July 28, 2006

Mr. Britt T. McKinney
Senior Vice President, and
Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Boulevard - NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000387/2006003 AND 05000388/2006003

Dear Mr. McKinney:

On June 30, 2006, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Susquehanna Steam Electric Station Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on July 14, 2006 with you and other members of your staff.

The 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.

The report documents one NRC-identified finding and four self-revealing findings of very low safety significance (Green). All of the findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs), consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any 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, its enclosure, and your response (if any) 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).

Sincerely,

/RA/

James M. Trapp, Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosures: Inspection Report 05000387/2006003 and 05000388/2006003
w/Attachment: Supplemental Information

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REGION I

Docket Nos.:	50-387, 50-388
License Nos.:	NPF-14, NPF-22
Report No.:	05000387/2006003 and 05000388/2006003
Licensee:	PPL Susquehanna, LLC (PPL)
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	Berwick, Pennsylvania
Dates:	April 1, 2006 through June 30, 2006
Inspectors:	A. Blamey, Senior Resident Inspector F. Jaxheimer, Resident Inspector J. Richmond, Reactor Inspector J. Josey, Reactor Inspector T. Fish, Senior Operations Engineer J. Furia, Senior Health Physicist J. Krafty, Reactor Inspector P. Presby, Operations Engineer L. Scholl, Senior Reactor Engineer
Approved by:	James M. Trapp, Chief Reactor Projects Branch 4 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000387/2006-003, 05000388/2006-003; 04/01/2006 - 06/30/2006; Susquehanna Steam Electric Station, Units 1 and 2; Operability Evaluations, Permanent Plant Modifications, Outage Activities, Identification & Resolution of Problems.

The report covered a 3-month period of inspection by resident inspectors and announced inspections by a regional senior health physicist, senior reactor inspectors, a senior operations engineer, an operations engineer, and reactor inspectors. Five Green findings, all of which were non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 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 and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing non-cited violation was identified for failure to comply with 10 CFR 50 Appendix B, Criterion III, Design Control. PPL did not correctly verify that the Power Range Neutron Monitoring System (PRNMS) modification would not adversely affect the design bases of the reactor protection system. This resulted in a Unit 1 reactor automatic shutdown (scram) on June 15, 2006, when the division II reactor protection system power supply was transferred to the alternate supply. PPL entered the issue into the corrective action program and installed a modification to prevent recurrence.

  The finding was more than minor because the condition affected the Design Control attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The finding is of very low safety significance because all mitigating systems were available and responded appropriately to the reactor scram. This finding is also related to the human performance cross-cutting area because PPL did not ensure supervisory and management oversight of work activities, including contractors such that there would be no adverse system interface issues in the PRNMS design which supports nuclear safety. (Section 1R20.2)

Cornerstone: Mitigating Systems

- Green. A self-revealing non-cited violation was identified for failure to have adequate work instructions in accordance with 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." This resulted in not identifing stem nut degradation prior to the failure of two Unit 1 residual heat removal (RHR) valves. PPL entered the issue into the corrective action program and has replaced the stem nuts on the two failed RHR valves, as well as other valves, that had degraded stem nuts.

The finding was more than minor because the condition affected the Procedure Quality attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The valves failed to stroke during the Spring 2006 refueling outage. The finding is of very low safety significance because the finding was determined to not require a quantitative assessment using Manual Chapter 0609, Appendix G, "Shutdown Operations Significant Determination Process." (Section 1R15)

- <u>Green</u>. A self-revealing non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Action was identified because PPL failed to adequately evaluate and correct degraded material in the "C" Emergency Service Water (ESW) pump breaker that caused a failure on April 5, 2006. PPL's corrective action for this failure included replacing the breaker with a new style breaker.

  The finding was more than minor because the condition affected the Equipment Performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. This finding is of very low safety significance because the finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification Allowed Outage Time, did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk significant per 10 CFR 50.65, for greater than 24 hours, and did not screen as potentially risk significant due to external events. This finding has a PI&R (evaluation) cross-cutting aspect because PPL did not perform a thorough evaluation of the problem so that the resolution addressed causes and extent of condition as necessary to prevent the subsequent failure of the 4Kv breaker due to material degradation.
  (Section 1R17)

- <u>Green</u>. The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action" because PPL did not correct long standing issues related to worker qualifications. This resulted in unqualified workers performing tasks important to safety as described by the Quality Assurance (QA) program. Inspectors observed that over a four year period, PPL took action to reconcile the qualification of the individuals involved in each event. PPL has developed a plan to address this issue and an effectiveness review of the implemented actions is scheduled for November 2006.

  This finding is more than minor because if left uncorrected, the tasks being performed by unqualified workers will become a more significant safety concern. An unqualified worker calibrating safety-related equipment affected the Equipment Performance attribute of the mitigating systems cornerstone and unqualified fire brigade members affect the Protection Against External Factors attribute of the same cornerstone. The finding affects the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events. This finding is of very low safety significance because the work performed by the unqualified individual performing the recirculation flow calibration did not result in a loss of system safety function, and did not represent an actual loss of safety function of any single train of equipment. The Significance Determination Process (SDP), Appendix F, does not specifically address fire brigade issues and allows for management discretion to determine issue significance. This performance issue was reviewed by NRC management and is determined to be a finding of very low safety significance. (4OA2.4)

Cornerstone: Barrier Integrity

- Green. A self-revealing non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for failure to have adequate work instructions prescribed in a maintenance procedure, which resulted in a reactor coolant system mechanical joint leak. PPL entered this condition into the corrective action program and properly reassembled the mechanical joint during the Unit 2 Spring maintenance outage.

  This finding is greater than minor because the condition affected the Procedure Quality attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers (reactor coolant system) protect the public from radionuclide releases caused by accidents or events. The finding was determined to be of very low safety significance because the reactor coolant system leak would not have resulted in exceeding the Technical Specification limit for identified leakage, nor would it have likely effect other mitigation systems resulting in a total loss of their safety function. (Section 1R20.1)

B. Licensee-Identified Violations.

None.

REPORT DETAILS

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 began the inspection period in a planned refueling and maintenance outage. On April 10, 2006, the unit was restarted and achieved full Rated Thermal Power (RTP) on April 15, 2006. The unit continued to operate at or near full RTP until June 15, 2006, when Unit 1 experienced an automatic shutdown while transferring the division II reactor protection system (RPS) power supplies. The unit was restarted and achieved full RTP on June 17, 2006, and remained at full RTP through the end of the inspection period.

Unit 2 began the inspection period at full RTP. On April 28, 2006, the unit was shutdown to correct an unidentified drywell leak. The unit was restarted on May 5, 2006, and achieved full RTP on May 9, 2006. Unit 2 remained at or near full RTP until June 30, 2006, when power was reduced to approximately 70% to perform a control rod sequence exchange.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01- 2 Samples)

.1 Adverse Weather - Seasonal Extremes

a. Inspection Scope

During the week of May 29, 2006, the inspectors reviewed PPL's preparations for extreme hot weather. Plant walkdowns for selected structures, systems and components (SSCs) were performed to determine the adequacy of PPL's weather protection activities. The inspectors also reviewed and evaluated plant conditions related to hot weather and reviewed considerations in PPL's Maintenance Rule station risk assessment. The readiness of the following systems was reviewed.

C Hot weather (95° F), reactor building closed cooling water (RBCCW) loads, reactor feed pump (RFP) coolers, and turbine building closed cooling water (TBCCW).

b. Findings

No findings of significance were identified.

.2 Adverse Weather - Site Readiness

a. Inspection Scope

Northeastern Pennsylvania experienced heavy rain and Susquehanna River flooding during the week of June 26, 2006. The inspectors reviewed PPL's preparations for heavy rain and high river level. This included a tour of the plant protected area, river water makeup pump house, and a review of the dry fuel cask transport path from the Reactor Building to the cask horizontal storage modules. The inspectors also monitored PPL's review of the emergency plan, emergency sirens, and plant staffing. The inspectors reviewed ON-000-002, "Natural Phenomena," and evaluated PPL's risk management actions which were taken in preparation for the heavy rains and flooding. The documents reviewed are listed in the Attachment.

- Heavy rain and Susquehanna River flooding during the week of June 26, 2006.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02 - 22 Samples)

a. Inspection Scope

The inspectors reviewed seven safety evaluations (SEs), all of which were either issued during the past two years or associated with plant modifications that were completed during the past two years. The SEs reviewed were in the Initiating Event, Mitigating Systems, and Barrier Integrity cornerstones. The selected SEs were reviewed to verify that changes to the facility or procedures as described in the Final Safety Analysis Reports (FSAR) were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved or adequately addressed. The reviews also included the verification that PPL had appropriately concluded that the changes and tests could be accomplished without obtaining license amendments.

The inspectors also reviewed fifteen screened-out evaluations for changes, tests and experiments for which PPL determined that SEs were not required. This review was performed to verify that the PPL's threshold for performing SEs was consistent with 10 CFR 50.59.

In addition, the inspectors reviewed the administrative procedures that were used to control the screening, preparation, and issuance of the SEs to ensure that the procedure adequately covered the requirements of 10 CFR 50.59.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04 - 5 Samples)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed partial 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. The walkdowns included the following systems:

- Units 1 and 2, "A," "B," "D" emergency service water (ESW) pumps and power supplies during vacuum breaker modification on the "C" ESW pump
- Units 1 and 2, "E" emergency diesel generator (EDG), when substituted for "C" EDG, due to an ESW leak
- Unit 1, division I residual heat removal (RHR) system during division II RHR work (yellow risk)
- Unit 2, 5 x 5 array of control rod drives (CRD) hydraulically isolated for maintenance on CRD 42-27

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors conducted one complete system walkdown of the Unit 1 and 2 125 VDC systems to assess the alignment and condition of the 125 VDC station batteries and associated battery chargers. The inspectors reviewed the system health report, closed condition reports (CR) for the last 12 months, system operating procedures, check-off lists, and system electrical prints. The inspectors evaluated ongoing maintenance and open condition reports associated with the 125 VDC system to determine the effect on system health and reliability. The inspectors evaluated the 125 VDC systems' overall condition, to verify whether the system was properly operated and maintained within design and licensing requirements.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 10 Samples)

.1 Fire Protection - Tours

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 inspected areas included:

- Units 1 and 2, emergency systems service water pump house, fire zones 0-51 and 0-52
- Units 1 and 2, "E" EDG while substituted for the "C" EDG
- Units 1 and 2, 125 VDC battery rooms
- Unit 1, division I and division II 4.16 KV switch gear rooms 719' elevation, fire zone 1-4C, 1-4D
- Unit 1, drywell, fire zone 1-4F
- Unit 1, division I core spray compartment, fire zone 1-1A
- Unit 1, division II core spray compartment, fire zone 1-1B
- Unit 1, high pressure coolant injection room, fire zone 1-1C
- Unit 2, common core spray valve room fire zone 2-5B

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation

a. Inspection Scope

On May 30, 2006, the inspectors observed an unannounced fire brigade drill in the radiological control area. The fire was a simulated oil fire at the Unit 2 turbine building breaker test shop. The inspectors assessed PPL's strategies to fight a fire on-site and to evaluate the readiness of PPL to prevent and fight fires.

The inspectors observed the fire brigade members don protective clothing and turnout gear. In addition, the inspectors observed the fire fighting equipment brought to the fire area scene to evaluate whether sufficient equipment was available and properly utilized. The inspectors observed fire fighting directions and radio communications between the brigade leaders, brigade members, and the control room. The inspectors reviewed the post drill critique to evaluate if the drill objectives' acceptance criteria were satisfied.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 Sample)

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and PPL procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors interviewed plant personnel, flood analysis and design documents, including the FSAR, engineering calculations, abnormal operating procedures and plant drawings. The inspectors performed walkdowns in the Unit 1 division I and division II residual heat removal pump compartments in the reactor building 645 foot elevations. The inspectors verified that adequate procedures were in place to identify and respond to flooding.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07 - 1 Sample)

a. Inspection Scope

The inspectors reviewed PPL's inspection, cleaning, and maintenance activities, and reviewed PPL's evaluation of the as-found conditions for the Unit 2 "B" reactor core isolation cooling (RCIC) room cooler (ZE228B). The inspectors verified whether PPL properly evaluated the results to identify adverse trends and ensure adequate heat transfer capabilities. The inspectors compared their observations against PPL's procedures and specifications to assess whether the heat exchangers were capable of performing their safety function under design basis accident conditions. The inspectors' review included the following documents:

C Unit 2 "B" RCIC room cooler (ZE228B), WO 583371

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 Sample)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On June 22, 2006, the inspectors observed licensed operator performance in the simulator during operator requalification training. The inspectors compared the operators' actions to Technical Specification requirements, emergency plan procedures, and emergency operating procedures. The inspectors also evaluated PPL's critique of the operators' performance to identify discrepancies and deficiencies in operator training. The following training scenario was observed:

- OP002, "Unexplained Reactivity Change, Unisolable RCIC Steam Line Break Discharging Outside of Secondary Containment Results in a Rapid Depressurization"

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 Samples)

a. Inspection Scope

The inspectors evaluated PPL's work practices and follow-up corrective actions for selected 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. The following issues were reviewed:

- Unit 2 125 VDC battery grounds and charger failure
- Nuclear Instrumentation (Unit 1 & 2 Source Range Monitor failures and Unit 2 local power range monitor failures)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 6 Samples)

a. Inspection Scope

The inspectors reviewed the assessment and management of selected maintenance activities to evaluate the effectiveness of PPL's risk management for planned and emergent work. The inspectors compared the risk assessments and risk management actions to the requirements of 10 CFR 50.65(a)(4) and the recommendations of NUMARC 93-01 Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities." The inspectors evaluated the selected activities to determine whether risk assessments were performed when required and appropriate risk management actions were identified.

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 emergent work activities. The inspectors performed control room and field walkdowns to verify whether the compensatory measures identified by the risk assessments were appropriately performed. The selected maintenance activities included:

- Units 1 and 2, 4Kv vacuum breaker modification schedule
- Units 1 and 2, ESS201/OX203 transformer outage cancelled due to external risks/weather
- Unit 1, high pressure coolant injection (HPCI) surveillance with one division of low pressure coolant injection (LPCI) out-of-service (OOS) (Yellow Risk)
- Unit 1, RHR division II outage, stem nut replacement for HV151F047B and HV151F048B (Yellow Risk)
- Unit 1, risk assessment of “SE-149-010"
- Unit 2, recirculation pump seal pressure and temperature change during startup from maintenance outage, ODM 777755

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14 - 2 Samples)

a. Inspection Scope

For the non-routine events described below, the Inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures, technical requirements, and analysis.

Unit 2 Unexpected Recirculation Pump Speed Increase

On May 5, 2006, the inspectors observed and reviewed operator and plant response to the reactivity addition and resulting power transient which followed resetting of the recirculation pump runback limiter logic. Although the oscillating power range monitor system was inadvertently enabled by the resulting reactor power increase, reactor scram limits were not reached and there were no complications from the power excursion. Operators locked the recirculation pump scoop tubes and PPL took appropriate actions to restore automatic control of the pump. Inspectors verified that the recirculation system and the integrated plant transient response were in accordance with design. Inspectors verified that PPL initiated appropriate corrective actions to address identified issues and determine cause.

Unit 1 Reactor Scram While Transferring Division II RPS Power Supply

On June 15, 2006, the inspectors observed and reviewed the plant response to a Unit 1 automatic shutdown (scram) from 100% reactor power. The scram occurred when the division II RPS power was transferred for Electric Power Monitoring Assemblies (EPA) breaker maintenance. There were no complications from the scram because all mitigating systems responded appropriately to the scram. The plant remained in hot shutdown until the cause of the scram was determined and corrected. This event is also discussed in section 1R20.2, “Refueling and Other Outages Activities.”

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 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, FSAR, and associated Design Basis Documents as references during these reviews. The issues reviewed included:

- Units 1 and 2, EDG common mode failure due to “C” EDG ESW leak, CR 784933
- Units 1 and 2, 4Kv breaker hardened grease, CR 768740
- Unit 1, motor operated valve stem nut failures, CR 768920
- Unit 2, elevated core spray keepfill pressure, CR 778841
- Unit 2, “A” reactor recirculation pump seal anomaly, ODM 777755
- Unit 2, HPCI slow (degraded) start time, CR 783341

b. Findings

Introduction. A Green self-revealing non-cited violation (NCV) was identified for failure to have adequate work instructions in accordance with 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." This resulted in not identifying stem nut degradation prior to the failure of two Unit 1 residual heat removal valves.

Description. On March 27, 2006, Unit 1 was shutdown for a refueling outage. During the performance of the division II, residual heat removal (RHR) logic system functional test the "D" RHR pump suction valve (HV151F004D) failed to close. PPL determined the valve's motor operator stem nut threads were severely worn and were not able to engage the valve stem threads to open or close the valve. After the failure, the valve stem showed evidence of brass particles in the grease and PPL determined that motor operated valve (MOV) diagnostic data could have been used to directly determine the condition of the stem nut threads. PPL reviewed additional stem nuts using diagnostic data in addition to stem grease inspections. On March 28, 2006, using diagnostic data, PPL identified indications of a severely degraded stem nut on the Unit 1 "C" RHR pump suction valve (HV151F004C). PPL performed a grease inspection of the accessible area of the valve stem and found no evidence of stem nut wear (brass particles on the valve stem and valve stem grease). However, PPL did not directly inspect the stem nut to fully evaluate the discrepancy in the diagnostic data and grease inspection. PPL incorrectly concluded that the diagnostic data, which indicated a severely degraded stem nut, was the result of a gap between the stem nut and stem nut lock nut. The Unit 1 "C" RHR pump suction valve failed to fully open on April 6, 2006 because the valve's motor operator stem nut was severely worn.

NRC Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves," requested PPL to have a program to "ensure that changes in required performance resulting from degradation (such as those caused by age) can be properly identified and accounted for." PPL's response to this generic letter, PLA-4578, "Susquehanna Steam Electric Station Generic Letter 96-05: 180 Day Response," dated March 17, 1997, stated that PPL performs periodic testing to assure MOVs are functioning properly, determine margins, to identify marginal MOVs and to identify any degradation from the previous tests. PPL determined that the periodic verification / testing being used to monitor MOV performance was ineffective at properly identifying degraded stem nut wear. Specifically, station procedure MT-GM-050, "Limitorque Type SMB-000 Through SMB-004 Operator Maintenance," does not have clear guidance or acceptance criteria for direct inspection of the stem nut. In addition, PPL had changed the periodicity of this procedure from a periodic overhaul to one based on the condition of valve stem grease or MOV diagnostic in early 1993.

Station procedure MT-GE-003, "Limitorque Valve Actuator Maintenance," specified a preventive maintenance task to "Inspect valve stem and stem nut for damage (i.e., check the stem for galling or pitting and check the valve packing area for brass particles indicating stem nut damage / excessive wear)." The failures of the Unit 1 "D" and "C" RHR pump suction valves demonstrated that this inspection is only effective for a high rate of stem nut wear, not long term stem nut wear. Finally, MT-EO-053, "Static or Dynamic Testing of Motor Operated Valves Using Quicklook," did not use diagnostic data to directly monitor or trend stem nut wear. PPL's procedural guidance did not contain appropriate quantitative or qualitative acceptance criteria to determine the condition of the valve's stem nut. The inadequate procedural guidance contributed to the undiagnosed degradation and ultimate failure of the stem nuts on the RHR pump suction valves HV151F004D and HV151F004C.

Analysis. The inspectors determined that PPL's inadequate procedural guidance to periodically evaluate the condition of the MOV stem nuts resulted in the failure of the Unit 1 "D" and "C" RHR pump suction valves on March 27, 2006, and April 6, 2006 constituted a performance deficiency and a finding. The finding was more than minor because the condition affected the Procedure Quality attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Prior to the Unit 1 shutdown and after shutdown both valves were successfully cycled. The valves failed on successive cycles after shutdown. The inspectors performed a Phase 1 screening using IMC 0609, Appendix G, "Shutdown Operations Significant Determination Process." The finding was determined to not require a quantitative assessment, and therefore the finding was of very low safety significance (Green) because the failure of the Unit 1 "D" and "C" RHR pump suction valves did not affect (1) the ability to monitor and control reactor vessel water level and temperature using station procedures, (2) did not affect required instrumentation and reactor coolant system isolations, (3) did not affect offsite power or reduce the required onsite emergency power supplies, and (4) did not reduce the required compliment of shutdown cooling and emergency vessel makeup equipment below the compliment required by Technical Specifications.

Enforcement. 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that, "Activities affecting quality shall be prescribed by documented instructions, procedures or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished." Contrary to this requirement station procedure MT-GM-050, "Limitorque Type SMB-000 Through SMB-004 Operator Maintenance," did not have the clear guidance or specific acceptance criteria needed to determine that the stem nuts were acceptable for continued operation. Not having adequate guidance to inspect these stem nuts resulted in the failure of the residual heat removal pump suction valves HV151F004D and HV151F004C on March 27, 2006 and April 6, 2006, respectively. Because this failure to comply with 10 CFR 50 Appendix B, Criterion V, is of very low safety significance and has been entered into PPL's corrective action program, (CR 768920), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as **NCV 05000387/2006003-01 Inadequate Procedures Resulted in Motor Operated Valve Failures.**

1R17 Permanent Plant Modifications (71111.17B - 9 Samples)

a. Inspection Scope

The inspectors reviewed nine risk-significant plant modification packages selected from the design changes that were completed within the past two years. The review was performed to verify that: (1) the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded through the modifications; and, (2) the modifications performed during increased risk configurations did not place the plant in an unsafe condition. The modifications reviewed are listed in Attachment 1.

The selected plant modifications were distributed among the Initiating Event, Mitigating Systems, and Barrier Integrity cornerstones. For the selected modifications, the inspectors reviewed the design inputs, assumptions, and design calculations to determine the design adequacy. The inspectors also reviewed field change notices that were issued during the installation to confirm that the problems associated with the installation were adequately resolved. In addition, the inspectors reviewed the post-modification testing, functional testing, and instrument and relay calibration records to determine readiness for operations. Finally, the inspectors reviewed the affected procedures, drawings, design basis documents, and FSAR sections to verify that the affected documents were appropriately updated.

For the accessible components associated with the modifications, the inspectors also walked down the systems to detect possible abnormal installation conditions.

In addition, the inspectors reviewed condition reports (CRs) associated with 10 CFR 50.59 issues and plant modification issues to ensure that PPL was identifying, evaluating, and correcting problems associated with these areas and that the planned or completed corrective actions for the issues were appropriate. The inspectors also reviewed self-assessments related to 10 CFR 50.59 SEs and plant modification activities at Susquehanna.

b. Findings

Introduction. A self-revealing non-cited violation of 10CFR50 Appendix B, Criterion XVI, Corrective Action was identified because PPL failed to evaluate and correct degraded material in the “C” Emergency Service Water (ESW) pump breaker that caused a failure of the breaker on April 5, 2006.

Description. On April 5, 2006, PPL experienced a failure of the “C” ESW pump breaker, a Westinghouse DHP Magnetic Air Circuit Breaker. Specifically, the breaker failed to close on demand. The ESW pump was declared inoperable and the 24 year old breaker was replaced with a new breaker design. The new breaker was part of an on ongoing modification, which began in 2003, to replace the old 4Kv breakers at the facility. PPL had the breaker examined by the vendor, and the analysis of the breaker determined that this failure was due to a hardened grease condition on bearings that were internal to the breaker. The hardened grease caused the internal mechanism of the breakers to bind and not move as required to shut the breaker.

The inspectors' review found that twice before, the "C" ESW pump had failed to start under similar circumstances. In January 2004, the "C" ESW pump did not start when an operator depressed the start pushbutton. PPL performed an inspection of the breaker and attributed this failure to a degraded interposing relay. The relay was replaced, and PPL performed no additional failure analysis of the breaker. The breaker was replaced, later in 2004, as part of a normal four year preventive maintenance (PM) procedure, not because of the failure. Again, in August 2005, the "C" ESW pump failed to start when the operator depressed the start pushbutton. PPL performed an apparent cause evaluation (ACE) for this event and concluded that the most likely cause was a degraded interposing relay for the breaker. PPL's implemented corrective actions included replacing the interposing relay and performing additional relay monitoring.

The inspectors noted the ACE performed in 2005 had ruled out hardened grease as a potential cause for the breaker failure, because the breaker had previously been overhauled in accordance with station procedures, and no indications of hardened grease were found. The inspectors reviewed the station procedures and determined that this was not an overhaul but a four year PM procedure. The inspectors determined that this PM was not adequate to rule out grease hardening as a potential cause because there were lubricated surfaces internal to the breaker that were not inspected. Additionally the inspectors reviewed a variety of industry operating experience and NRC guidance documents that document numerous examples of potential corrective/preventive actions to address age management issues associated with breakers which included hardening grease.

The inspectors noted that industry guidance for breaker overhauls suggests a 10 to 15 year performance interval, but PPL had not performed an overhaul of this breaker during its 24 year period of use. Additionally, the inspectors noted during their review of the ACE for the previous failure in 2005 that PPL had improperly ruled out hardened grease as a failure mechanism based on performance of the 4 year PM. Therefore, the inspectors determined PPL did not fully evaluate hardened grease (the breaker not closing due to mechanical resistance) as a potential cause of the "C" pump not starting. Inspectors concluded that the corrective actions taken for the previous events including industry operating experience were not successful in preventing the failure of the "C" ESW breaker in April 2006 due to hardened grease.

<u>Analysis</u>. The inspectors determined that this issue was a performance deficiency because PPL had not fully evaluated a 2005 failure, which resulted in the failure to identify the cause of the condition, correct the condition, and prevent the April 2006 "C" ESW pump breaker failure. The inspectors determined that this deficiency was reasonably within PPL's ability to identify and correct prior to April 2006 based on related industry OE and the ACE conducted by PPL for the August 2005 failure.

The finding was more than minor because the condition affected the Equipment Performance attribute of the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events.
In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a Phase 1 significance determination process (SDP) screening and determined the issue to be of very low safety significance (Green). The finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification Allowed Outage Time, did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk significant per 10CFR50.65, for greater than 24 hours, and did not screen as potentially risk significant due to external events. The performance deficiency associated with the failure of the “C” ESW pump breaker had a PI&R (evaluation) cross-cutting aspect because PPL did not perform a thorough evaluation (ACE) of the problem in 2005 so that the resolution addressed causes and extent of condition as necessary to prevent the April 2006 failure of the 4Kv breaker due to material degradation.

Enforcement. 10 CFR 50 Appendix B, Criterion XVI, “Corrective Actions,” requires, in part, that “Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” Contrary to the above, PPL failed to fully evaluate hardened grease as an identified condition adverse to quality for 4Kv breakers. PPL’s implemented corrective actions were not adequate to prevent the “C” ESW pump breaker failure due to hardened grease on April 6, 2006. Because the finding was of very low safety significance and has been entered into the corrective action program in condition report 699219, this violation is being treated as an NCV, consistent with section VI.A.I of the NRC Enforcement Policy. **NCV 05000387, 388/2006003-02, Failure to Identify Material Degradation Which Resulted in a Failure of the "C" ESW Pump Breaker.**

1R19 Post-Maintenance Testing (71111.19 - 6 Samples)

a. Inspection Scope

The inspectors observed portions of post-maintenance testing 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. The post-maintenance testing activities reviewed included:

- Units 1 and 2, "A" EDG replacement of air start shuttle valves and test, CR 780778 and PCWO 780791
- Units 1 and 2, "E" EDG, "B" 125 VDC battery charger disassemble, clean, and inspect, RLWO 715966
- Units 1 and 2, "C" EDG testing after the ESW pipe replacement to the "C" EDG
- Unit 1, control rod scram time testing, SR-155-004
- Unit 2, stem nut inspection and replacement on RHR motor-operated valves HV251F006A and 6D, PCWO 771000 and PCWO 774054
- Unit 2, RCIC trip and throttle valve trip arm linkage pin realignment and fastener torquing, PCWO 774054

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 2 Samples)

.1 Other Outage Activities: Unit 2 Drywell Unidentified Leakage

a. Inspection Scope

The inspectors reviewed the outage risk management plan for the Unit 2 drywell unidentified leakage outage, conducted April 28, to May 5, 2006, to confirm that PPL had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspector observed and / or reviewed the outage activities listed below.

- Identification of the drywell leak
- Establishment of a reactor vessel cool down rate
- Outage configuration controls including:
    1) availability and accuracy of reactor coolant system instrumentation;
    2) availability of nuclear instrumentation;
    3) electrical power alignments;
    4) decay heat removal system operation;
    5) availability of reactor inventory makeup water systems; and
    6) secondary containment controls and integrity.
- Drywell walkdown after shutdown and prior to final closeout
- Replacement of control rod drive 42-27 O-Ring, including system clearances
- Reactor startup, including a plant restart review, in-sequence critical, scram time testing, resetting the reactor scoop tubes and reactor power increase

During the conduct of the maintenance outage activities, the inspectors reviewed the associated documentation to ensure that the tasks were performed safely and in accordance with plant Technical Specification requirements and operating procedures.

b. Findings

Introduction. A Green self-revealing NCV was identified for failure to have adequate work instructions in accordance with 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which resulted in a reactor coolant system mechanical joint leak.

Description. On April 28, 2006, Unit 2 was shutdown to identify and correct an unidentified reactor coolant system leak in the drywell. PPL had been trending the increased leakage and pro-actively shutdown Unit 2 when the leakage reached approximately 1.2 gpm, which was less than the TS limit of 5 gpm. The source of the leak was determined to be a flange to flange mechanical joint connecting the reactor head vent line to the reactor head (N7 flange). Once the reactor was depressurized PPL determined that the eight studs used to secure the mechanical joint did not have the appropriate torque value and the mechanical joint flange faces were not parallel. Station procedure ME-2RF-101, "Unit 2 Reactor Vessel Reassembly," Revision 2, specified a torque value of 285 ft-lbs, but did not specify any requirements to ensure that the flange faces were parallel. The as found torque values ranged from 90 ft-lbs to 150 ft-lbs and the as found gap between the flanges was not consistent. The gap (misalignment) ranged from 190 mills as measured at the 0E location to 249 mills as measure at the 270E location.

PPL determined the cause of the leak was not correctly assembling the mechanical joint (misalignment). Specifically, the integrity of the mechanical joint requires adequate gasket crush to prevent leakage and proper torque to maintain the integrity of the mechanical joint. In an effort to "streamline" the vessel reassembly procedures PPL removed previous guidance in procedure ME-2RF-002, "Reactor Head Piping and Insulation Structure Installation (Unit 2)," that verified adequate gasket crush and adequate flange torque values. The new procedure, ME-2RF-101, "Unit 2 Reactor Vessel Reassembly," issued on November 6, 2002, contained only guidance on required torque values. This is a performance deficiency because PPL removed prescribed documented procedural instructions that were needed to correctly assemble this mechanical joint. Removal of these instructions resulted in elevated reactor coolant leakage.

Analysis. The inspectors determined that PPL's inadequate procedure to assemble a mechanical joint on the reactor head vent line constituted a performance deficiency and a finding. This finding is greater than minor because it is associated with the Barrier Integrity Cornerstone of Procedure Quality and affected the cornerstone objective of providing reasonable assurance that physical design barriers (reactor coolant system) protect the public from radio nuclide releases caused by accidents or events.
The misalignment resulted in inadequate gasket crush and elevated leakage through this flange. Unit 2 was shutdown on April 28, 2006, to correct the elevated leakage. The inspectors performed a Phase 1 screening using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Finding for At-Power Situations," using the Initiating Events - LOCA Initiators. The finding was determined to be of very low safety significance (Green) because the reactor coolant system leak would not have

resulted in exceeding the Technical Specification limit for identified leakage, nor would it have likely affected other mitigation systems resulting in a total loss of their safety function.

Enforcement. 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires in part that, activities affecting quality shall be prescribed by documented instructions, procedures or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. Contrary to this requirement station procedure ME-2RF-101, "Reactor Vessel Reassembly," did not have specific documented guidance to properly align the reactor head vent flanges. Misalignment of these flanges on March 19, 2005, resulted in elevated reactor coolant system leakage and a planned maintenance outage to correct the leakage. Because this failure to comply with 10 CFR 50 Appendix B, Criterion V, is of very low safety significance and has been entered into PPL's corrective action program (CR 775285), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as **NCV 05000388/2006003-03 Inadequate Procedure Results in Elevated Reactor Coolant System Leakage.**

.2 Other Outage Activities: Unit 1 Reactor Scram During Division II RPS Power Transfer

a. Inspection Scope

On June 15, 2006, Unit 1 experienced an automatic reactor shutdown (scram) from 100% RTP during transfer of the division II RPS power supply. The inspectors' review, of the operators response, is documented in section 1R14, "Operator Performance During Non-Routine Evolutions and Events." During the unplanned outage the inspectors reviewed the apparent cause of the scram and the corrective actions implemented to prevent recurrence of the scram. The outage schedule, risk management plan and the plant restart review were all reviewed during the outage to confirm that PPL had appropriately considered risk and corrected the cause of the unplanned reactor scram. The unit was restarted and achieved full RTP on June 17, 2006.

b. Findings

Introduction. A Green self-revealing non-cited violation was identified for failure to comply with 10 CFR 50 Appendix B, Criterion III, Design Control. PPL did not correctly verify that the Power Range Neutron Monitoring System (PRNMS) modification would not adversely affect the design bases of the RPS.

Description. On June 15, 2006, the Unit 1 reactor scrammed, from 100% power, when division II RPS power was transferred to the alternate supply for maintenance on the EPA breakers. The reactor scram signal was from the PRNMS which was replaced through modification during the previous refueling outage in April of 2006. The RPS signal was initiated because the Average Power Range Monitor 12 (APRM12) and

APRM14 receive an input from the reactor mode switch. When the reactor mode switch is not in RUN the APRM scram setpoint is reduced to 15% power. The relays that are used to provide the mode switch position to APRM12 and APRM 14 are powered from division II of RPS. When the division II RPS power was transferred to the alternate supply, the mode switch relays were momentarily de-energized, causing ARPM 12 and ARPM14 scram setpoint to be reduced to less than 15% power. The Voter modules of the PRNMS logic saw the loss of power to the division II RPS logic and prevented the scram setpoint reduction, as designed, for these two ARPMs. However, when power was restored to division II RPS the Voter logic responded quicker than the APRM logic, essentially enabling the reduced scram setpoint (15% power) on APRM 12 and APRM 14. Since the reactor was at full power, a reactor scram resulted.

The NRC determined that the PRNMS modification did not meet the FSAR design basis. The FSAR, section 7.1.2a.1.1.1.1, requires in part, that “The loss of one RPS power supply shall neither cause nor prevent a reactor scram.” PPL should have identified this design interface issue during the modification design review. Station procedure NEPM-QA-0241-2, Rev. 1, “Design Review Checklist,” (DCP 618882, Rev. 2) indicated that all potentially applicable design considerations had been correctly evaluated for applicability and all design considerations had been correctly and completely addressed on December 22, 2005. However, the PRNMS / RPS interface issue was not fully evaluated and therefore, did not identify and prevent the adverse interface issue created by the PRNMS modification. On June 17, 2006, PPL completed a modification to correct this condition.

Analysis. The inspectors determined that PPL’s failure to correctly verify that the PRNMS modification would not adversely affect the design bases of the RPS constituted a performance deficiency and a finding. This finding is greater than minor because the condition affected the Design Control attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The inspectors performed a Phase 1 screening using IMC 0609 Appendix A, “Determining the Significance of Reactor Inspection Finding for At-Power Situations,” using Initiating Events - Transient Initiators. The finding was determined to be of very low safety significance (Green) because all mitigating systems were available and responded appropriately to the reactor scram.

This finding is also related to the human performance cross-cutting area because PPL did not ensure supervisory and management oversight of work activities, including contractors such that there would be no adverse system interface issues in the PRNMS design which supports nuclear safety.

Enforcement. 10 CFR 50 Appendix B, Criterion III - Design Control states in part that, “Measures shall be established to assure that applicable regulatory requirements and the design basis as defined in 10 CFR 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated in specifications, drawings, procedures, and instructions.”

Contrary to the above, PPL did not correctly verify that the PRNMS modification would not adversely affect the design bases of the RPS, on December 22, 2005, which resulted in a Unit 1 reactor scram on June 15, 2006, when the division II RPS power was transferred to the alternate supply. To correct this problem, PPL installed a modification on to prevent recurrence. Because this finding is of very low safety significance and has been entered into PPL's corrective action program (CR 786735), this violation is being treated as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as **NCV 05000387/2006003-04 Inadequate Design Review of PRNMS Modification Resulted in a Reactor Scram.**

1R22 Surveillance Testing (71111.22 - 6 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. The observed or reviewed surveillance tests included:

- Units 1 and 2, recirculation flow element calibration (semi-annual), SO-150-002 and SO-250-002
- Units 1 and 2, control rod stroke timing in mode 1 or 2, TP-055-010
- Unit 1, core flow calibration, TP-164-031
- Unit 1, RHR division I valve stroke time and system flow surveillance, SO-149-A05 and SO-149-A02 (In-service Test sample)
- Unit 1, local leak rate test of IC-PS-IN610A on penetration X-3B, SE-159-114
- Unit 1, functional test of division II RHR from the remote S/D panel, SE-149-010

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 2 Samples)

a. Inspection Scope

The inspectors reviewed temporary plant modifications to determine whether the temporary changes 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 FSAR, Technical Specifications, and assessed the adequacy of the safety determination screenings and evaluations. The inspectors also assessed configuration control of the temporary changes by reviewing selected drawings and procedures to verify whether appropriate updates had been made. The inspectors compared the actual installations to the temporary modification documents to determine whether the implemented changes were consistent with the approved documents. The inspectors reviewed selected post installation test results to verify whether the actual impact of the temporary changes had been adequately demonstrated by the test. The following temporary modifications and documents were included in the review :

- "E" EDG Automatic Transfer Switch, EC 752045
- Emergency service water uncoated piping to the "C" EDG lube oil cooler, EC 785128

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 1 Sample)

a. Inspection Scope

On April 25, 2006, the inspectors observed a control room simulator based training event (2006 White Team Emergency Drill/simulator). The inspectors assessed licensed operator adherence to emergency plan implementing procedures, and their response to simulated degraded plant conditions to identify weaknesses and deficiencies in event classification and notification. The inspectors reviewed PPL's critique of the simulator control room participants to evaluate PPL's identification of weaknesses and deficiencies. The inspectors compared PPL's identified findings against the inspectors' observations to determine whether PPL adequately identified problems.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 6 Samples)

a. Inspection Scope

The inspectors reviewed PPL's self assessments, Quality Assurance (QA) audits, Licensee Event Reports, and Special Reports related to the access control program since the last inspection, and determined that identified problems were entered into the corrective action program for resolution.

The inspectors reviewed corrective action reports related to access controls. Included in this review were high radiation area radiological incidents (non-Performance Indicators [PIs] identified by PPL) in high radiation areas <1R/hr that have occurred since the last inspection in this area.

For repetitive deficiencies or significant individual deficiencies in problem identification and resolution identified above, the inspectors determined that PPL's self-assessment activities were also identifying and addressing these deficiencies.

The inspectors reviewed PPL documentation packages for all PI events occurring since the last inspection; determined if any of these PI events involved dose rates >25 R/hr at 30 centimeters or >500 R/hr at 1 meter; and, determined what barriers had failed and if there were any barriers left to prevent personnel access. For unintended exposures >100 mrem Total Effective Dose Equivalent (or >5 rem Skin Dose Equivalent or >1.5 rem Lens Dose Equivalent), the inspectors determined if there were any overexposures or substantial potential for overexposure.

The inspectors reviewed radiological problem reports since the last inspection which found that the cause of the event was due to radiation worker errors; determined if there is an observable pattern traceable to a similar cause; and, determined if this perspective matches the corrective action approach taken by PPL to resolve the reported problems. The inspectors discussed with the Radiation Protection Manager any problems with the correction actions planned or taken. The inspectors verified adequate posting and locking of entrances to all high dose rate - high radiation area, and very high radiation area (if reasonably accessible).

The inspectors reviewed radiological problem reports since the last inspection that found that the cause of the event was radiation protection technician error; determined if there is an observable pattern traceable to a similar cause; and, determined if this perspective matched the corrective action approach taken by PPL to resolve the reported problems.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed PPL's self-assessments, audits, and special reports related to the as low as reasonably achievable (ALARA) program since the last inspection and determined that PPL's overall audit program's scope and frequency (for all applicable areas under the Occupational Cornerstone) meet the requirements of 10 CFR 20.1101(c).

The inspectors determined that identified problems are entered into the corrective action program for resolution. The inspector reviewed dose significant post-job (work activity) reviews and post-outage ALARA report critiques of exposure performance, and determined that identified problems are properly characterized, prioritized, and resolved in an expeditious manner.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 1 Sample)

a. Inspection Scope

The inspectors reviewed the plant Final Safety Analysis Report to identify applicable radiation monitors associated with transient high and very high radiation areas including those used in remote emergency assessment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems
(71152 - 2 Annual Samples, 1 Semi-Annual Sample)

.1 Review of Items Entered into the Corrective Action Program

As required by Inspection Procedure (IP) 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed screening of all items entered into PPL's corrective action program. This was accomplished by reviewing the description of each new action request / condition report and attending daily management meetings.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by IP71152, Identification and Resolution of Problems, the inspectors performed a review of PPL's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was not only focused on repetitive equipment and corrective maintenance issues but also considered the results of daily inspector corrective action program item screening discussed in Section 4OA2.1. The review also included issues documented outside the normal corrective action program in system health reports, corrective maintenance WOs, management meeting and maintenance rule assessments. The inspectors' review nominally considered the six-month period of

January 2006 through June 2006, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with the results contained in PPL's latest integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in PPL's trend report were reviewed for adequacy.

b. Assessment and Observations

The inspectors identified a trend in the area of recurrent equipment problems. Specifically, the implementation of previous corrective actions were unable, in some cases, to prevent recurrent equipment issues. During the first six months of 2006, the following equipment / component problems recurred.

- Control Rod Drive Power Supply Failure, CR 753990
- "C" ESW Pump Failed to Start, CR 768740
- Failure of MOV Stem Nut on HV151F004C, CR 768920
- ESW piping pressure boundary leak downstream of the "C" EDG lube oil cooler, CR 785664

PPL documented this trend in CR 771319. PPL has determined that the increase in number of repetitive equipment problems was related to (1) insufficient validation of key assumptions used to establish the cause of the equipment failures, (2) less than adequate consideration of uncertainty in the evaluated failure cause, and (3) actions developed before the cause report is completed. PPL's corrective actions, included in part, additional procedure guidance to (1) improve the technical decisions, (2) ensure condition reports are not closed prior to the analysis being performed, and (3) feedback to generate additional condition reports and analysis if the extent fo condition includes a larger population than original identified.

.3 Annual Sample: Review of "C" Emergency Service Water Pump Breaker Problems

a. Inspection Scope

Inspectors reviewed the technical evaluations for the first two failures of the "C" ESW pump to start when the control room hand switch was depressed. The first failure to start was on February 25, 2004 and the second failure was on August 10, 2005. This PI&R Sample was selected after the first repeat failure of the ESW pump breaker to close and start the pump. During the inspection a third failure of the ESW pump to start occurred on April 5, 2006. The evaluation and corrective actions for this third breaker failure were added to the scope of the inspection.

b. Findings and Observations

The related finding has been documented in Section 1R17 of this report.

For each failure of the ESW pump to properly start, PPL entered the issue into the Corrective Action program. Inspectors determined that PPL was effective at identifying problems and placing them in the corrective action program.

Inspectors found that the technical evaluation for the second “C” ESW pump breaker failure was more detailed than the first evaluation. However, PPL’s second evaluation still focused on the overheating and the melting of relay contacts and there was no detailed evaluation of other potential causes. The PPL cause evaluation was completed approximately six months after this repeat failure of the ESW breaker. The inspectors concluded that PPL missed an opportunity to perform a detailed technical evaluation of the 2004 and 2005 ESW pump breaker failures. Because the evaluation focused on one potential cause (failed interposing relay), a condition adverse to quality, hardened grease in 4Kv breakers, continued uncorrected.

For the first two failures of the “C” ESW pump to start, inspectors found that uncertainties in the cause determination were not factored into the corrective action plan. PPL did not perform a detailed effectiveness review of the 4Kv breaker replacement project and did not perform a detailed review of actual preventive maintenance activities including vendor overhaul for the 4Kv breakers.
Inspectors determined that PPL did not fully utilize risk insights into the scheduling of 4Kv breaker replacements because several of the highest risk systems and components had the breaker modification delayed for several years.

After the third failure, PPL determined that the failures in 2005 and 2006 were most likely caused by hardened grease in the breaker linkage and that the “burned up” relays were not the cause but rather a result of the degraded condition. Inspectors observed that PPL’s failure to evaluate all possible failure mechanisms of the breaker and the failure to address uncertainty in the determined cause, allowed a degraded material condition to continue uncorrected. See Section 1R17 of this report.

.4 Annual Sample: Review of PPL Personnel Qualification Problems

a. Inspection Scope

The inspectors reviewed PPL’s initial evaluation and a sample of associated corrective actions for Training and Qualification condition reports issued between December 2001 and May 2006. This sample was selected due to the potential for an adverse trend and increased significance to the repetitive issues with unqualified workers. The inspectors interviewed personnel on the use and management of the training and qualification matrix. The inspectors also reviewed PPL’s actions taken to address NRC Finding, FIN 05000387/2005-009-01, which identified that a fire brigade leader did not participate in the required drills to maintain his qualification. The inspectors evaluated PPL’s threshold for identifying problems and effectiveness of actions taken to resolve problems. The inspectors compared station activities and the identified problems in the corrective action system against the Quality Assurance procedures for plant staff training requirements and qualification management.

b. Findings and Observations

Introduction. The inspectors identified a Green NCV because PPL did not correct long standing issues related to worker qualification as required by 10 CFR 50 Appendix B Criterion XVI. This resulted in unqualified workers performing tasks important to safety as described by the Quality Assurance (QA) program.

Description. On January 11, 2006, an unqualified worker performed a calibration procedure for reactor recirculation flow modules. The flow modules provide a safety-related flow bias input to the reactor protection system that is used to scram the reactor. On June 7, 2006, PPL identified that seven station fire brigade members no longer met fire brigade training requirements. A site Fire Brigade is equipped and trained to ensure self-sufficiency with respect to manual firefighting capability as described by QA procedures and the Fire Protection Review Report. Both of these unqualified worker events were the result of longstanding problems surrounding the maintenance and use of the training and qualification program at PPL. These conditions did not adversely impact plant operations.

PPL's QA procedures describe the requirements for providing the training of personnel performing activities affecting quality and the requirements to assure suitable proficiency is achieved and maintained. Specifically, NDAP-QA-0010, "Minimum Qualification and Training Requirements" and OPS-6, "Qualifications, Training and Certification of Personnel" require that individuals complete required training in accordance with the appropriate training program. The inspectors found 42 individual workers had performed tasks without required qualifications and identified that PPL's corrective actions were not effective in eliminating the contributing causes to the unqualified worker events identified at PPL from 2002 through the first half of 2006. This included PPL's response to a fire brigade leader not meeting qualification requirements (fire drills) which was identified by the NRC on December 1, 2005. Inspectors found that PPL was not effective in preventing tasks being performed by unqualified workers primarily due to narrowly focused evaluations and corrective actions that did not address contributing causes. The contributing causes for the unqualified worker events described by this finding included: supervisors not verifying qualifications in accordance with NDAP-QA-500, and the tracking of worker qualifications by methods that are outside of the described policies and procedures. The inspectors observed that each time the deficiency became evident, PPL took action to reconcile the training or qualification of the individuals involved, but did not correct the broader weaknesses in the training and qualification verification process.

Analysis. This finding is a performance deficiency because PPL failed to implement prompt and appropriate corrective actions for a long standing deficiency in compliance with Quality Assurance Program procedures, which provide for the training of personnel performing activities affecting quality and assures that suitable proficiency is achieved and maintained. This finding is more than minor because if left uncorrected, the tasks being performed by unqualified workers will become a more significant safety concern. An unqualified worker calibrating safety-related equipment affects the Equipment Performance attribute of the mitigating systems cornerstone and unqualified fire brigade members affect the Protection Against External Factors attribute of the same cornerstone. The finding affects the cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events. This finding is of very low safety significance because the work performed by the unqualified individual performing the recirculation flow calibration did not result in a loss of system safety function, and did not represent an actual loss of safety function of any single train of equipment. The Significance Determination Process (SDP), Appendix F, does not specifically address fire brigade issues and allows for management discretion to determine issue significance. This performance issue was reviewed by NRC management and is determined to be a finding of very low safety significance (Green).

This finding is related to the problem identification and resolution cross-cutting area because PPL failed to take appropriate corrective actions to address the adverse trend in unqualified worker events in a timely manner and commensurate with the significance.

Enforcement. 10 CFR 50 Appendix B, Criterion XVI states in part, "that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected." Contrary to the above, since 2002, PPL failed to implement prompt and appropriate corrective actions to address continued non-compliance with Quality Assurance procedures which specify the requirements for the training and qualification of personnel performing activities affecting quality. Because this finding was found to be of very low safety significance (Green) and was entered into the PPL corrective action program (CR 760898), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as **NCV 05000387, 388/2006003-05, Ineffective Corrective Actions to Assure Training and Qualification of Workers as Required by 10 CFR 50 Appendix B, Criterion XVI.**

Observations

Inspectors determined that, in general, PPL was effective at identifying problems and placing them in the corrective action program. However, inspectors found that corrective actions were not effective in eliminating the contributing causes. PPL's effectiveness review of the actions taken prior to 2005 for training and qualification issues (CR 521474) originally concluded that the actions to address causes were not effective. This evaluation was later revised to recommend only a continuation of management oversight because PPL believed that the corrective actions resulted in a decrease in unqualified worker and training matrix condition reports in 2003 and 2004. Inspectors found that the prescribed management oversight did not eliminate the identified contributing causes.

Inspectors also observed an increasing trend of training and qualification issues at Susquehanna. Over the last four years there were over 30 unqualified workers identified as well as dozens of lower level qualification and training condition reports initiated each year. Inspectors found an increased significance to the condition reports and issues in the last two years. The PPL Corrective action process still contains a high influx of condition reports and some of the 2006 training and qualification issues are identical to previous condition reports from past years.

Inspectors observed that there were multiple examples, during the last 4 years, where the same or similar contributing factors were evident in the identified issues. The most significant causal factors included:

- Supervisors not verifying qualifications in accordance with NDAP-QA-500.
- Training Matrix and other databases not accurately reflecting the training qualifications of Nuclear Department Personnel.
- Individuals misunderstanding the available information in the Training and Qualficaiton Matrix (TMX).
- The tracking of individual qualifications by using methods that are outside of the TMX database including the addition of tools that were not verified to properly interface with TMX.
- Less than adequate impact analysis when training requirements, processes or data is changed to identify affected plant activities and the maintenance of qualification records.

Inspectors observed that these contributing causes contributed to the unqualified worker events in 2006.

Inspectors reviewed CR 730944 which was written on December 1, 2005 to address a NRC identified finding where a Fire Brigade Leader was assigned duty when his drill qualifications had expired. Inspectors found that after 6 months there were no actions taken to address the causes for this issue. In May, inspectors questioned why no actions had been taken, PPL implemented a peer review of the manual tracking method which provided an added barrier until the standard method of qualification tracking in TMX could be established. When the method of qualification tracking was later changed, six operators were removed from Fire Brigade duties, because they were not current with all the required training (CR 784838).

PPL has developed a plan to improve the overall training and qualification performance at Susquehanna. The apparent cause evaluation and corrective actions are provided in CR 760898. The actions are all scheduled for completion by June 30, 2006 and an effectiveness review is scheduled by November of 2006.

.5 Cross-References to PI&R Findings Documented Elsewhere

Section 1R17 describes a condition in which PPL did not fully evaluate the failure of the “C” ESW pump breaker in 2005, which resulted in not preventing a subsequent failure due to material degradation.

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

.1 (Closed) LER 05000387/2005-002-00 TS Required Shutdown Due to Excessive Control Cell Friction

On October 28, 2005, Susquehanna Unit 1 was shutting down for a planned maintenance outage to address control cell friction issues. Prior to shutdown, PPL's control rod testing program identified four control rods that were inoperable due to excessive control cell friction. PPL also identified other control rods that exhibited increased control cell friction, due to slow control rod settling times, but met Technical Specification requirements. PPL determined that to maintain the planned outage schedule they would not slow the shutdown to perform additional testing to prove the control rods were operable, instead they would declare the control rod inoperable and continue with the plant shutdown. Technical Specifications 3.1.3.F, "Control Rod Operability," require all control rods to be fully inserted 12 hours after determining that nine or more control rods do not meet Technical Specification requirements. At 11:32 p.m. on October 28, 2005, PPL declared the ninth control rod inoperable and entered Technical Specification 3.1.3.F, "Control Rod Operability." All control rods were inserted at 8:05 a.m., on October 29, 2005, maintaining compliance with Technical Specifications.

During this maintenance outage, PPL rechanneled 58 fuel assembles and performed testing prior to and during startup to verify control rod operability. PPL performed and continues to implement control cell friction testing in accordance with the "Channel Management Action Plan" (PL-NF-02-007). This management plan specifies test population, test methodology and test frequencies to ensure that control cell friction is identified before any control rod exceeds Technical Specification requirements so that actions can be taken to ensure that Technical Specification requirements are not exceeded. The LER was reviewed by the inspectors and no findings of significance were identified and no violation of NRC requirements occurred. PPL documented this condition in CR 721548. This LER is closed.

.2 (Closed) LER 05000387/2006-001-00 TS Required Shutdown Due to Excessive Control Cell Friction

On March 4, 2006, Susquehanna Unit 1 was shutting down for a refueling outage. Prior to shutdown, PPL identified several control rods that exhibited increased control cell friction, due to slow control rod settling times, but met Technical Specification requirements. PPL determined that to maintain the planned outage schedule they would not slow the shutdown to perform additional testing to prove the control rods were operable, instead they would declare the control rod inoperable and continue with the plant shutdown. Technical Specifications 3.1.3.F, "Control Rod Operability," require all control rods to be fully inserted 12 hours after determining that nine or more control rods do not meet Technical Specification requirements. At 5:17 a.m. on March 17, 2006, PPL declared the ninth control rod inoperable and entered Technical Specification 3.1.3.F, "Control Rod Operability." All control rods were inserted at 7:43 a.m., on March 4, 2006 maintaining compliance with Technical Specifications.

During the refueling outage, PPL rechanneled three fuel assembles, replaced 292 fuel assemblies and performed testing prior to and during startup to verify control rod operability. PPL performed and continues to implement control cell friction testing in accordance with the “Channel Management Action Plan” (PL-NF-02-007). This management plan specifies test population, test methodology and test frequencies to ensure that control cell friction is identified before a control rod exceeds Technical Specification requirements so that actions can be taken to ensure that Technical Specification requirements are met. The LER was reviewed by the inspectors and no findings of significance were identified and no violation of NRC requirements occurred. PPL documented this condition in CR 756415. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

.1 Cross Reference to Human Performance Findings Documented Elsewhere

Section 1R20.2 describes an NCV because PPL did not ensure supervisory and management oversight of work activities, including contractors such that there would be no adverse system interface issues in the PRNMS design which supports nuclear safety.

4OA5 Other Activities

.1 IP 92709, Licensee Strike Contingency Plans

a. Inspection Scope

The inspectors performed IP 92709, “Licensee Strike Contingency Plans.”
The inspectors reviewed PPL procedures and supporting information pertaining to the continued operation of the facility under strike conditions. This inspectors reviewed each department’s plan and compared the plan to plant Technical Specification, Final Safety Evaluation Report, the Emergency Plan, and regulations (10 CFR 50.54(m) and 10 CFR 55 for licensed operators). The review included consideration of overtime needed to maintain shift coverage and skill proficiency.

b. Findings

No findings of significance were identified.

.2 Implementation of Temporary Instruction (TI) 2515/165 - Operational Readiness of Offsite Power and Impact on Plant Risk

a. Inspection Scope

The objective of TI 2515/165, “Operational Readiness of Offsite Power and Impact on Plant Risk,” was to gather information to support the assessment of nuclear power plant operational readiness of offsite power systems and impact on plant risk. The inspector evaluated PPL procedures against the specific offsite power, risk assessment, and system grid reliability requirements of TI 2515/165.

The information gathered while completing this TI was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation on April 3, 2006.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On June 16, 2006 and July 14, 2006, the resident inspectors presented the inspection results to Mr. B. McKinney, Senior Vice President, and Chief Nuclear Officer, and other members of his staff, who acknowledged the findings. Susquehanna management stated that none of the information reviewed by the inspectors was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

K. Backenstoe, Senior Engineer
M. Baughman, Manager Nuclear Training
D. Brophy, Senior Engineer, Nuclear Regulatory Affairs
L. Casella, System Engineer
S. Cook, Quality Assurance Manager
A. Fitch, Shift Operations Supervisor
J. Fritzen, Radiological Operations Supervisor
L. Frace, Maintenance Supervisor
L. Fuller, Senior Engineer, System Design Engineering
E. Gerlach, ASME Engineer
J. Grisewood, Manager Corrective Action and Assessment
R. Hock, Radiological Operations Supervisor
L. Humpf, Chemistry Foreman
J. Jeanguenat, System Engineer
J. Kelly, Civil/Structural Design Supervisor
R. Kessler, Senior Health Physicist - ALARA
E. Miller, Senior Engineer, Nuclear Regulatory Affairs
J. Paciotti, Security Operations Supervisor
R. Saccone, Vice President - Nuclear Operations
V. Schuman, Radiological Protection Manager
R. Sheranko, Senior Component Engineer
S. Sienkiewicz, NDE Supervisor
L. Supon, Mechanical Systems Engineer
J. Van Horn, Production Supervisor - FIN
J. Williams, Senior Reactor Operator
D. Zaprazny, Supervising Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Opened and Closed

05000387/2006003-01	NCV	Inadequate Procedures Resulted in Motor Operated Valve Failures. (Section 1R15)
05000387, 388/2006003-02	NCV	Failure to Identify Material Degradation Which Resulted in a Failure of the “C” ESW Pump Breaker. (Section 1R17)

05000388/2006003-03	NCV	Inadequate Procedure Results in Elevated Reactor Coolant System Leakage. (Section 1R20.1)
05000387/2006003-04	NCV	Inadequate Design Review of PRNMS Modification Resulted in a Reactor Scram. (Section 1R20.2)
05000387, 388/2006003-05	NCV	Ineffective Corrective Actions to Assure Training and Qualification of Workers as Required by 10 CFR 50 Appendix B, Criterion XVI. (Section 4OA2.4)

Closed

05000387/2005-002-00	LER	TS Required Shutdown Due to Excessive Control Cell Friction (Section 4OA3.1)
05000387/2006-001-00	LER	TS Required Shutdown Due to Excessive Control Cell Friction (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED
(Not Referenced in the Report)

Section 1R01: Adverse Weather Protection

GN-000-005, hot weather

Section 1R04: **Equipment Alignment**

M - 151, Unit 1 P&ID residual heat removal
NDAP-QA-0308, Rev. 4, “Removal of 1 Control Rod Drive Mechanism for Maintenance During Operation Mode 4 or 5

Section 1R05: Fire Protection

FP-013-168, revision 5
FP-013-169, revision 4
FP-013-170, revision 5
FP-013-171, revision 4
FP-013-175, revision 5
FP-013-236, revision 5
FP-213-289, revision 4
NTP-QA-53.1, Attachment D, Fire Brigade Drill Critique Form
Fire Brigade Quarterly Drill, Scenario # 44, revision 0

Section 1R07: Heat Sink Performance

2B RCIC heat exchanger Inspection Report (WO 583371)
Engineering Work Request 621195, review of ECCS and RCIC room cooler inspection interval
NIMS data sheet for 2B RCIC room cooler 2E228B

Section 1R12: Maintenance Effectiveness

PLI-93543, Maintenance Rule Expert Panel Meeting 2005-0523 Minutes
Maintenance Rule Basis Document, System 02, revision 3
System Health Report, System 102 & 202, June 2006
PM Work Instruction E1923-53, dated July 15, 2004
PM Work Instruction E1924-53, dated July 15, 2004
CR 665179, 782620, 782287, 783734

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

AR 777192
OP-AD-338
EWR-777476

Section 1R15: Operability Evaluations

NDE Report ISI-06-095 thru ISI-06-109
OFR 784933, revision 0
CR 784890

Section 1R17: Permanent Plant Modifications

10 CFR 50.59 Applicability Determinations

A-01-2144, Unit 2 EPU Upgrades for EHC System, 10/18/2005

Safety Evaluations

E-01-31, Manual Positioning of Normal Feedwater Heater Drain Valves, Rev. 0
E-01-41, Install NUMAC Power Range Neutron Monitoring System (PRNMS), Unit 1, DCP 618882, Rev. 0
92-3020, Snubber Elimination - Unit 1/Phase II, DCP 92-3020A, D, F, H and I, Rev. 0
E-01-38, FSAR Internal Flood Design Basis Clarifications, Rev. 0
E-01-46, LDCN 4313 – Revision to Recirculation Pump Seizure Analysis in Susquehanna FSAR Section 15.3.3, Rev. 0
E-01-42, Clarification of Table B 3.6.1.3-1, footnote (b), Rev. 0
E-01-47, EPU High Pressure Turbine Replacement Engineering Changes 677648 and 677650 - Updated Turbine Missile Analysis, Rev. 0

10 CFR 50.59 Screened-out Evaluations

5059-01-812, Implementation of POWERPLEX-III, Rev. 1
5059-01-1080, Elimination of 10 CFR 50.59 & 72.48 Screens for Equivalent Changes, Rev. 0
5059-01-2406, Exception to 10 CFR 50 App. J Testing Requirements for RHR CIVs 1(2)F050A(B) and 1(2)F122A(B), Rev. 0
5059-01-2409, Reactor Feed Pump Suction Flange Changeout Units 1 and 2 - EPU, Rev. 0
5059-01-2372, Instrument Line Break Change to Original Licensing Bases, Rev. 0
5059-01-2373, Re-analysis of Instrument Line Break, Rev. 0
5059-01-1695, Loss of Reactor Building Closed Cooling Water, Rev. 0
5059-01-2341, FSAR Section 2.2.3.1.3, Toxic Chemicals, Rev. 0
5059-01-1512, Unit 1 ESW Isolation Valve Replacement Including LDCN 3590, Rev. 0
5059-01-1877, Fire Protection Cross-Tie to Condensate Transfer, Rev. 0
ECO 696686, Residual Heat Removal and Recirculation Tubing Relocation, Rev. 0
5059-01-2105, Defense in Depth for Residual Heat Removal Keepfill, Rev. 0
5059-01-1675, RHR Suppression Pool Cooling, Rev. 0
5059-01-1895, 4Kv Breaker Replacement Project, Rev. 0
5059-01-2054, Diesel Generator's A, B, C, D, E Governor Upgrade, Rev. 0

Audits and Self-Assessments

Self Assessment Report NRA-06-03, Evaluate Implementation of the 50.59 Program at SSES, dated June 1, 2006
Susquehanna Steam Electric Station Independent Technical Review 575618 Modification Assessment PLI 93185, dated July 27, 2004

Calculations

EC-PIPE-16204, Addition of ½" Vent SPGBB107-3 for HV151F016A, Rev. 0
EC-004-1025, Breaker Scheme Control Logic Review for Cutler-Hammer Series VR 4Kv Vacuum Breakers, Rev. 0
EC-002-0663, 125VDC Utilization Voltage and Load Profile for Circuit 1D61, Rev. 3
0109-0057-DLH1, Evaluation of SSES EDG Frequency Response During Design Basis LOOP/LOCA Transient Loading with Replacement Woodward 2301A Governor, Rev. 0

System Health Reports and Trending Data

054 - ESW Emergency Service Water, 2nd Quarter 2006
CPG - Electrical Breakers, 2nd Quarter 2006

Vendor Information

IB 32-253-4B, Instructions for Porcel-line Type Westinghouse DHP Magnetic Air Circuit Breakers
Instructions for Installation, Operation and Maintenance of Type DHP-VR Vacuum Replacement Circuit Breakers for DHP Switchgear, 11/2003

Corrective Action Reports

757465	527456	697527	706523
764746	592744	561961	720713
782885	764471	780266	730470
635250	600070	556423	768740
683621	574643	570477	531339
677958	579984	686618	
755237	577451	532815	
768343	580339	699219	

Drawings

J-400, Control Structure Room Arrangement and Panel Location, Sheet 2A, IDCN 3
J-400, Control Structure Room Arrangement and Panel Location, Sheet 2A, IDCN 4
SE-149-310, Hydrostatic Pressure Test Diagram RHR System-A, Rev. 0
GBB-107-1, Isometric-Reactor Bldg. RHR Unit No.1, IDCN 12
M-151, Unit 1 P&ID Residual Heat Removal, Rev. 60
E106260, High Pressure Coolant Injection, Rev. 48
D107295, Unit Common Schematic Emergency Service Water Pump C, Rev. 41

Miscellaneous

Risk-Informed Inspection Notebook For Susquehanna Station, Rev. 2
Regulatory Guide (RG) 1.187, Guidance For Implementation of 10 CFR 50.59, Changes, Tests, and Experiments, dated November 2000
Susquehanna Unit 1 Final Safety Analysis Report, Rev. 61
Susquehanna Unit 2 Final Safety Analysis Report, Rev. 61
NEI 96-07, Guidelines For 10 CFR 50.59 Implementation, Rev. 1
Non Destructive Examination Report # ISI-04-045
Licensing Document Change Notice 4326, Excess Flow Check Valve Leakage Criteria
Safety Guide 11, Instrument Lines Penetrating Primary Reactor Containment, dated 3/10/71
Replacement Item Evaluation 91-0075, Rev. 0
Design Criteria 18874-SRP-001Q, Design Criteria for the Snubber Elimination Program, Rev. 5
TP-178-040, Post Modification Test for PRNMS Operability, Rev. 0
SE-000-017, ASME Leak Inspection for ASME Class 1,2, and 3 Piping and Components, Rev. 1, performed 3/23/04
SE-000-017, ASME Leak Inspection for ASME Class 1,2, and 3 Piping and Components, Rev. 1, performed 3/11/04
Z1205-01, RHR System Leakage Test 149-301, Rev. 10, performed 3/27/06
SE-149-400, RHR System Leak Quantification Per SE-149-400, Rev. 8, performed 3/27/06
SE-100-002, ASME Class 1 Boundary System Leakage Test, Rev. 18, performed 4/4/06
EC-013-1865, Cross-Tie Flow from Fire Protection to RHR for Alternate Low Pressure Makeup to the Reactor Pressure Vessel, Rev. 0
GE-NE-0000-0048-3167-RO, Engineering Evaluation of Pump Seizure Event for Susquehanna Steam Electric Station Units 1 and 2 EPU/MELLLA+ Task 0900B, February 2006

WO 580010, Performed ISI at Unit 1 Cross Tie Piping between Fire Protection and Condensate Transfer
EC-FLOD-0500, Re-Evaluate Maximum Flood Depth in Reactor Building Piping/Penetration Room on Elevation 683', Rev. 2
EC-FLOD-0001, Internal Flood Evaluation from Moderate Energy Pipe Cracks and Sprinkler System Actuations, Rev. 1
SO-054-015, Two Year ESW RPI, Rev. 8, performed 3/11/04
SO-054-015, Two Year ESW RPI, Rev. 8, performed 3/23/04
30-054-004, 9207 ESW/TBCCW/RBCCW Isolation Valve Exercising, Rev. 9, performed 3/23/04
TP-054-097, Quantification of Leakage Past ESW to SW Swap-over Valves, Rev. 1, performed 3/11/04
TP-054-097, Quantification of Leakage Past ESW to SW Swap-over Valves, Rev. 1, performed 3/23/04
TP-054-99, Directing ESW Flow through CCW Heat Exchangers, Rev. 0, performed 3/11/04
E239584, Susquehanna Division 1 Operable Boundary Emergency Service Water, Rev. 2
OP-149-003, RHR Operation In Fuel Pool Cooling, Rev. 21
ON-164-002, Loss of Reactor Recirculation Flow, Rev. 23
ES-150-003, RCIC Manual Injection with Loss of AC and DC Power, Rev. 2
Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 164 to Facility Operating License No. NPF-11 and Amendment No. 150 to facility Operating License No. NPF-18
ON-135-001, Loss of Fuel Pool Cooling/Coolant Inventory, Rev. 25
ON-037-001, Loss of Condensate Transfer System, Rev. 5
General Electric standard Application For Reactor Fuel (Supplement for United States)
Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Rev. 1
E1113, Replacement 4Kv Vacuum Breakers, Rev. 2
CKTBRK-1, PPL Preventive Maintenance Background Information 13.8 and 4Kv Breakers, Rev. 0
TR-106857-V2, Preventive maintenance program basis: Medium Voltage Switchgear, 8/1996
TR-1000010, Routine Preventive and Condition-Based maintenance for Westinghouse DHP Circuit Breakers, Rev. 1

Operating Experience

IN 93-26, Grease Solidification Causes Molded Case Circuit Breaker Failure to Close, 4/7/1993
IN 98-38, Metal-Clad Circuit Breaker Maintenance Issues Identified by NRC Inspections, 10/15/1998

IN 95-22, Hardened or Contaminated Lubricants Cause Metal-Clad Circuit Breaker Failures, 4/21/1995
IN 99-13, Insights from NRC Inspections of Low and Medium Voltage Circuit Breaker Maintenance Programs, 4/29/1999

Modifications

392620, POWERPLEX-III Hardware, Rev. 0
487206, Addition of Vent Valve to Division I RHR, Rev. 0
541697, Replace HV 11315 and HV 21315, Rev. 1
762459, Doc Only - Change Excess Flow Check Valve Leakage Criteria Bounded Technical Requirement, Rev. 0
432625, Unit 1 ESW Isolation Valve Replacements, Rev. 2
556794, Fire Protection Cross Tie to Condensate Transfer, Rev. 0
696686, Residual Heat Removal and Recirculation Tubing Relocation, Rev. 0
602544, Diesel Generator “A” Governor Replacement, Rev. 0
486204, 4Kv Breaker Replacement, Rev. 2

Procedures

MFP-QA-4002, Drawing Control and Classification, Rev. 11
NDAP-QA-0206, Replacement Item Evaluation, Rev. 0
NDAP-QA-1220, Engineering Change Process, Rev. 0
NDAP-QA-0726, 10CFR50.59 and 10CFR52.48 Implementation, Rev. 9
NDAP-QA-0524, Equipment Reliability and Station Health Process, Rev. 4
NEIM-00-1172, Nuclear Design Engineering Quality Review Team, Rev. 0
5059RM, 10 CFR 50.59 Resource Manual, Rev. 8
MT-GE-005, Circuit Breaker and Switchgear Inspection and Maintenance 5 and 15 KV, Rev. 23
MT-GE-048, Cutler-Hammer Type DHP-VR 4.16KV Circuit Breaker and Switchgear Inspection and Maintenance, Rev. 4

Completed Surveillances

TP-024-163, “A” Diesel Generator Governor Replacement Post Maintenance Testing, performed 9/21/05
TP-024-145, Diesel Generator “A” Restoration, performed 9/28/05
TP-043-002, Initial Installation of Unit 1 and Unit 2 Mechanical Vacuum Pump Vacuum Circuit Breakers, performed 2/4/05
TP-054-098, Initial Installation of ESW Pump Vacuum Circuit Breakers, performed 5/23/06

Work Orders

511302	582231	582276

Section 1R19: Post Maintenance Testing

SM-102-E03, performed June 6, 2006
RTSV 640556
RLWO 715966

Section 2: Radiation Safety

Condition Reports:
751697; 755883; 755600; 755780; 756406; 756517; 756628; 756635; 756690; 756690; 756718; 756759; 756780; 756786; 756788; 756791; 757080; 757085; 757213; 757247; 757250; 757251; 757605; 757613; 757673; 757709; 757729; 757775; 757777; 758054; 758093; 758500; 758828; 758999; 759116; 759214; 759216; 759230; 759303; 759628; 759686; 759753; 759806; 760395; 760549; 760948; 761254; 761319; 761662; 761846; 762692; 762950; 763150; 763767; 763857; 764171; 764366; 764685; 764872; 765305; 765409; 765869; 766279; 766600; 767275; 767952; 768199; 768352; 768461; 768724; 769166; 769179; 769248; 769836; 769876; 769869; 770300; 771209; 771218; 771471; 771549; 771781; 772657; 772991; 773035; 773147; 773236; 774740; 775932; 775990; 776014; 776328; 776637; 776901; 778177; 778191; 778690; 778842; 780782; 784409; 784456; 784675

Y2005 Condition Report Trend Review

ALARA In-Progress Reviews: 20060200; 20061320-0; 20061320-1; 20061360; 20061370; 20061383

Uni1 1 14th Refueling and Inspection Radiological Performance Report (DRAFT)

Self-Assessments: AR 751895/HPS-06-05

QA Performance Assessments: AR 692957; AR 752337; AR 754503; AR 773271

Updated Final Safety Analysis Report, Section 12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Section 4OA2: Identification and Resolution of Problems

PPL Trend Reports:

Corrective Action First Quarter Trend Report MRA 754573
CR 783655

Recurrent Safety Related Equipment Failures:

CR 753990, Control Rod Drive Power Supply Failure, (NCV 05000388/2006002-002)
CR 768740, “C” ESW Pump Failed to Start, (this report section 1R17, 4OA2.3
CR 766551, MOV Stem Nuts, (this report section 1R15, NCV 05000387/2006003-001)
CR 783655, Quarterly Trend Report Identified an Emerging Adverse Trend in Station Implementation of the Corrective Action Program

Condition Reports:

CR 552196, CR 699219, CR 730944, CR 443446, CR 503826, CR 521474, CR 538594, CR 735096, CR 545542

Work Orders:

700059, 552211,

USA-70885: Failure Analysis of an ABB Relay
NDAP-QA-1304 Rev 0, Management of Worker Qualification Tracking System Procedure (TMX)
3.1, OPS 6, Qualification Training and Certification of Personnel
3.2, NDAP-QA-0010, Minimum Qualifications and Training Requirements
3.3, NTP-QA-13.2, Nuclear Training Groups Records Management
AR 780065, Welders trained in support of dry fuel storage 2006 campaign were given complete credit without being trained and evaluated on all of the MM 350 course objectives
Effectiveness review of CR 521474 CRA 538594
NTG Self-Assessment NTG-04-14

Section 4OA5: IP 92709, Licensee Strike Contingency Plans

Susquehanna Steam Electric Station (SSES) Business Continuity Plan (Labor Disruptions)
Sections: Security Rev. 4
Chemistry Rev. 2
Operations, Rev. 2
Station Engineering, Rev. 1
Configuration Management, Rev. 1
Nuclear Design Engineering, Rev. 1
Site Access Services, Rev. 0
Nuclear Facilities Management, Rev. 1
Nuclear Regulatory Affairs, Rev. 0
Nuclear Fuels and Analysis, Rev. 1

E-Plan, Rev. 46, Table 6.1
TS, Section 5.2.2, “Administrative Staff”
TRO, Section 4.1“Administrative Controls”
NDAP-QA-0300, Rev. 20, “Conduct of Operations”

Qualification Guides:
Business Continuity Qualification Document - SRO to NPO Level 1
Business Continuity Qualification Document - SRO to NPO Level 2
Business Continuity Qualification Document - SRO to NPO Level 3
Business Continuity Qualification Document - SRO to NPO Level 4
Business Continuity Qualification Document - SRO to NPO Level 5

Section 4OA5: TI 2515/165 - Operational Readiness of Offsite Power and Impact on Plant Risk

SO-024-013, Offsite Power Source and Onsite Class 1E Operability Test, Rev. 14
OI-TA-008, Shift Technical Advisor Responsibilities, Rev. 6
OI-AD-029, Emergency Load Control, Rev. 10
NDAP-QA-1902, Maintenance Rule Risk Assessment and Management Program, Rev. 0
NDAP-00-1912, Scheduling and Coordination of Work, Rev. 5

LIST OF ACRONYMS

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document and Access Management System
ALARA	As Low As Is Reasonably Achievable
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
EDG	Emergency Diesel Generator
EPA	Electric Power Monitoring Assemblies
EPU	Extended Power Uprate
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
gpm	Gallons per Minute
HPCI	High Pressure Coolant Injection
IP	Inspection Procedure
Kv	Kilovolts
LDCN	Licensing Document Change Notice
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MOV	Motor-Operated Valve
NCV	Non-cited Violation
NDAP	Nuclear Department Administrative Procedure
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OE	Operating Experience
OOS	Out-Of-Service
PCWO	Plant Component Work Orders
PI	[NRC] Performance Indicator
PI&R	Problem Identification and Resolution
PM	Preventive Maintenance
PPL	PPL Susquehanna, LLC
PRNMS	Power Range Neutron Monitoring System

QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RFP	Reactor Feed Pump
RG	[NRC] Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RR	Reactor Recirculation
RSPS	Risk Significant Planning Standard
RTP	Rated Thermal Power
SDP	Significant Determination Process
SE	Safety Evaluation
SSC	Structures, Systems, and Components
SSES	Susquehanna Steam Electric Station
TBCCW	Turbine Building Closed Cooling Water
TS	Technical Specifications
VDC	Volts Direct Current
WO	Work Order