November 7, 2003

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000387/2003004 AND 05000388/2003004

Dear Mr. Shriver:

On September 27, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Susquehanna Steam Electric Station Units 1 and 2. The enclosed integrated inspection report presents the results of that inspection, which was discussed with Mr. R. Anderson, Vice President - Nuclear Operations, and other members of your staff on October 17, 2003.

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

This report documents an unresolved item related to the ineffective implementation of maintenance work instructions to tighten a linkage connecting bolt on the "D" emergency diesel generator. The bolt connects the governor positioner arm to the fuel supply rack. The error resulted in an inoperable emergency diesel generator. This finding is unresolved pending completion of a risk significance determination. This finding does not present an immediate safety concern because, as a follow-up corrective action, PPL appropriately tightened the bolt on the "D" emergency diesel generator and verified that the same bolt on the remaining four emergency diesel generators were also adequately tightened.

In addition, one NRC identified finding and one self-revealing finding of very low safety significance (Green) were identified. The findings were determined to involve violations of NRC requirements. A licensee-identified violation that was determined to be of very low safety significance is listed in Section 4OA7 of this report. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these findings as a non-cited violations (NCVs), consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148 “Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures,” and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by the order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) ‘02, and the remaining inspection activities for Susquehanna were completed in April 2003. The NRC will continue to monitor overall safeguards and security controls at Susquehanna.

In accordance with 10CFR2.790 of the NRC’s "Rules of Practice," a copy of this letter and 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).

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

Sincerely,

/RA/

Mohamed Shanbaky, Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 05000387/2003004, 05000388/2003004
w/Attachment: Supplemental Information

cc w/encl: J. H. Miller, President - PPL Generation, LLC
R. L. Anderson, Vice President - Nuclear Operations for PPL Susquehanna LLC
R. A. Saccone, General Manager - Nuclear Engineering
A. J. Wrape, III, General Manager, Nuclear Assurance
T. L. Harpster, General Manager - Plant Support
K. Roush, Manager, Nuclear Training
G. F. Ruppert, Manager, Nuclear Operations
J. D. Shaw, Manager, Station Engineering
T. P. Kirwin, Manager, Nuclear Maintenance
R. M. Paley, Manager, Work Management
Director, Bureau of Radiation Protection
R. E. Smith, Jr., Manager, Radiation Protection
W. F. Smith, Jr., Manager, Corrective Action & Assessments
D. F. Roth, Manager, Quality Assurance
R. R. Sgarro, Manager, Nuclear Regulatory Affairs
R. Ferentz, Manager - Nuclear Security
C. D. Markley, Supervisor - Nuclear Regulatory Affairs
W. E. Morrissey, Supervising Engineer
M. H. Crowthers, Supervising Engineer
H. D. Woodeshick, Special Office of the President
B. A. Snapp, Esquire, Associate General Counsel, PPL Services Corporation
R. W. Osborne, Allegheny Electric Cooperative, Inc.
Board of Supervisors, Salem Township
J. Johnsrud, National Energy Committee
Supervisor - Document Control Services
Commonwealth of Pennsylvania (c/o R. Janati, Chief, Division of Nuclear Safety, Pennsylvania Bureau of Radiation Protection)

Distribution w/encl: H. Miller, RA/J. Wiggins, DRA (1)
M. Shanbaky, DRP
D. Florek, DRP
S. Hansell, DRP - SRI Susquehanna
J. Richmond, DRP - RI Susquehanna
J. Jolicoeur, RI EDO Coordinator
R. Laufer, NRR
R. Guzman, NRR
R. Clark, PM, NRR (Backup)
Region I Docket Room (with concurrences)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.: 05000387, 05000388

License Nos.: NPF-14, NPF-22

Report No.: 05000387/2003004, 05000388/2003004

Licensee: PPL Susquehanna, LLC

Facility: Susquehanna Steam Electric Station

Location: 769 Salem Boulevard
Berwick, PA 18603

Dates: June 29, 2003 to September 27, 2003

Inspectors: S. Hansell, Senior Resident Inspector
J. Richmond, Resident Inspector
F. Jaxheimer, Resident Inspector
M. Ferdas, Resident Inspector, Hope Creek
J. Noggle, Senior Health Physicist

Approved by: Mohamed M. Shanbaky, Chief
Projects Branch 4
Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000387/2003-004, 05000388/2003-004; 06/29/2003 - 09/27/2003; Susquehanna Steam Electric Station, Units 1 and 2. Personnel Performance During Non-Routine Plant Evolutions and Identification and Resolution of Problems.

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

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- **Green.** A self-revealing event resulted in a non-cited violation of Technical Specification section 5.4.1, because a plant control operator did not implement operating procedure OP-145-001, "Reactor Feed Pump and Reactor Feed Pump Lube Oil System," Section 2.11, "Emergency Governor and Trip Lockout Exerciser Test," as written for the "C" reactor feed pump. As a result, the "C" reactor feed pump tripped and the reactor automatically shutdown due to low reactor vessel water level.

  This finding is greater than minor because it is similar to example 4.b in NRC Inspection Manual 0612 Appendix E, "Examples of Minor Issues." In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspector determined that the finding was of very low safety significance (Green) using a Phase 2 significance determination process evaluation.

  A contributing cause of this finding is related to the Human Performance cross-cutting area, in that a control room operator did not follow a reactor feed pump test procedure. As a result, the "C" reactor feed pump tripped and the reactor automatically shutdown due to low reactor water level.

  A second contributing cause of this finding is related to the problem identification and resolution cross-cutting area because PPL did not take meaningful corrective actions for a September 1999 trip of a reactor feed pump while performing the same emergency governor trip test. The same probable cause for the 1999 reactor feed pump trip, self/peer checking standards not applied, was the same probable cause of the 2003 event. (Section 1R14)

Summary of Findings (cont'd)

Cornerstone: Mitigating Systems

- **Green.** The inspectors identified a non-cited violation of 10 CFR 50 Appendix B Criterion XVI of very low safety significance (Green). On March 19, 2003, PPL returned the "D" EDG to an operable status without adequately determining the cause of a linkage connecting bolt to fall off. The bolt connects the governor positioner arm to the fuel supply rack. PPL's initial repair was not sufficient to prevent repetition because although the bolt was initially reinstalled, it was not tightened to the required torque value of 25-30 foot pounds as required by the diesel vendor and plant procedures. On March 21, 2003, PPL removed the EDG from service and tightened the linkage connecting bolt to the required torque value.

  This finding is greater than minor because it adversely impacts the equipment performance attribute of the mitigating system cornerstone and adversely affects the cornerstone objective in that the finding is associated with the reliability of the "D" EDG to respond to initiating events and prevent core damage. This finding is of very low safety significance (Green) using phase one of the significance determination process. The issue does not result in an actual loss of safety function of a system or the loss of safety function of a single train for greater than the Technical Specification allowed outage time of 72 hours. In addition, the finding is not risk significant due to seismic, fire, flooding, or severe weather initiating events.

  This finding is related to the Problem Identification and Resolution (PI&R) cross-cutting area because PPL did not identify the cause of a significant condition adverse to quality and take corrective actions to prevent recurrence. (Section 4OA2)

- **TBD.** The inspectors identified an unresolved item related to Technical Specification 5.4.1. On July 5, 2000, PPL did not implement written work instructions to tighten a linkage connecting bolt on the "D" emergency diesel generator (EDG) between the governor and the fuel supply rack linkage. As a result, on March 19, 2003, after approximately 205 diesel run hours, the bolt fell out and disconnected the governor from the fuel rack during an EDG monthly surveillance run. This finding is greater than minor because it relates to the equipment performance attribute of the mitigating systems cornerstone and adversely affects the cornerstone objective, in that, the "D" EDG was incapable of performing its safety function for a period of time in excess of its Technical Specification allowed outage time. This finding is unresolved pending completion of a significance determination.

  This finding is related to the Problem Identification and Resolution cross-cutting area because PPL did not identify this performance deficiency during their corrective action cause review for the "D" EDG in-service failure of March 19, 2003.

  This finding is also related to the Human Performance cross-cutting area, because maintenance technicians did not adequately implement written work instructions. (Section 4OA2)

B. Licensee Identified Violations

Summary of Findings (cont'd)

Violations of very low safety significance, which were identified by PPL, have been reviewed by the inspectors. Corrective actions taken or planned by PPL have been entered into PPL's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 began the inspection period at full power. On September 11, reactor power was reduced to 63% due to a fire at the "B" reactor feedwater pump turbine. Reactor power was returned to 100% on September 15th. On September 24th, a loss of the "C" reactor feedwater pump resulted in a reactor automatic shutdown. The reactor remained shutdown in a maintenance outage at the end of the period.

Unit 2 operated at or near full power during the inspection period except for control rod pattern adjustments and main turbine control valve testing.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed PPL's preparations for high winds and severe thunderstorm weather conditions and performed plant walkdowns for selected structures, systems, and components. The walkdowns and reviews were conducted to determine the adequacy of PPL's weather protection activities and system features. The inspectors reviewed and evaluated plant conditions related to the severe weather and PPL's risk assessment. This inspection activity represented two samples. The events occurred on the following dates:

- Loss of plant auxiliary power due to severe thunder and lightning storms on July 22, 2003;
- Plant response to tropical storm Isabel and loss of station auxiliary power on September 18 and 19, 2003.

Procedures and Documents

- Emergency action level (EAL) for sustained high winds
- EP-RM-001, revision 4, "EAL Technical Basis"
- NDAP-00-0024, "Winter Operation Preparations and Severe Winter Weather Operation"
- ON-000-002, "Natural Phenomena"

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

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

- "A," "B," "D," and "E" EDGs, while "C" EDG inoperable due to slow start time, CR 484120
- Emergency Service Water supply to TBCCW and RBCCW Units 1 and 2, CRs 485442 and 485476
- Security control center during a loss of station auxiliary power

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

Routine Plant Area Inspections. 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. This inspection activity represented six samples. The areas and documents reviewed included:

Plant Areas and Fire Zones

- Unit 1 "B" RHR pump room during system outage work
- "A" EDG room during the delay of a planned overhaul
- Unit 1 RHR LPCI line to review missing pipe insulation
- Units 1 and 2 reactor building during tropical storm Isabel impact on Susquehanna
- Units 1 and 2 upper relay rooms
- Unit 1 barriers between the "A," "B" and "C" RFPT rooms and the main condenser

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

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

- FSAR Section 9.3.3, "Equipment and Floor Drainage System"
- ON-169-002, "Flooding in the Reactor Building"
- Design Basis Document DBD-010, " HELB, MELB, and Internal Flooding"
- Maintenance Rule Basis Document for Plant Leak Detection System-76D

a. Findings

No significant findings were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On September 8, 2003, the inspectors observed licensed operator performance in the simulator during the operator re-qualification training. The inspectors compared their observations to Technical Specifications, emergency plan implementation, and the use of emergency operating procedures. The inspectors' evaluation focused on the operating crew's satisfactory completion of crew critical tasks, and satisfactory implementation of the emergency plan and emergency action level (EAL) classifications for the simulated plant conditions. Critical tasks are operational limits placed on key reactor plant and containment parameters that will ensure safety margins are maintained during the simulated malfunctions. The review included a comparison of the simulator's ability to model the actual plant performance. The inspectors also evaluated PPL's critique of the operators' performance to identify discrepancies and deficiencies in operator training. This inspection activity represented one sample. The observed training scenario included:

- OP003-06-01, "Transfer of Start-up Bus 10 to T-10, Momentary Loss of 1A ESS Bus with Loss of "A" RPS Bus, Loss of Cooling to "A" Reactor Recirculation Pump, Recirculation Pump Dual Seal Failure, and Primary Containment Control"

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12Q)

a. Inspection Scope

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

Equipment Issues

- "B" RHR pump system outage window
- Unit 2 "A" CRD pump replacement due to degraded discharge pressure

Procedures and Documents

- Maintenance Rule Bases Document for residual heat removal and control rod drive systems
- MT-GM-015, "Torque Guidelines"
- MT-GM-001, "Coupling Alignment (Horizontal Equipment)
- TP-055-014, "CRD Pump Performance Curve"
- Work orders 487043, 488468, and 423083

b. Findings

No significant observations or findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

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 issue. The inspectors performed control room and field walkdowns to verify whether the compensatory measures identified by the risk assessments were appropriately performed. This inspection activity represented five samples. The selected maintenance activities included:

- "C" EDG did not start in <10 seconds during monthly surveillance test, CR 484120
- Unit 2 feedwater level control system low load valve controller replacement, CR 486182, WO 437465 and RLWO 480684
- T-20 transformer tap changer failure to operate in automatic, CR 491482
- "A" EDG volt regulator failure during post maintenance testing, after overhaul
- Unit 1 HPCI out-of-service, concurrent with "B" RFPT fire and plant transient

b. Findings

No findings of significance were identified.

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

1. Unit 1 “B” Reactor Feed Pump Turbine Fire

a. Inspection Scope

On September 10, 2003, an oil leak on the Unit 1 "B" reactor feed water pump turbine resulted in a small fire. Operators reduced power to 70% to allow removal of the feedwater pump from service. While the feedwater pump was being removed from service, reactor water level decreased to 30 inches and an automatic reactor recirculation pump runback occurred which reduced reactor power to 63%. The plant fire brigade extinguished the fire in less than 10 minutes. PPL entered this into their corrective action program as condition report 508017.

The inspectors reviewed operating logs, core thermal power limits, plant procedures, and interviewed plant personnel for this issue to independently determine what occurred and evaluate the initiating cause. The inspectors assessed personnel performance during this event to evaluate whether the operator response was appropriate and in accordance with procedures and training. This inspection activity represented one sample.

Procedures and Documents

- ON-013-001, "Response to Fire"
- ON-164-002, "Loss of Reactor Recirculation Flow"
- OP-145-001, "RFP and RFP Lube Oil System"
- Event Notification Report 40147
- Condition Reports 508017 and 333197

b. Findings

No findings of significance were identified.

2. Unit 1 Reactor Scram Due to Loss of “C” Reactor Feed Pump

a. Inspection Scope

The inspectors reviewed an unexpected trip of the Unit 1 “C” reactor feedwater pump (RFP) due to an operator error during weekly RFP testing that resulted in an automatic reactor shutdown on September 24, 2003, at 12:53 a.m. The automatic reactor shutdown occurred due to reactor water level decreasing to +13 inches after the RFP trip. Normal reactor water level is +35 inches. Reactor power was 100% at the time of the RFP trip. When reactor level dropped to + 30 inches the “A” and “B” reactor recirculation pumps received an automatic runback signal and started to reduce power. The “A” and “B” RFPs increased feedwater flow to the reactor in response to the low reactor water level. The feedwater flow increase and recirculation pump power

reduction response did not prevent reactor water level from reaching the automatic reactor shutdown setpoint.

After the automatic shutdown, reactor water level dropped to - 48 inches. High pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems automatically started at - 38 inches as designed. The “A” and “B” RFPs were also in service injecting feedwater into the vessel. Reactor water injection from HPCI, RCIC, and the RFPs recovered water level to +35 inches within approximately two minutes and HPCI and RCIC were secured.

All control rods fully inserted into the core as designed. When reactor water level dropped to -38 inches, after the scram, both reactor recirculation pumps tripped as designed. All other containment isolation valves closed as designed. The reactor recirculation pumps were restarted at 1:20 a.m. to restore forced circulation through the reactor vessel. Reactor pressure was controlled with the main condenser bypass valves and no safety relief valves opened on the initial scram.

The inspectors reviewed the operator actions, plant response before and after the automatic shutdown, and plant procedures related to the “C” reactor feedwater pump trip and subsequent automatic actuation of the reactor protection system. The review focused on the operator performance during the weekly RFP test and the quality of the written procedure used to perform the test. This inspection activity represented one sample. The following documents were included in the review:

Procedures and Documents

- OP-145-001, "Reactor Feed Pump and RFP Lube Oil System"
- ON-164-002, “Loss of Reactor Recirculation Flow"
- ON-100-101, “Scram"
- Unit 1 Control Room Operator Logs
- EO-100-102, “RPV Control”
- Condition Reports 510950 and 200000

b. Findings

Introduction

A Green self-revealing NCV was identified because a plant control operator (PCO) did not implement the reactor feedwater pump operating procedure OP-145-001, “Reactor Feed Pump and RFP Lube Oil System,” as written in accordance with Technical Specification 5.4.1.a. The error resulted in a “C” RFP trip and a Unit 1 automatic reactor shutdown.

Description

On September 24, 2003, a self-revealing finding was identified when a PCO did not implement operating procedure OP-145-001, “Reactor Feed Pump and RFP Lube Oil System,” Section 2.11, “Emergency Governor and Trip Lockout Exerciser Test,” as written for the “C” RFP. Instead of re-setting the “C” RFP trip as discussed in step 2.11.8, the PCO moved the RFP turbine emergency governor key lock switch from “Lockout” to “Normal,” prior to verifying that the turbine trip signal was reset. As a result, the “C” RFP tripped and the Unit 1 reactor automatically shutdown due to low reactor vessel water level.

Operating procedure OP-145-001 contains clear direction to verify the RFP turbine trip was reset and includes a 30 second time delay prior to moving the RFP turbine emergency governor key lock switch from “Lockout “ to “Normal.” A second PCO was providing a peer check of each switch movement during the test, but could not prevent the operator error.

Analysis

The performance deficiency is that a plant control operator did not correctly implement a plant operating procedure. The finding was similar to example 4.b in Inspection Manual Chapter (IMC) 0612, Appendex E, “Example of Minor Issues” and was greater than minor because the procedure error resulted in a “C” RFP trip and a Unit 1 automatic reactor shutdown transient. This finding affects the Initiating Events Cornerstone because it resulted in a “C” RFP trip and a Unit 1 automatic reactor shutdown. This finding affects the Mitigating System Cornerstone because the finding is associated with the availability of one RFP, a component in a mitigating system.

In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspector determined that the finding was of very low safety significance (Green) using a Phase 2 significance determination process (SDP) evaluation. The Phase 1 SDP screening criteria specified a Phase 2 evaluation because the finding affected two cornerstones. A more detailed analysis for external events and large early release frequency were not needed because the increase in the risk of core damage was less than 1E-7 per year.

The Phase 2 evaluation used the Risk-informed Inspection Notebook for Susquehanna Units 1& 2, Revision 1, dated September 25, 2002, with the following assumptions:

1) One of the three feedwater pumps was unavailable; all other equipment supporting the function of the power conversion system feed (PCS feed) and late inventory makeup (LI) were available.
2) The exposure time was less than three days.
3) The only initiating event evaluated was TRANS because there was no effect on the LI mitigation capability of the condensate pumps in the SLOCA, SORV, MLOCA or ATWS initiating events.

4) Within the TRANS worksheet the initiating event frequency was taken from 1 to 0, because the initiating event occurred, and the PCS Mitigation Capability remained at full credit of 3, because two of the three feedpumps remained available.

The dominate transient core damage sequences included loss of PCS followed by either: a loss of containment heat removal and failure to vent the containment; or loss of high pressure injection and failure to depressurize the reactor. There was no impact on the capability to either: remove containment heat and vent the containment; or inject water at high pressure and depressurize the reactor.

A contributing cause of this finding was related to the Human Performance cross-cutting area, in that a control room operator did not follow a reactor feed pump test procedure. As a result, the “C” reactor feed pump tripped and the reactor automatically shutdown due to low reactor water level.

A second contributing cause of this finding was related to the Problem Identification and Resolution (PI&R) cross-cutting area because PPL did not take meaningful corrective actions for a September 1999 trip of a RFP while performing the same emergency governor trip test. The same probable cause for the 1999 RFP trip, self/peer checking standards not applied, was the probable cause of the 2003 event.

Enforcement

Technical Specification section 5.4.1 requires, in part, that written procedures shall be established and implemented as recommended in NRC Regulatory Guide (RG) 1.33, Revision 2, February 1978, Appendix A. RG 1.33 Appendix A, Section 4.0, "Feedwater System," is a system that meets the requirements of NRC's Regulatory Guide 1.33.

Contrary to the above, on September 24, 2003, a plant control operator (PCO) did not implement reactor feedwater pump (RFP) operating procedure OP-145-001, “Reactor Feed Pump and RFP Lube Oil System,” Section 2.11, “Emergency Governor and Trip Lockout Exerciser Test,” as written for the “C” RFP. Instead of re-setting the “C” RFP trip as required in step 2.11.8, the PCO moved the RFP turbine emergency governor key lock switch to from “Lockout” to “Normal,” step 2.11.11, prior to verifying that the turbine trip signal was reset. Because this violation is of very low safety significance and PPL entered this finding into their corrective action program (CR 510950), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy: **NCV 05000387/2003004-01, Did Not Properly Implement Feed Pump Test and Caused Feed Pump Trip.**

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability determinations that were selected based on risk insights, to assess the adequacy of the evaluations, the use and control of compensatory measures, and compliance with the Technical Specifications. In addition, the inspectors reviewed the selected operability determinations to verify whether the determinations were performed in accordance with NDAP-QA-0703, "Operability Assessments." The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report (FSAR), and associated Design Basis Documents as references during these reviews. This inspection activity represented six samples. The issues reviewed included:

- Unit 2 final feedwater temperature and turbine control valve position were different than expected for the Siemens turbine upgrade, CR 486104
- "D" EDG jacket water air entrainment during hot restart of the EDG engine, CR 486481 and 486501
- Containment isolation valves have minimal margin (HV 16108A1/A2 and HV 16116A1/A2)
- T-20 tap changer operability unknown, Division 2 class 1E 4160 volt buses are reading a high voltage, PCWO 491200
- "A," "B," "C" and "D" EDG lube oil and jacket water heat exchangers have possible undersized welds, CR 492649
- Unit 1 HPCI full flow test valve weld failure and subsequent repairs, CR 506771

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors reviewed the most significant control room deficiencies, status control tags, and selected corrective action reports to determine whether the functional capability of a system or a human reliability response during an event would be affected. The equipment performance and material condition list contains the station operator work-arounds. The inspectors evaluated the operators' ability to implement normal, off-normal, and emergency operating procedures during postulated plant transients with the existing equipment deficiencies. The review included an evaluation of the cumulative and synergistic effects of the identified operator work-arounds. This inspection activity represented two samples. The most risk significant operator work-arounds included:

- T-20 tap changer automatic voltage controller not working; T-20 tap changer is being operated in manual, CR 491482; operating procedure OP-003-001

- EDG Standby Jacket Water Pump one hour run after EDG operation to remove entrapped air from the jacket water system, CRs 487045 and 486481

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

- OP-024-001, Fast Start for the "C" EDG after troubleshooting slow start
- "D" EDG SO-024-001 and SE-024-001 section 6.6, following protective relay replacement and raise-lower switch replacement, CR 486233 and 486238
- SO-249-A02, "A" and "C" RHR Quarterly Flow Surveillance Test, after planned system outage
- Unit 1 feedwater level control system flow summer card replacement
- Unit 1 HPCI TP-152-032, "Instrumented Test Run of the HPCI Pump," performed after repairing failed weld on the full flow test valve

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- SI-251-202, Unit 2 Core Spray "B" Loop High Pressure In-leakage Quarterly Functional Test
- SE-024-001 “D” Emergency Diesel Integrated Test, largest load and full load reject tests
- “E” EDG Surveillance Test after “E” EDG was substituted for the “A” EDG, SO-024-001, August 19
- Unit 2 HPCI, SO-252-002 Quarter Flow Test, Unit 1 HPCI OOS due to adequate flow during Unit 1 flow test
- SI-180-308, “Unit 1 24 Month Calibration - RWCU, MSIV, PCIS, SCIS for reactor vessel water level 1 and 2 isolation

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On September 8, the inspectors observed a control room simulator based training event. The inspectors assessed licenced operator adherence to emergency plan implementing procedures, and their response to simulated degraded plant conditions to identify weaknesses and deficiencies in classification and notification. The inspectors observed 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 failures. This inspection activity represented one sample. The inspectors’ review included the following documents and procedures.

- Susquehanna Emergency Plan, revision 42
- EP-PS-100, "Emergency Director - Control Room"

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

The most recent radio-chemical radioactive waste stream analyses were reviewed for appropriate use in classifying waste shipments for transport in accordance with 10 CFR 61.55, which included: chemical wastes, liquid radwaste filter media, condensate filter waste, reactor water cleanup sludge, dry active waste, bead resin, fuel pool and recirculation system chemical decontamination wastes, and various mechanical filter wastes. Program processes to ensure continued validity of the 10CFR 61.55 samples during plant operation changes since the previous inspection in this area were also reviewed with respect to Branch Technical Position guidelines.

On August 6, 2003, the inspector observed a condensate demineralizer bead resin shipment (no. 03-145), that was packaged and prepared for shipment, surveyed, quality control inspected, and shipped offsite. These activities were reviewed with respect to PPL procedures, 10 CFR Parts 61, 71 and 49 CFR Parts 170-189 requirements.

The inspector reviewed the following six radioactive shipment records for compliance with radwaste shipping procedure, WM-PS-100, revision 7, “Shipment of Radioactive Waste”; and federal regulations in 10 CFR Parts 20, 61, and 71 and 49 CFR Parts 170-189.

- Shipment no. 03-47, contaminated turbine components, shipped March 14, 2003
- Shipment no. 03111, reactor water cleanup filter media, shipped May 8, 2003
- Shipment no. 03-117, dry active waste, shipped May 22, 2003
- Shipment no. 03-126, liquid radwaste filter media, shipped June 19, 2003
- Shipment no. 03-128, spent fuel scrapings, shipped June 12, 2003
- Shipment no. 03-145, bead resins, shipped August 6, 2003

The inspector reviewed PPL’s oversight of the radwaste transportation program. The inspector reviewed Quality Assurance audit of the radioactive material shipping program conducted in December 2001. The criteria used for this review was the audit requirements specified in 10 CFR 71.137 and 10 CFR20.1101(c).

Liquid and solid radwaste processing plant equipment spaces were walked down and reviewed with respect to radwaste processing design and abandoned radwaste processing equipment descriptions in the Updated Final Safety Analysis Report (UFSAR) Sections 11.2 and 11.4 and the Process Control Program (PCP). Any radwaste processing changes since the previous inspection in this area were reviewed with respect to 10 CFR50.59 evaluations. During the solid radwaste processing system walkdown, the processes for transferring radwastes into shipping containers were reviewed to ensure appropriate sampling and waste characterization of radwaste shipments.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed PPL's performance indicator (PI) data to verify whether the PI data was accurate and complete. The inspectors compared the PI data against the guidance contained in NEI 99-02. This inspection activity represented four samples. The following NRC PIs and PPL documents were included in this review:

Procedures and Documents

- Nuclear Energy Institute (NEI) 99-02, revision 2, "Regulatory Assessment Performance Indicator Guideline"
- NDAP-QA-0737, "Regulatory Performance Assessment"
- LI-00-018, "Preparation of Performance Indicator Data, NRC Submittals, and Cornerstone Assessment Reports"

Mitigating Systems Cornerstone PIs

- Unit 1 & 2 Reactor Core Isolation Cooling System Unavailability
- Unit 1 and 2 High Pressure Coolant Injection System Unavailability

For the previous 4 quarters, the inspectors examined selected samples of PI data, PI data summary reports, cornerstone assessment reports, and plant records, which included selected control room narrative logs, and condition reports. In addition, the inspectors interviewed the responsible system engineers.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Routine PI&R Review

a. Inspection Scope

The inspectors reviewed selected condition reports (CRs), as part of the routine baseline inspection documented in this report. The CRs were assessed to verify whether the full extent of the various issues were adequately identified, appropriate evaluations were performed, and reasonable corrective actions were identified. The inspectors evaluated the CRs against the requirements of NDAP-QA-0702, "Action Request and Condition Report Process," and 10 CFR 50, Appendix B.

In addition, the inspector reviewed twenty-three CRs relating to the processing and shipping of radioactive material between November 2001 and July 2003 to evaluate PPL's threshold for identifying and resolving problems in implementing the radioactive material transportation program. The condition reports were evaluated against the criteria contained in the PCP, 10 CFR Parts 20, 61, and 71 and 49 CFR Parts 170-189.

b. Findings

One licensee identified violation (CR 483279) of very low safety significance was identified and is described in Section 4OA7 of this report.

2. Annual Sample Review - "D" EDG Unplanned Shutdown

a. Inspection Scope

The inspectors reviewed PPL's root cause analysis and associated corrective actions for an unplanned shutdown of the "D" emergency diesel generator (EDG) on March 19, 2003. During the monthly surveillance test, the Woodward governor positioner arm became disconnected from the fuel supply rack when the connecting bolt fell out. After the bolt was re-installed, the inspectors observed portions of the post maintenance surveillance test and also observed the vibration level on the governor to fuel rack linkage connection. The inspectors also reviewed the initial operability assessment and common cause failure determination for the other EDGs. This inspection activity represented one sample. The following documents were included in the review:

Procedures and Documents

- Maintenance Rule Bases Document for emergency diesel generators
- Operability Assessment for condition report 460227
- MT-AD-509, "Control of Minor Maintenance Activities"
- Condition Reports 460227, 498084, 498436, and 504149
- Work orders 265805, 460312, 460811, and 460834
- MT-GM-015, "Torquing Guidelines"

b. Finding and Observations

Finding 1 - "D" EDG Bolt Failure - Cause Not Determined Prior to Return to Service

Introduction

The inspectors identified a non-cited violation of 10 CFR 50 Appendix B Criterion XVI of very low safety significance (Green). On March 19, 2003, PPL returned the "D" EDG to an operable status without adequately determining the cause of a linkage connecting bolt to fall off. The bolt connects the governor positioner arm to the fuel supply rack. PPL's initial repair was not sufficient to prevent repetition because although the bolt was initially reinstalled, it was not tightened to the required torque value of 25-30 foot pounds as required by the diesel vendor and plant procedures. On March 21, 2003, PPL removed the EDG from service and tightened the linkage connecting bolt to the required torque value.

Description

On March 19, 2003, during the monthly surveillance test, the Woodward governor positioner arm became disconnected from the fuel supply rack when the linkage connecting bolt fell out. As a result the governor was unable to position the fuel rack to control the electrical load on the EDG. PPL staff made an attempt to reconnect the governor positioner arm to the fuel rack while the EDG was still running. This induced significant electrical load changes on the EDG and the PPL staff manually tripped the EDG. PPL re-installed the bolt, wrench tight, restarted the EDG, and then completed the surveillance test to return the EDG to an operable status on the same day.

PPL initiated a Level 3 cause evaluation to determine the most probable cause. In addition, PPL performed an operability assessment to evaluate the as-left EDG condition. PPL's operability assessment concluded that "vibration of a loose connection" was the probable cause for the linkage connecting bolt falling out.

PPL's operability assessment did not identify that the cause of the "loose connection" was improper installation of the linkage connecting bolt during maintenance activities performed on July 5, 2000. Specifically, the bolt was not tightened to the proper torque values per PPL maintenance instructions in accordance with EDG vendor requirements.

PPL's reinstallation method of the bolt on March 19, 2003, did not tighten the bolt to a specific torque value. After the inspectors interviewed the maintenance supervisor, PPL recognized that it had not correctly re-installed the bolt and that the bolt was required to be tightened to a torque value per vendor requirements and PPL procedures. Cooper-Bessemer Engineering Standard SD-123 required the linkage connecting bolt to be tightened to a torque value of 25-30 foot pounds.

On March 21, 2003, PPL removed the "D" EDG from service, loosened the linkage connecting bolt, and tightened the bolt to a torque value of 25-30 foot pounds as required. For the "D" EDG, PPL did not check the as-found torque value of the bolt. For

the other four EDGs, PPL checked the as-found torque values of the linkage connecting bolts and verified they were at least 25 foot pounds.

Analysis

This finding is a performance deficiency because PPL returned the “D” EDG to service without properly tightening the linkage connecting bolt between the governor and the fuel supply rack linkage. Traditional enforcement does not apply because the issue did not have any actual safety consequence, or potential for impacting the NRC’s regulatory function, and is not the result of any willful violation of NRC requirements. This finding is greater than minor because it adversely impacts the equipment performance attribute of the mitigating system cornerstone and adversely affects the cornerstone objective in that the finding is associated with the reliability of the “D” EDG to respond to initiating events and prevent core damage.

This finding is of very low safety significance (Green) using phase one of the significance determination process. The issue does not result in an actual loss of safety function of a system or the loss of safety function of a single train for greater than the Technical Specification allowed outage time of 72 hours. In addition, the finding is not risk significant due to seismic, fire, flooding, or severe weather initiating events.

This finding is related to the Problem Identification and Resolution (PI&R) cross-cutting area because PPL did not identify the cause of a significant condition adverse to quality and take corrective actions to prevent recurrence.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” requires that nonconformances are promptly identified and corrected and for significant conditions adverse to quality, the cause of the condition be determined and corrective actions be taken to prevent repetition. Contrary to this requirement, on March 19, 2003, PPL returned the “D” EDG to an operable status without adequately determining the cause of a linkage connecting bolt to fall off. The bolt connects the governor positioner arm to the fuel supply rack. PPL’s initial repair was not sufficient to prevent repetition because the bolt was not tightened to the required torque value. Cooper-Bessemer Engineering Standard SD-123 requires the linkage connecting bolt to be tightened to a torque value of 25 - 30 foot pounds. On March 21, 2003, PPL removed the EDG from service and tightened the linkage connecting bolt to the required torque value. Because this violation is of very low safety significance and PPL entered this finding into their corrective action program (CRs 460227and 498084), this violation is being treated as an non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000387/2003004-02 and 05000388/2003004-02, “D” EDG Bolt Failure - Cause Not Determined Prior to Return to Service.**

Finding 2 - Maintenance Work Instructions Not Implemented to Tighten a "D" Emergency Diesel Generator Governor Bolt.

Introduction

The inspectors identified an unresolved item related to Technical Specification 5.4.1. On July 5, 2000, PPL did not implement written work instructions to tighten a linkage connecting bolt on the "D" emergency diesel generator (EDG) between the governor and the fuel supply rack linkage. As a result, on March 19, 2003, after approximately 205 diesel run hours, the bolt fell out and disconnected the governor from the fuel rack during an EDG monthly surveillance run.

Description

On March 19, 2003, the Woodward governor positioner arm became disconnected from the fuel supply rack when the linkage connecting bolt fell out. PPL identified that the governor had been installed by work order (WO) 265805, "D EDG Replace Main Governor," on July 5, 2000. PPL determined that no subsequent work had been performed that would have removed or loosened the bolt which fell out. Work order 265805 instruction step 6.11.1, required maintenance technicians to "tighten the associated fasteners and torque per Cooper-Bessemer engineering standard SD-123." Cooper-Bessemer engineering standard SD-123 required the bolt to be tightened to a torque value of 25-30 foot pounds.

The inspectors determined that work order 265805 written remarks and attached torque data sheet had no data to indicate that the linkage connecting bolt had been tightened to a torque value. The data indicated that only the governor mounting bolts had been tightened to a torque value. PPL's generic maintenance instruction MT-GM-015, "Torquing Guidelines," required the technicians to document the actual torque applied and torque wrench used for any bolt that was tightened to a torque value. Based on review of the records, the inspectors concluded that the linkage connecting bolt had not been tightened to the required torque value when the governor was installed in July 2000. PPL has revised the governor work instructions to include a specific step to tighten the governor linkage bolt to the vendor required torque value.

On March 19, 2003, after approximately 205 diesel run hours, the bolt fell out and disconnected the governor positioner arm from the fuel rack during an EDG monthly surveillance run. On March 21, 2003, PPL removed the "D" EDG from service, loosened the linkage connecting bolt, and tightened the bolt to a torque value of 25-30 foot pounds as required. For the "D" EDG, PPL did not check the as-found torque value of the bolt. For the other four EDGs, PPL checked the as-found torque values of the linkage connecting bolts and verified they were at least 25 foot pounds.

Analysis

This finding is a performance deficiency because PPL did not adequately implement written maintenance instructions during replacement of the EDG governor. This finding is greater than minor because it relates to the equipment performance attribute of the mitigating systems cornerstone and adversely affects the cornerstone objective, in that, the "D" EDG was incapable of performing its safety function for a period of time in

excess of its Technical Specification allowed outage time. This finding is an unresolved item pending completion of a Significance Determination Process for At-Power Situations.

A contributing cause of the finding is related to the Human Performance cross-cutting area because maintenance technicians did not adequately implement written work instructions.

This finding is also related to the Problem Identification and Resolution cross-cutting area because PPL did not identify this performance deficiency during their corrective action cause review for the "D" EDG in-service failure of March 19, 2003.

Enforcement

Technical Specification 5.4.1 requires, in part, that written procedures shall be established and implemented as recommended in NRC Regulatory Guide (RG) 1.33 Appendix A. RG 1.33 Appendix A, section 9.a, "Procedures for Performing Maintenance," required pre-planned maintenance activities be performed in accordance with written procedures for maintenance that can affect the performance of safety related equipment. PPL work order 265805, "D EDG Replace Main Governor," instruction step 6.11.1, required maintenance technicians to "tighten the associated fasteners and torque per Cooper-Bessemer engineering standard SD-123." Contrary to the above, September 9, 2003, the NRC identified that on July 5, 2000, PPL did not implement work order 265805 instruction step 6.11.1, to "tighten the associated fasteners and torque per Cooper-Bessemer engineering standard SD-123," for the bolt that connected the governor to the fuel supply rack linkage. This finding does not present an immediate safety concern because, as a follow-up corrective action, PPL verified that the linkage connecting bolts on all 5 EDGs were tightened to acceptable torque values. Pending determination of the findings safety significance, this finding is identified as an unresolved item (URI): **URI 05000387/2003004-03 and 05000388/2003004-03, Maintenance Work Instructions Not Implemented to Tighten a "D" Emergency Diesel Generator Governor Bolt.**

Observations

PPL did not enter the missed torque requirement into their corrective action program until questioned by the inspectors on August 19, 2003, after PPL completed their Level 3 cause evaluation and the inspectors' subsequent review of PPL’s apparent cause evaluation.

PPL’s actions to prevent recurrence were narrowly focused and limited in scope. The linkage connecting bolt threaded into the governor arm and did not have enough space to install a locking nut. The EDG vendor guidance suggested that a thread locking compound be used in this application. As of August 20, 2003, PPL had not applied thread locking compound to any of the five EDG linkage connecting bolts. PPL initiated CR 498436 to evaluate this issue.

The inspectors' review of PPL's Level 3 cause evaluation also identified that the initial condition report classification level was lower than the recommended classification level in OESI-AD-001, "Action Request Process-Subtype Condition Report and Management Processing." PPL initiated CR 504149 to evaluate this issue. PPL's corrective action program provides the highest level of review for a Level 1 issue, followed by a Level 2, and then a Level 3, which would receive the least management attention or review. The "D" EDG bolt issue matched a number of examples that would have resulted in a Level 1 root cause determination. Examples of issues that required a Level 1 review included:

- The issue was a significant condition adverse to quality
- Condition if left uncorrected, would affect plant safety, reliability, or public safety

PPL's performed a common cause failure assessment after the "D" EDG became inoperable on March 19, 2003. The inspectors concluded that additional information could have been reviewed and documented to support PPL's position that the operable EDGs were not affected by the conditions that led to the inoperable "D" EDG. PPL determined that the most probable cause was "vibration of a loose connection." PPL examined the governor to fuel supply mechanical linkage for the other four EDGs, verified no loose connections existed, and concluded that a common cause failure did not exist. PPL did not specifically verify that the governor bolts were tightened to a torque value of 25-30 foot pounds, as required by Cooper-Bessemer Engineering Standard SD -123 until prompted by the inspector. The five EDG vibration levels were not reviewed initially or documented in PPL's common cause failure determination. PPL did not assess whether the common maintenance work instruction used to remove and install the linkage connecting bolt on the diesels was adequate.

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

Section 4OA2 Finding 1, describes a finding where PPL did not identify the cause of a significant condition adverse to quality associated with the "D" emergency diesel generator (EDG) and take corrective actions to prevent recurrence. The "D" EDG was returned to service with an inadequately tightened linkage connecting bolt on the "D" emergency diesel generator (EDG) between the governor and the fuel supply rack.

Section 1R14 describes a finding because PPL did not take meaningful corrective actions for a September 1999 trip of a reactor feed pump while performing the same emergency governor trip test. The same probable cause for the 1999 reactor feed pump trip, self/peer checking standards not applied, was the same probable cause of the 2003 event.

4OA3 Event Follow-up (71153)

1. (Closed) LER 05000387/2003003-00 Both Trains of Standby Gas Treatment Inoperable due to Inadequate Maintenance and Inadequate Operability Testing

On November 19, 2002, maintenance was performed on a standby gas treatment system (SGTS) damper and the damper was returned to an operable status without

performing an adequate post maintenance or operational test. Four months later, on April 16, 2003, PPL discovered that the damper could not perform its safety function to modulate the air flow from the secondary containment into the SGTS, to control reactor building pressure while SGTS is aligned to the secondary containment. As a result, the "B" train of SGTS was unavailable for 4 months, and both the "A" and "B" trains of SGTS were unavailable on four separate occasions (e.g., a safety system functional failure), when the "A" train of SGTS was removed from service for scheduled maintenance, during February 10 to 12, February 24 to 25, April 11, and April 12, 2003. PPL corrected the condition and restored the damper to an operable condition on April 16, 2003.

This self-revealing violation was documented and discussed in detail in NRC Inspection Report 50-387,388/2003-003, in section 1R19.2. The inspectors reviewed PPL's apparent cause evaluation and corrective actions taken and planned, to verify whether they appeared reasonable. No new issues were identified in the inspectors' review of this LER. This finding was documented in PPL's corrective action program as condition report 467829. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

Cross-References to Human Performance Findings Documented Elsewhere

Section 1R14 describes a finding where a control room operator did not follow a reactor feed pump test procedure. As a result, the "C" reactor feed pump tripped and the reactor automatically shutdown due to low reactor water level.

Section 4OA2 describes a finding where maintenance technicians did not adequately implement written work instructions. As a result, the EDG was returned to service in a degraded condition and subsequently became unable to perform its safety function.

4OA6 Meetings, Including Exit

On October 17, 2003, the resident inspectors presented the inspection results to Mr. R. Anderson and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-identified Violations

The following violation of very low safety significance (Green) was identified by PPL and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation.

- 10 CFR71.5 requires NRC licensee's to comply with DOT regulations 49CFR170-189. 49CFR172.504(e), Table 1, requires RADIOACTIVE placards on transport vehicles for Radioactive Yellow III label package shipments. On June 12, 2003, a 4 gallon package containing spent fuel scrapings, classified

and labeled as Radioactive Yellow III, was delivered to the Wilkes-Barre/Scranton airport by private automobile without RADIOACTIVE placards. This event is documented in PPL's corrective action program as condition report 483279. This finding is of very low safety significance because the potential radiation exposure to the public was very low (less than 1 mrem/hr for a 40 minute duration), package integrity was not lost, and no radiation limit was exceeded.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINT OF CONTACT

PPL Personnel

M. Micca, Radioactive Materials Shipper
R. Smith, Radiation Protection Manager
R. Steiger, Radwaste Management Specialist
E. Banks, Effluent Management Supervisor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000387,388/2003004-03	URI	Maintenance Work Instructions Not Implemented to Tighten a "D" EDG Bolt

Opened and Closed

05000387/2003004-01	NCV	Did Not Properly Implement Feed Pump Test and Caused Feed Pump Trip.
05000387,388/2003004-02	NCV	“D” EDG Bolt Failure - Cause Not Determined Prior to Return to Service.

Closed

05000387/2003003-00	LER	Both Trains of Standby Gas Treatment Inoperable due to Inadequate Maintenance and Inadequate Operability Testing (Section 4OA3.1)

Discussed

None

LIST OF DOCUMENTS REVIEWED

(Not Referenced in the Report)

LIST OF ACRONYMS

CFR	Code of Federal Regulations
CR	Condition Report
EAL	Emergency Action Level
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
FSAR	[SSES] Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
LER	Licensee Event Report
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PCO	Plant Control Operator
PCP	Process Control Program
PCS	Power Conversion System
PI	[NRC] Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post Maintenance Test
PPL	PPL Susquehanna, LLC
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RFP	Reactor Feed Pump
RFPT	Reactor Feed Pump Turbine
RG	[NRC] Regulatory Guide
RHR	Residual Heat Removal
RWCU	Reactor Water Cleanup
SDP	Significant Determination Process
SGTS	Standby Gas Treatment System
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
TS	Technical Specifications
WO	Work Order