Pressures and Temperature Response to a HELB, RCIC, HPCI, RHR piping, Reactor Water Cleanup, Reactor Building Main Steam Tunnel"
- FSAR Appendix 3.6.A
- Training Material, TM-OP-034, Secondary Containment"

b. Findings and Observations

Findings

No findings of significance were identified.

Observations

In 1989, PPL identified a nonconforming condition in that caulking had been applied on reactor building blowout panels which inadvertently increased the blowout pressure in some cases by a factor of 2 to 3 over the original design. PPL performed calculations at the time to determine what the new release pressures would be given the adhesive nature of the caulk which had been applied around the perimeter of the panels. They also reviewed the impact on the structural integrity of the various rooms given a high energy line break (HELB). At the time a formal design change was not performed for the increase in the blowout panel design pressure since the recommendation was made to replace the caulk and restore the original design.

The inspector noted that critical key assumptions in a 1989 calculation, EC-012-3094, which determined the caulk strength (caulk thickness and ultimate tensile strength), may have been non-conservative for the Unit 2 HPCI/RCIC panel based on a plant walkdown. PPL engineering personnel stated that the caulk properties such as strength or thickness had never been formally evaluated or validated and that was why the existing plan was to restore the blowout panels to their original design by replacing the caulk. The inspector noted that PPL had previously performed sensitivity studies and determined that a considerable degree of margin existed such that the environmental qualification (EQ) program or structural integrity of the affected areas would not be impacted by the above concern. The inspector determined that while several of the blowout panels had recently been restored to their original design bases, it had been nearly 14 years since the issue was identified and the recommendation was made to remove the adhesive caulk. Additionally, the defacto design change made in 1989 to the panels had never been formally verified or validated in accordance with PPL's design change process. Notwithstanding, the inspector did not find any subsequent PPL analysis which had utilized the original .5 (psi) blowout panel release setpoint as a design input.

The inspector determined that the issue constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” as the restoration back to original design or formal analysis and validation of the change had not been performed for several years. This issue was considered to be minor and not subject to enforcement action when evaluated within the guidelines of Manual Chapter 0612 considering that the functionality of the blowout panels was never challenged based on the available margin. PPL had previously initiated CR 264271 on June 14, 2000, to put in place a final resolution to this issue. The inspector determined that the completed and proposed actions within the CR appeared reasonable and appropriate. PPL has restored several blowout panels and plans are in place to continue this effort.

The inspectors identified a minor issue during their walkdown associated with several broken shear bolts on RHR area blowout panels. PPL initiated CR 580714 to address the nonconforming condition.

2. Occupational Radiation Safety

a. Inspection Scope

The inspector reviewed condition reports related to the problems identified in the radiation protection program issued between February and April 2004. Reviewed condition reports including those related to access controls in high radiation areas <1R/hr that have occurred since the last inspection in this area.

The inspector reviewed condition reports since the last inspection that identified the cause of the event was radiation protection technician error and radiation worker error. The inspector determined there was no observable pattern traceable to a similar cause, and determined this perspective matches the corrective action approach taken by PPL to resolve the reported problems.

The inspector reviewed corrective action program reports related to exposure significant radiological incidents that involved radiation monitoring instrument deficiencies since the last inspection in this area.

b. Findings

No findings of significance were identified.

3. Public Radiation Safety

a. Inspection Scope

The inspector selected issues in the Condition Report (CR) system associated with the RETS/ODCM performance during 2004 for detailed review. The inspector met with the plant chemist to discuss these CRs. These reports were reviewed to ensure that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized.

b. Findings

No findings of significance were identified.

4. Routine PI&R Review

a. Inspection Scope

The inspectors reviewed selected condition reports (CRs), as part of the routine baseline inspection documented in this report. The CRs were assessed to verify whether the full extent of the various issues were adequately identified, appropriate evaluations were performed, and reasonable corrective actions were identified. The inspectors evaluated the CRs against the requirements of NDAP-QA-0702, "Action Request and Condition Report Process," and 10 CFR 50, Appendix B. During this inspection period, the inspectors performed a screening review of each item that PPL entered into their corrective action program, to assess whether there were any unidentified repetitive equipment failures or human performance issues that might warrant additional follow-up.

b. Findings

Within this limited review, no findings of significance were identified.

5. Semi-Annual PI&R Trend Review

a. Inspection Scope

The inspectors reviewed 366 action request (AR) items that were categorized as Management sub type, as part of the semi-annual baseline inspection documented in this report. Fifteen of the ARs were reviewed in detail to verify whether the full extent of the issues were adequately identified, appropriate evaluations were performed, and reasonable corrective actions were identified. The inspectors evaluated the ARs against the requirements of NDAP-QA-0702, "Action Request and Condition Report Process," and 10 CFR 50, Appendix B. The 15 ARs reviewed in detail were: 407895, 446277, 460357, 426969, 505502, 511458, 512650, 516425, 523714, 537496, 542385, 460357, 515519, 516737, and 534579.

Our follow-up review included condition reports (CRs) 460227, 498084, 504149, and 555676. The CR's were all related to the documentation of emergency diesel generator (EDG) problems that occurred in the 2003 and 2004. The CR's documented the station's response to degraded and inoperable EDG events. This inspection activity represented 1 semi-annual PI&R trend review.

b. Findings and Observations

Within this limited review, no findings of significance were identified.

The inspector's review of AR 446277 noted a missed opportunity to improve the station's response to specific weaknesses related to emergency diesel generator (EDG) emergent issues. AR 446277 was initiated on January 22, 2003, as an AR/CR and was changed to an AR-Management by the PPL screening team. The AR included deficiencies in the engineering and station response to two inoperable EDG's and the entrance into a dual unit shutdown Technical Specification Limiting Condition of Operation (LCO). AR 446277 was closed on February 17, 2004, with no action taken due to "other changes have been made diminishing the value of pursuing this AR."

Since the AR closure, additional EDG failures have occurred that included deficiencies in the engineering response to the problems. For example, CR 555676 documented an unplanned emergency start of the "A" EDG on March 7, 2004. The "A" EDG started and energized the 4.16 KV safety related bus due to the normally closed offsite power supply breakers being open for testing. The "A" EDG ran at minimum load for approximately 10 hours due in part to a delay in engineering's response and providing feedback to plant operators related to the stopping the EDG. Plant operators had an approved procedure to restore the EDG from the emergency start condition, but waited to stop the EDG until engineering provided their input.

4OA3 Event Follow-up (71153 - 2 Samples)

1. (Closed) LER 05000387/2002-006-01: Supplemental LER on Failure of Startup Transformer ST-20

On October 3, 2002, the Unit 2 startup transformer (T-20) failed, which resulted in a loss of one of the two Technical Specification required off-site power sources for both Unit 1 and Unit 2. The Unit 2 reactor was manually shutdown due to the loss of both reactor recirculation pumps. The transformer explosion resulted in the declaration of an Unusual Event, the lowest of four emergency classifications.

The inspectors reviewed the event and documented their assessment in NRC Inspection Report 50-387,388/2002-006, Section 1R14.2. The original LER (supplement 0) was reviewed and documented in NRC Inspection Report 50-387,388/2003-002, Section 4OA3.3.

This LER Supplement documented PPL's final conclusions for the root cause of the transformer failure, and the corrective actions taken or planned to prevent recurrence. The inspectors reviewed the supplemental information; no additional findings were noted. This issue was documented in PPL's corrective action program as condition report 426082. This LER is closed.

2. (Closed) LER 05000387/2002-008: Unauthorized Change in Plant Mode

On April 2, 2002 with Unit 1 in Mode 4 at 0% power and scram time testing in progress following the completion of the ASME Class 1 hydrostatic test, reactor coolant temperature reached 202^{o}F. With RCS temperature greater than 200 degrees the temperature requirements for Mode 4 were no longer met and Mode 3 requirements became applicable.

The cause of the event was a lack of recognition as to when the hydrostatic test evolution completed. The initial application of TS 3.10.1, Inservice Leak and Hydrostatic Testing, which allows operation up to 212^{o} F was later determined not applicable to the plant conditions on April 2, 2002 since hydrostatic testing was no longer in progress.

This LER documented PPL's final conclusion for the root cause of the TS violation, and the corrective actions completed to prevent recurrence. The LER also describes the delay (event date versus report date) in processing this event report and corrective actions to prevent recurrence. The inspectors reviewed the LER information and the corrective action documents and no additional findings were noted. The TS violation is documented in PPL's corrective action program as condition report # 395023. The deficiency in a timely conclusion regarding reportability is documented in condition report # 529653. This LER is closed.

4OA5 Other

1. TI 2515/156, Offsite Power System Operational Readiness

Cornerstones: Initiating Events, Mitigating Systems

a. Inspection Scope

The inspector performed Temporary Instruction 2515/156, Offsite Power System Operational Readiness. The inspector collected and reviewed information pertaining to the offsite power system specifically relating to the areas of the maintenance rule (10 CFR 50.65), the station blackout rule (10 CFR 50.63), offsite power operability, and corrective actions. The inspector reviewed this data against the requirements of 10 CFR 50 Appendix A General Design Criterion 17, Electric Power Systems, and Plant Technical Specifications. This information was forwarded to NRR for further review.

a. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On July 8, 2004, the resident inspectors presented the inspection results to Mr. B. McKinney, Vice President - Nuclear Site Operations and other members of your staff, who acknowledged the findings. The inspectors asked PPL whether any material examined during the inspection should be considered proprietary. No proprietary information was identified.

. Annual Assessment Meeting

On April 27, 2004 the NRC held a meeting with PPL, that was open for public observation, to discuss the results of the NRC's assessment of PPL's performance at Susquehanna Steam Electric Station for the period January 1, 2003 through December 31, 2003. The handouts from the meeting are available electronically from the NRC's document system (ADAMS) under accession numbers ML041190634 and ML041210095.

4OA7 Licensee-Identified Violations

The following two violations of very low safety significance (Green) were identified by PPL and are violations of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

High Radiation Area on Unit 2 Turbine Deck - Tech Spec Violation

Plant Technical Specification 5.7.1 requires that areas having dose rates in excess of 100 millirem per hour measured 30 centimeters from the source of radiation be posted, barricaded and access controlled as a high radiation area. Access to, and the activities in, each such area shall be controlled by means of an RWP that includes specification of the radiation dose rates in the immediate work area. On March 22, 2004, PPL determined that workers replacing light bulbs in the ceiling above the Unit 2 turbine deck entered an area in which radiation levels in excess of 100 millirem per hour measured 30 centimeters from the source of radiation existed, but the workers had not been briefed for work in that area. This event is documented in PPL's condition report system as CR-561450.

<u>High Radiation Area in Unit 1 Drywell - Tech Spec Violation</u>

On March 2, 2004, a worker entered the Unit 1 drywell, a posted high radiation area. The worker had not been briefed on the radiological conditions in this area, nor was he logged in on an RWP which allowed for access to high radiation areas. This event is documented in the PPL's condition report system as CR-553890. These findings are of only very low safety significance because they did not involve entry into an area with radiation levels in excess of 1000 millirem per hour measured 30 centimeters from the source of radiation or personnel over-exposure.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
R. Anderson, Vice President - Nuclear Operations
B. McKinney, Vice President - Nuclear Site Operations

Modification and 10 CFR 50.59 Inspection
C. Dodge, Simulator Supervisor
D. Filchner, Senior Engineer
A. Fitch, Assistant Operations Manager - Work Control
J. Helsel Assistant Operations Manager - Training
C. Hess, Simulator Instructor
D. Krupp, Simulator Engineer
E. Miller Licensing Specialist
R. Mullock, Design Engineer
M. Radansky, Senior Engineer
R. Sheranko, Component Engineer
D. Shuey, Design Engineer

Public Radiation Safety
F. Hickey, Nuclear Chemistry Health Physicist
B. Rhoads, Manager - Plant Chemistry
T. Rydzewski, System Engineer
L. Vnuk, Senior Chemist

Occupational Radiation Safety
J. DeMarinis, Health Physicist - ALARA
J. Fritzen, Radiological Operations Supervisor
R. Kessler, Health Physicist - ALARA
C. Madara, Health Physicist - ALARA
V. Schuman, Radiological Operations Supervisor
R. Smith, Radiation Protection Manager
L. Wolf, Health Physics Operations Foreman

TI 2515/156, Offsite Power System Operational Readiness
R. Collier - System Engineer
F. Czysz - Senior Engineer, Plant Analysis
D. Gladey - Senior Engineer, Design Engineering
A. Kissinger- Supervisor Operations Engineering
J. Kocher - Senior Assessor, Corrective Action and Analysis
B. O'Rourke - Senior Engineer, Regulatory Affairs
D. Steffenauer - Unit Coordinator (Work Control)

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000387/2004003-01	FIN	Loss of One Offsite Power Source to Unit 2 (Operating Unit) (Section 1R14.1)
05000388/2004003-02	FIN	Loss of One Offsite Power Source to Unit 1 (Outage Unit) (Section 1R14.1)

Closed

05000387/2002-006-01	LER	Supplemental LER on Failure of Startup Transformer ST-20 (Section 40A3.1)
05000387/2002-008	LER	Unauthorized Change in Plant Mode (Section 40A3.2)

Discussed

None

LIST OF DOCUMENTS REVIEWED

(Not Referenced in the Report)

Table 1
Radiological Effluent Monitors and Flow Rate Meters Reviewed

Effluent Monitors

liquid radwaste discharge radiation monitor
Units 1 and 2 service water radiation monitors
Units 1 and 2 RHR service water radiation monitors
reactor building ventilation radiation monitors (low range noble gas & accident channels)
turbine building ventilation noble gas monitor (low range noble gas & accident channels)
standby gas treatment system radiation monitors (low range noble gas & accident channels)

Flow Measurement Device

liquid radwaste effluent flow monitor
cooling tower discharge flow monitor
reactor building ventilation purge noble gas flow monitor
turbine building ventilation purge noble gas flow monitor
standby gas stack flow and sampler flow rate monitors

TI 2515/156

SO-024-013, Offsite Power Source and Onsite Class 1E Operability Test
OI-AD-029, Emergency Load Control
OI-AD-032, Station Operation Reporting
NDAP-QA-1902, Maintenance Rule Risk Assessment and Management Program
NDAP-QA-0340, Protected Equipment Program
PJM Nuclear Plant Communication Protocol, December 17, 2003
WM-Scheduleing-01, Guideline for Work Scheduling
MR Expert Panel Meeting minutes from April 29, 1998.
LER # 84-013-00
CR# 507062,and CR # 572108
EC-004-1019, Minimum System voltage for Operation

Modification and 10 CFR 50.59 Inspection

Procedures

PPL 50.59 Resource Manual, Rev.2
NDAP-QA-0726, Rev.9, “10 CFR 50.59 and 10 CFR 72.48 Implementation”
NDAP-QA-1202, Rev.8, “Nuclear Department Modification Program”
ODCM-QA-003, Rev.3, “Effluent Monitor Setpoints”
NDAP-QA-0702, Rev. 13, Action Request and Condition Report Process
OP-000-001   Breakers, Revision 9
SSES Emergency Plan Rev.44 (Dec 2003)
Alarm Response to Main Steam Line Hi-Hi Rad in Control Room, Unit 1, June 22 2004
Alarm Response to Main Steam Line Hi-Hi Rad in Control Room, Unit 2, June 22 2004

Permanent Plant Modifications

DCP 294805, Rev.0, Unit 1 Main Steam Line Rad Monitor SCRAM and Isolation Deletion
DCP 294809, Rev.0, Unit 2 Main Steam Line Rad Monitor SCRAM and Isolation Deletion
DCP 93-3027, Rev.1, Replacement of Actuator and Motor on HV155F002, HPCI Turbine Steam Supply Line Inboard Containment Isolation Valve
DCP 99-3035, Rev.0, Replacement of Actuator Motor for HV155F002, HPCI Turbine Steam Supply Line Inboard Containment Isolation Valve
ECO 99-3039C, HWC Dose Mitigation - Snubber Removal, Reactor Vessel (Unit 1).
ECO 99-3039A, HWC Dose Mitigation - Snubber Removal, FWE & RWCU Systems.
ECO 99-3039B, HWC Dose Mitigation - Snubber Removal, Recirc & RHR.

DCP 561773, Unit 1 Reactor Reticulation Nozzle N1B Weld.
ECO 342923, Installation of EGS Style Quick Electrical Disconnect on Inboard and Outboard MSIVs.
ECO 412750, Rev.0, Elimination of HPCI Valve HV255F079 LLRT Failures; Replacement of Actuator Motor to a larger size [2 ft-lbs to 5 ft-lbs]
DCP 93-3070, Rev. 1, HPCI Pump Suction Auto-Transfer to Suppression Pool Logic Elimination - Unit 1
DCP 93-3071, Rev. 1, HPCI Pump Suction Auto-Transfer to Suppression Pool Logic Elimination - Unit 2
DCP 337343, Rev. 0, Control Room Recorder Replacement - Panel 1C601

50.59 Screens
5059-01-0888, DCP 220438, New ECCS Keepfill Piping, Revision 0
5059-01-0997, Install Vent Valve in RHR Division 1, Revision 0
5059-01-1007, Level/Power Control, Revision 2
5059-01-1022, Replacement Fisher Controls Air Regulator, Revision 2
5059-01-1051, Fill and Vent RHR Piping between HV151F010A and HV151F010B (Unit 2 also applicable), Revision 25
5059-01-1076, Elimination of HPCI Valve HV255F079 LLRT Failures, Revision 0
5059-01-1275, 0251578 Is Throttled to Control Level Indication, Revision 0
5059-01-1306, Automatic Depressurization System and Safety Relief Valves, Revision 14/13
5059-01-1331, Unit 2 RWCU System Bypass Valve 244020 Closed Due to Leakby, Revision 0
5059-01-1444, Clarify the Requirements for Surveilling the Fire Detection Instrumentation Located in Inaccessible Fire Zones in the Technical Requirements Manual, Revision 0
5059-01-1546, PCAF to SO-1(2)51-A(B)02, Revision 0
5059-01-1554, Unit 2 RWCU System Bypass Valve 244020 Closed Due to Leakby, Revision 0
5059-01-1630, Supplemental Decay Heat Removal, Revision 6
5059-01-1639, Reactor Vessel/Cavity Flood Up and Let Down During Refueling Outages, Revision 1
5059-01-1727, Miscellaneous FSAR Changes in Response to CRA 484473, Revision 0
5059-01-1775, Revision to TSB 3.3.5.1, Revision 0
5059-01-1844, Use-As-Is Disposition for Rippled D160 Control Blades, Revision 0
5059-01-1849, Clarification of Units1 & 2 TSB 3.3.8.1 Regarding Unit 1 Loss of Power Instrumentation, Revision 0

50.59 Evaluations
220438, Rev. 0, ECCS and RCIC Backup Keepfill System
NL-95-001, Rev. 2, Supplemental Decay Heat Removal Temporary Cooling Equipment
NL-99-046, Rev. 0, Loss of All Decay Heat Removal - EP-DS-005

Audits, Surveillances & Self-Assessments

SO-152-004, “Stroke Time Testing for HV-155F002/CLOSE,” April 04, Dec 03
SE-159-029, Rev.14, “LLRT of Steam to HPCI Turbine Penetration # X-11,” Apr. 29, 2004

SO-252-004, “Stroke Time Testing for HV-255F079/CLOSE,” Apr 03, May 03, Dec 03
SO-252-015, Rev.7, “HPCI RPI Check”, Apr 03
SE-259-098A, Rev.12, “Operability Test HV-255F079”, Apr 03
SI-279-201, Rev.15, “Functional Test – MSL Rad Monitors," Unit 1 Oct. 29-30 2003
SI-279-201, Rev.15, “Functional Test – MSL Rad Monitors," Unit 2 Oct. 27-28 2003
2002-033, Plant Modification Program Audit (April 2002)
434890, Engineering Audit (July - September 2003)
DE-0407, Modification Focused Self-Assessment (April - June 2004)

Corrective Action Documents Reviewed

384717	586464	319165	321287	458100	97-3684
393369	399852	399852	434890	439935	440080
440081	474559	508783	508784	508807	508869
508870	508894	521470	527259	527262	532815
538385	568172	579095	580466	580932	581545
062886					

Work Orders Reviewed

PCWO 480252, Mar 22 2004, “PMT VOTES Testing of HV155F002”

Drawings Reviewed

M-1273, Rev.5, “MOV Detail Data Drawing for HV155F002”
M-1430, Rev.6, “MOV Detail Data Drawing for HV255F079”

Calculations Reviewed

EC-006-0523, Rev.1, “Unit 1 Class 1E 480V AC MISC Calc EH4 Cable Ampacity Calc for MOV HV155F002”
EC-006-0591, Rev.5, “Safety Related 480V Molded Case Breaker and Overload”
EC-049-0677, Rev.0, “Assessment to Ensure the Base Material and Bolt Material are not Overstressing for Specified Values and Determination of Required Bolt Torque Values”
EC-052-0532, Rev.6, “MOV Data Detail Calc for HV255F079”
EC-052-0544, Rev.6, “MOV Data Detail Limit Switch Setting and Torque Switch Settings for HV155F002”
EC-052-0552, Rev.0, “Weak-Link Analysis for HV155F002/F003”
EC-052-0570, Rev.0, “Overthrust (Continuous) Evaluation for HV255F079”
EC-052-1044, Rev.1, “Max Thrust and Seismic Analysis MOV Limiting Component Analysis for HV1(2)55F075(9)
EC-052-1048, Rev.0, “Max Thrust & Seismic Analysis for MOV Limiting Component Analysis for HV1(2)55F002(3)”
EC-052-1054, Rev.0, “Max Thrust and Seismic Analysis for HV255F079”
EC-079-1013, Rev.0, “MSL Rad Monitor High Trip Setpoint Change to Support HWC

Determination of Allowable Value and Analytical Limit for TS Change."
EC-079-1016, Rev.0, "MSL Rad Monitor High Trip and Alarm Setpoints for HWC Operations"
EC-088-0503, Rev.14, "Voltage Drop Calc for GL 89-10 DC MOVs"
EC-088-0504, Rev.10, "250VDC Class 1E Breaker and Overload Calculation"
EC-PIPE-1401, Rev.3, "MSL 'B' Inside Containment"
EC-PIPE-2611, Rev.1, "PSTR HPCI Vacuum Breaker Line"
EC-RADN-1010, Rev.4, "Realistic Dose Analysis for FHA, LOCA, and CRD Accidents FSAR Ch.15"
EC-RADN-1111, Rev.0, "Control Rod Drop Accident Offsite Doses (DBA)"
EC-VALV-0509, Rev.24, "Determination of Limiting Values for Full Stroke Times for Power Operated Valves"
EC-VALV-0570, Rev.11, "Design Basis Development for Priority 2 MOVs"
EC-VALV-1072, Rev.21, "Actuator Sizing & Diagnostic Test Acceptance Criteria for GL 89-10 DC Rising Stem MOVs"
EC-VALV-1073, Rev.20, "Actuator Sizing & Diagnostic Test Acceptance Criteria for GL 89-10 AC Unit 1 Rising Stem MOVs"
M-VLV-271, Rev.6, "MOV Data Detail, Limit Switch Settings and Torque Switch Settings for HV-155F002"
EC-PIPE-16147, Rev. 0, Snubber Elimination-Group IV.
EC-PIPE-16146, Rev. 0, PSTR, PSUP, Unit 1 Snubber Elimination-Group I & II.

Miscellaneous Documents Reviewed

EWR 358962
EWR 420262
EWR 452157
GE Topical Report, NEDO-31400A, Class 1, Oct 1992, "Safety Evaluation for Eliminating The Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor."
NRC Letter dated May 15, 1991, Acceptance for Referencing of Licensing Topical Report NEDO-31400
PICN 442414
PICN 452276
Training Material #EG270, 10CFR50.59 Continuing Training, Revision 1
PPL 50.59 Resource Manual, Revision 2

LIST OF BASELINE INSPECTIONS PERFORMED

7112201	Radiological Effluents Technical Specification	2OS1

LIST OF ACRONYMS

AC	Alternating Current
ADAMS	Agencywide Documents and Management System
AR	Action Request
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CR	Condition Report
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOF	Emergency Operations Facility
EP	Emergency Preparedness
EP-DS	Emergency Plan - Damage Support procedure
FSAR	Final Safety Analysis Report
HELB	High-Energy Line Break
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LHGR	Linear Heat Generation Rate
MCPR	Maximum Critical Power Ratio
MELB	Moderate-Energy Line Break
NCV	Non-cited Violation
NDAP	Nuclear Department Administrative Procedure
NEI	Nuclear Energy Institute
NERO	Nuclear Emergency Response Organization
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ODCM	Offsite Dose Calculation Manual
OSC	Operations Support Center
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post Maintenance Test
PPL	PPL Susquehanna, LLC
PSIG	Per Square Inch Gauge
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specification

RHR	Residual Heat Removal
RPM	Radiation Protection Manager
RWP	Radiation Work Permit
SDP	Significance Determination Process
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
TS	Technical Specification
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
VHRA	Very High Radiation Area
WO	Work Order