November 12, 2004

Mr. Bryce L. Shriver
President, PPL Generation, LLC and
Chief Nuclear Officer
PPL Generation, LLC
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000387/2004004 AND 05000388/2004004

Dear Mr. Shriver:

On September 30, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Susquehanna Steam Electric Station, Units 1 and 2. The enclosed integrated inspection report presents the results of that inspection, which was discussed with Mr. B. McKinney, Vice President - Nuclear Site Operations and other members of your staff on October 7, 2004.

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

This report documents three findings of very low safety significance (Green). One of the findings was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issue was entered into your corrective action program, the NRC is treating this finding as non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. Additionally, one licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. If you contest the NCV 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 U.S. 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, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of the NRC's document

system (ADAMS). ADAMS is accessible from the NRC Web site at
http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

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

Docket Nos. 50-387; 50-388, 72-28
License Nos. NPF-14, NPF-22

Enclosure: Inspection Report 05000387/2004004 and 05000388/2004004
Attachment: Supplemental Information

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REGION I

Docket Nos.	50-387, 50-388, 72-28
License Nos.	NPF-14, NPF-22
Report No.	05000387/2004004, 05000388/2004004
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	769 Salem Boulevard Berwick, PA 18603
Dates:	July 1, 2004 through September 30, 2004
Inspectors:	A. Blamey, Senior Resident Inspector J. Richmond, Resident Inspector F. Jaxheimer, Resident Inspector N. McNamara, EP Inspector J. Furia, Sr. Health Physicist T. Burns, Reactor Inspector J. Wray, Health Physicist
Approved by:	Mohamed M. Shanbaky, Chief Projects Branch 4 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000387/2004004, 05000388/2004004; 07/01/2004 - 09/30/2004; Susquehanna Steam Electric Station, Units 1 and 2; Maintenance Implementation, Post Maintenance Test, Identification and Resolution of Problems.

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

A. NRC-Identified Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding because PPL did not complete an evaluation of the condition of Unit 2 transformer 2X270 as required by a station procedure before energizing the transformer. Not completing the evaluation allowed a degraded transformer to be returned to service. The transformer faulted shortly after being placed in service which resulted in a loss of main condenser vacuum.

  This finding is greater than minor because it adversely impacts the equipment performance attribute of the Initiating Events cornerstone and the finding adversely affected the cornerstone objective, in that, it is associated with an event that upset plant stability. This finding was considered to have very low safety significance (Green), using phase 1 of the significance determination process. The failure of transformer 2X270 did not increase the likelihood of an LOCA initiator, and did not increase the likelihood of a reactor trip and the likelihood that mitigation functions would be lost. In addition, the finding did not increase the likelihood of a fire or flood event.

  A contributing cause of this finding is related to the Human Performance cross-cutting area because PPL did not complete the required retest and engineering evaluation of transformer 2X270 prior to energizing the transformer.
  (Section 1R19)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10CFR 50.65 paragraph (b)(2) of the Maintenance Rule, because PPL did not scope the Unit 1 and Unit 2 reactor building (RB) equipment and floor drain systems (EFDS) into the Maintenance Rule program and as a result did not demonstrate the effectiveness of preventive maintenance for the RB EFDS. The inclusion of the RB EFDS in the scope of the monitoring program was necessary because the RB EFDS are relied upon to mitigate internal flooding events. Failure of the EFDS to function could have prevented safety-related structures, systems and components from fulfilling their safety-related function.

This finding was more than minor because it had greater significance than similar issues described in the NRC Inspection Manual Chapter 0612, "Examples of Minor Issues," Section 1.h and 1.i. In addition, the RB EFDS's performance is associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. On August 18, 2004, the Unit 1 RB EFDS was unable to pass 80 gpm as assumed in the Final Safety Analysis Report during an overflow of the reactor water cleanup backwash receiving tank. Inspectors identified that system performance problems were such that a Maintenance Rule (a)(2) demonstration could not be justified. This finding was considered to have very low safety significance because the finding did not contribute to an actual loss of mitigation equipment functions, and did not increase the likelihood of a fire or flooding event.

A contributing cause of this finding was related to Problem Identification and Resolution cross-cutting area. PPL had eleven previous EFDS blockages and the evaluation of those events did not recognize that portions of the non-safety related EFDS were relied upon to mitigate accidents or transients. Therefore, PPL did not monitor the EFDS under the maintenance rule and this contributed to the degradation of the RB EFDS. (Section 1R12).

C Green. A finding of low safety significance was identified because PPL did not adequately evaluate and correct a degraded condition associated with the high engine operating temperatures and repetitive overheating of the diesel driven fire pump (DFP) which occurred following engine shutdown.

This issue is greater than minor because it affected the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. This finding is of very low safety significance, based on a Phase 1 significance determination process evaluation, because the finding did not result in the loss of a function of equipment designed as risk significant for greater than 24 hours and the finding does not increase the potential or risk of a seismic event , flood or severe weather event.

A contributing cause of this finding is related to the Problem Identification and Resolution (PI&R) cross-cutting area. PPL did not sufficiently evaluate the condition to identify and correct the reduced cooling water flow to the DFP engine. This resulted in ineffective corrective actions because the DFP was removed from service several times without taking action to correct the DFP high engine coolant temperature issue. (Section 4OA2.3)

B. Licensee Identified Violation

A violation of very low safety significance, which was identified by PPL, has been reviewed by the inspectors. Corrective actions taken or planned by PPL have been entered into PPL's corrective action program. This violation 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 August 22, 2004, reactor power was reduced to 70 percent power due to a problem with the "B" reactor feedwater pump turbine control valve operator linkage. This problem with the linkage was corrected and Unit 1 was returned to 100 percent power on August 25, 2004. On September 10, 2004, reactor power was reduced to 70 percent power for planned maintenance on feedwater heaters. Maintenance activities were completed and Unit 1 was returned to 100 percent power on September 13, 2004. Unit 1 was operated at or near full power for the remainder of the inspection period.

Unit 2 was operating at or near full power at the beginning of the inspection period. On July 29, 2004, reactor power was rapidly reduced to 78 percent power due to the loss of a 13.8 KV stepdown transformer 2X270. The power loss impacted condenser air removal capability reducing condenser vacuum. Alternate equipment was put in service and Unit 2 returned to 100 percent power on July 30, 2004. Unit 2 reactor power was reduced to 70 percent on September 18, 2004 for a control rod sequence exchange and returned to full power on September 19, 2004. Unit 2 was operated at or near full power for the remainder of the inspection period, with exceptions for brief power reductions to support control rod pattern adjustments or to support transmission and distribution limitations (minimum generation alerts).

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather (71111.01- 3 Samples)

a. Inspection Scope

The inspectors reviewed PPL's preparations for adverse 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 hot weather and storm preparations. The inspectors reviewed the procedures for hot weather and high wind protection of the associated systems. This inspection activity represented three samples. The areas, components, and documents reviewed included:

Structures, Systems, and Components

- Onsite / Offsite electrical power system, hurricane Charley, August 13, 2004
- Onsite / Offsite electrical power system, hurricane Ivan, September 15, 2004
- Closed cooling water systems, August 13, 2004, hot weather preparation

Procedures and Documents

- NDAP-00-0024, “Winter Operation Preparations and Severe Weather Operation”
- ON-000-002, “Natural Phenomena”
- ON-000-005, “Hot Weather”
- CR 549002, Unit 2 ISO Phase Flux Plates Continue to be Hot (422°F)

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04Q - 6 samples)

1. Partial System Walkdowns

a. Inspection Scope

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

- Fire water system with alternate suction alignment and diesel fire pump troubleshooting in progress
- Unit 2 division II RHR with division I RHR in SOW, August 11, 2004
- Unit 1 division I core spray (following the RWCU overflow event - section 1R06)
- Unit 2 turbine building closed cooling water (TBCCW) during and post maintenance on service water pressure safety valve (PSV) to B cooler, September 7, 2004
- Unit 1, 4 KV emergency bus alignment during 4 KV emergency bus 1C feeder breaker replacement, September 7, 2004
- ESW following flow balance testing, September 16, 2004

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 11 Samples)

1. Routine Plant Area Observations

a. Inspection Scope

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

- Unit 2 reactor core isolation cooling (RCIC) room, FP-213-239
- Unit 2 high pressure coolant injection (HPCI) room, FP-213-238
- Reactor building 719' elevation, 3-hour fire barriers between Units 1 and 2, CR 582588
- Unit 1 Reactor Building 670' elevation, Fire Zones 1-2B and 1-2D
- Unit 1 Reactor Building 683' elevation, Fire Zone 1-3A
- Control Structure Fire Zones 0-26M and 0-26R, CO2 Manual Spurt Suppression
- Unit 2 Reactor Building 683' elevation, Fire Zones 2-3B-W and 2-3C-W
- Control Room, Tech Support Center and adjacent areas, FP-013-156
- Unit 1 battery rooms and DC distribution panel areas, Fire Zones 0-28j, k, l & m.
- Unit 1 & 2 TBCCW pump and heat exchanger area, Fire Zone 0-21A
- Unit 1 Switchgear rooms, 719' elevation, FP-113-115

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 2 Samples)

1. Internal Flood Protection

a. Inspection Scope

On August 18, 2004, during an extended backwash evolution on the reactor water cleanup demineralizer at Susquehanna Unit 1, the backwash receiving tank overflowed and placed approximately 1500 gallons of contaminated water into the reactor building equipment floor drain system. The drain header became blocked due to the resin from the receiving tank and rust that was displaced from the inside of the drain piping. The water flowed up and out of the blocked drains on a lower elevation, across the floor and down into the division II core spray and the high pressure coolant injection system compartments. The water entered these compartments by flowing though gaps between the equipment hatch floor plugs and the floor that were not sealed. The division II core spray and high pressure injection system compartments both had approximately 2 inches of water on the floor.

The inspectors reviewed the event and its impact on safety related equipment. The inspectors compared the plant response to the expected plant response based on the flood protection design features specified in the Final Safety Analysis Report (FSAR), engineering analysis. The inspectors conducted Unit 1 walkdowns to independently assess the leakage paths, water accumulation, and the operability of equipment that was in the affected areas of the spill. This inspection activity represented one sample. Documents reviewed during the inspection are listed in the attachment, "List of Documents Reviewed."

b. Findings

Large equipment hatches are installed in the ceilings of all the emergency core cooling systems (ECCS) rooms to allow for removal of equipment during maintenance activities. The equipment hatches are normally closed by the use of large equipment hatch plugs which are placed in these openings. PPL reviewed the design of the equipment hatch plugs after the August 18, 2004, event and determined that the plugs would not provide a water tight seal between the plug and the surrounding floor, even though this was specified in the FSAR, Section 3.4. In addition, the plant specific analysis did not consider any leakage through the equipment hatch plugs. The analysis accounted for leakage around doors and through the floor drain system but not through the equipment hatch plugs. Therefore, the analysis implied that the equipment hatch plugs were water tight. PPL is currently reanalyzing the flooding issue to determine if the equipment hatch floor plugs must be water tight. This analysis is being performed under condition report 600070. This issue is unresolved pending PPL's completion of the flooding analysis and the NRC review of this analysis. This finding is identified as **URI 05000387/2004004-01, "Equipment Hatch Floor Plugs are not Watertight as Indicated in the FSAR."**

2. External Flood Protection Measures

a. Inspection Scope

The inspectors reviewed PPL's external flood analysis, flood mitigation procedures, and design features to verify whether they were consistent with the PPL design requirements. The inspectors walked down selected risk significant plant areas, including the moats and surrounding areas for large on-site tanks. The inspectors evaluated the condition and adequacy of flood detectors, sump pumps, sump level alarm circuits, and other flood protection design features to assess whether the flood protection design features were adequate and operable. During the walk downs, the inspectors also evaluated whether there were any unidentified or unanalyzed sources of flooding, including holes and unsealed penetrations in floors and walls. This inspection activity represented one sample. The specific areas included:

- Turbine and reactor building protection from circulating water flume rupture
- “A” through “D” emergency diesel generator rooms
- “E” emergency diesel generator building

The inspectors reviewed PPL's flood mitigation procedures, flood alarm response procedures, and selected preventive maintenance tasks for flood detectors and flood barriers to evaluate whether component functionality was routinely verified. In addition, the inspectors reviewed PPL's corrective action program, including system health reports, and interviewed selected maintenance personnel to verify whether previous flood related issues had been appropriately identified, evaluated, and resolved. The following procedures were included in the review:

- NE-94-001, Section 5.2, “Susquehanna IPE for External Events - Floods”
- FSAR Section 2.4.2, “Hydrologic Engineering - Floods”
- FSAR Section 3.4, “Water Level (Flood) Design”
- EC-RISK01024, “External Flood Effects”
- CR# 598927, Turbine Building Overhead Door Opening

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07 - 2 Samples)

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 “A” residual heat removal (RHR) room cooler (2E230A) and the Unit 2 “A” high pressure coolant injection (HPCI) room cooler (2E229A). 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. This inspection activity represented two samples. The inspectors' review included the following documents:

- WO 541964, Unit 2 Division I RHR Room Cooler clean and inspect
- WO 541922, Unit 1 Division I HPCI Room Cooler clean and inspect
- EC-CHEM-1018, Justification for the Assurance of Adequate Heat Removal Capabilities Using the Susquehanna Heat Exchanger Preventive Maintenance Program

b. Findings

No findings of significance were identified.

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

1. Routine Licensed Operator Requalification

a. Inspection Scope

On August 25, 2004, the inspectors observed a simulator sessions of operator requalification training. The inspectors compared the actions taken during the simulator scenario to classroom objectives, and compliance with Technical Specifications, NRC orders, and 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 deficiencies in operator training. This inspection activity represented one sample. The observed training scenario included Security Events, Station Procedure ON-000-010.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12Q - 5 Samples)

1. Routine Review of Maintenance Implementation

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 four samples. The following issues were reviewed:

Equipment Issues

C Loss of Remote Position Indication for SV-22651

- C Loss of Remote Position Indication for HV-15725
- C Loss of Remote Position Indication for HV-22603
- C Emergency lighting system 07 return to A2 status and functional failure of lighting battery 20135
- C Reactor Building Floor and Equipment Drains

Procedures and Documents

- C NDAP-QA-0413, “Maintenance Rule Program”
- C Unexpected Loss of DC Power to the Inboard MSIV and SV-22651, (CR 597331)
- C Loss of Open Indication for Containment Isolation Valve HV-22603, (CR 597822)
- C Primary Containment Instrument Gas Maintenance Rule Basis Document
- C Emergency Lighting Maintenance Rule Basis Document

b. Findings

No findings of significance were identified.

2. Equipment Floor Drain System

a. Inspection Scope.

The inspectors evaluated PPL’s work practices and preventive maintenance activities for the reactor building (RB) equipment and floor drain systems (EFDS) to assess the effectiveness of PPL’s maintenance activities. The inspectors reviewed the performance history of the RB EFDS to assess PPL’s problem identification and to evaluate whether PPL had appropriately monitored, evaluated and dispositioned issues in accordance with PPL procedures and the requirements of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance.” The inspectors reviewed the associated system design basis, including the Final Safety Analysis Report (FSAR) and the RB internal flood design calculations, to assess the adequacy of PPL’s actions. In addition, the inspectors performed field walkdowns and interviewed PPL staff to verify whether the identified actions were appropriate and to verify that known performance problems were included in the corrective action process. This inspection activity represented one sample.

b. Findings

Introduction. A Green non-cited violation (NCV) of 10 CFR 50.65 (b)(2) was identified because PPL did not scope an accident and transient mitigation function of the EFDS into the Maintenance Rule monitoring program. The inspectors concluded that not having the EFDS function scoped into the monitoring program allowed the deterioration of system performance such that the Unit 1 RB EFDS could not perform the intended design function on August 18, 2004 (Section 1R14).

Description. The NRC identified that PPL did not correctly scope the RB EFDS into the maintenance rule program. The Maintenance Rule, 10 CFR 50.65 paragraph (b)(2), requires that systems whose failure to function as designed could prevent safety-related structures, systems and components from fulfilling their safety-related functions should be scoped into the maintenance rule program. Since the RB EFDS was not scoped into the maintenance monitoring program, PPL did not establish performance criteria to demonstrate the effectiveness of the RB EFDS maintenance. Therefore, PPL did not identify less than adequate preventive maintenance and the system failed to meet design requirements on August 18, 2004, when the reactor water cleanup (RWCU) backwash receiving tank overflowed and the EFDS became blocked.

The RB EFDS are required to mitigate internal flooding events such as moderate and high energy line breaks; as well as, fire deluge system actuations to prevent this water from impacting safety-related equipment. The Susquehanna FSAR, Section 3.4, "Water Level (Flood) Design," states the capacity of a single floor drain is approximately 80 gpm. The PPL design calculation EC-FLOD-0500, assumed 80 gpm per floor drain and a maximum of 200 gpm for each reactor building drain header. During the August 18, 2004, event one RB EFDS header became blocked which prevented the required flow assumed in the FSAR and supporting analysis.

The inspectors also identified that PPL had previously identified occurrences of EFDS blockage. During the past three years there were eleven occurrences of backed up EFDS in the Unit 1 and Unit 2 reactor buildings. The evaluation of these occurrences did not recognize that portions of the non-safety related floor drain system are relied upon to mitigate accidents or transients and did not specify adequate corrective actions to prevent recurrence of the EFDS blockage. PPL had established and performed a routine task to unplug portions of the reactor building and turbine building EFDS. However, the preventive maintenance tasks were scheduled to ensure a functional drain system before the beginning of a refueling outage, not as a deliberate measure to maintain the systems functional during the plant operating cycle. Therefore, the inspectors determined that the maintenance history and associated system performance problems indicated that PPL had missed opportunities to place these components in the maintenance rule scope and as a result did not effectively maintain the function of the EFDS through appropriate preventive maintenance.

Analysis. The finding was a performance deficiency because PPL did not scope sections of the system into the Maintenance Rule Program and consequently did not provide an adequate Maintenance Rule (a)(2) demonstration for the RB EFDS. 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 was not the result of any willful violation of NRC requirements. The finding was more than minor because it had greater significance than similar issues described in the NRC Manual Chapter 0612, "Examples of Minor Issues", Section 1.h and 1.i. In addition, the RB EFDS's performance is associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objectivity to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences because on August 18, 2004, the Unit 1 RB EFDS was unable to pass 80 gpm as assumed in the Final Safety Analysis Report during an overflow of the reactor water cleanup backwash receiving tank.

This finding was considered to have very low safety significance (Green) using the NRC's Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power situations because the finding did not contribute to a loss of mitigation equipment functions and did not increase the likelihood of a fire or flooding event.

A contributing cause of this finding was related to Problem Identification and Resolution cross-cutting area. PPL had eleven previous EFDS blockages and the evaluation of those events did not recognize that portions of the non-safety related EFDS were relied upon to mitigate accidents or transients. Therefore, PPL did not monitor the EFDS under the maintenance rule and this contributed to the degradation of the RB EFDS.

Enforcement. 10CFR 50.65 (b)(2) requires, in part, that the scope of the monitoring program specified in paragraph (a)(1) include non-safety related structures, systems and components whose failure can prevent safety-related structures, systems and components from fulfilling their safety-related functions.

10 CFR (a)(2) states that "Monitoring as specified in paragraph (a)(1) of this section is not required where it has been demonstrated that the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance, such that the structure, system, or component remains capable of performing its intended function."

Contrary to the above, PPL did not include sections of the RB EFDS in the scope of the monitoring program as specified in 10 CFR 50.65(b)(2). The inclusion of the RB EFDS in the scope of the monitoring program was necessary because the system is utilized in the mitigation of internal flooding events. As a result of not scoping, a RB EFDS function into the monitoring program, PPL did not effectively control the performance or condition of the system through appropriate preventive maintenance, as required by 10CFR 50.65(a)(2). The RB Equipment and Floor drain system did not perform its intended function on August 18, 2004. System performance demonstrated that the system would not have functioned for more severe internal flooding events to protect safety-related equipment. Therefore, the inspectors concluded that as a result of this finding the RB EFDS system had not been effectively controlled through the performance of appropriate preventive maintenance and, as a result, a Maintenance Rule (a)(2) demonstration could not be justified.

Because this finding was of very low safety significance and it was entered into the PPL corrective action program, this finding is being treated as non-cited (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **NCV 05000387/2004004-02, "Reactor Building Floor and Equipment Drains Not Fully Scoped into the Maintenance Rule."**

1R13 Maintenance Risk Assessments & Emergent Work Evaluation (71111.13 - 5 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 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 Troubleshooting triple notch of control rod 18-23, TP-055-010
C Removal of division I ESW, draining and refill and venting, AR 592531
C RCIC remote shutdown panel switch replacement, RLWO 595679
C Loss of remote position indication for HV-22603, CR 597822
C 1A20301 breaker replacement (alternate power to 1C ES BUS), CR 603513

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14 - 2 Samples)

3. Unit 1 Reactor Water Cleanup Backwash Receiving Tank Overfill

a. Inspection Scope

The inspectors reviewed the station response to this August 18, 2004 overflow of approximately 1500 gallons of contaminated water from an extended backwash evolution on the Unit 1 reactor water cleanup demineralizer on August 18, 2004. This event resulted in unexpectedly placing contaminated water into the division II core spray and the HPCI system compartments. The inspectors reviewed the initial operator and radiation protection response to this event. The inspectors conducted Unit 1 walkdowns to independently assess the leakage paths, water accumulation, and additional water sources. This event is discussed further in Section 1R06, “Flood Protection Measures,” and Section 1R12, “Maintenance Implementation,” of this report. This inspection activity represented one sample.

b. Findings

No findings of significance were identified.

2. Unit 2 13.8 KV Stepdown Transformer (2X270) Electrical Fault

a. Inspection Scope

The inspectors reviewed the July 29, 2004 plant transient and operator response following the electrical fault on transformer 2X270. This fault resulted in isolation of the condenser offgas system, which reduced condenser vacuum and require the operators to reduce reactor power to 78 percent. Specifically, the inspectors reviewed the operator actions to stabilize the plant and the ascension to full reactor power. The inspectors compared the equipment and operator responses for the July 29 transient to the similar event that occurred on April 28, 2004. The inspectors reviewed and evaluated PPL’s April 28, 2004, event root cause analysis and the corrective actions that were taken to prevent recurrence. This event is discussed further in section 1R19, “Post Maintenance Testing,” of this report. This inspection activity represented one sample.

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, 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:

- Replacement of leaking ESW supply and return valves to “E” EDG, CR 544629, 592531
- Unit 2 control rod double notch and lockup, CR 591003, 591010, 591011
- CIG Header pressure below SO-100-006 required valve of 90 psig, CR 597117
- No Flow indicated during SLC pump surveillance SO-253-004, CR 599727
- Reactor building internal flood/physical separation (Floor Plugs), CR 600070
- Breaker 1A20309 failed to auto open on closure of 1A20301, CR 603513

b. Findings

No findings of significance were identified.

1R16 Operator Work-Around Cumulative Review (71111.16 - 3 Samples)

a. Inspection Scope

The inspectors reviewed the loss of the safety parameter display system (SPDS) and the loss of the system particulate, iodine, and nobel gas (SPING) terminals in the control room and the technical support center to determine how the effected system would impact the emergency response staff's assessment capability during an event. In addition, the inspectors reviewed significant control room deficiencies, status control tags, and selected corrective action reports to determine whether the functional capability of a system or staff emergency response actions would be affected. The inspectors evaluated the operators' ability to implement abnormal and emergency operating procedures during plant transients with the existing equipment deficiencies. The review included an evaluation of the cumulative and synergistic effects of the identified operator workarounds. This inspection activity represented two individual samples and one cumulative effects of operator workarounds. The following documents were included in the review:

Procedures and Documents

- Loss of the safety parameter display system, (CR 598870)
- Loss of the SPING terminal in the control room and TSC, (CR 603800)
- Control Room Deficiency list dated 8/1/04
- Operator Workaround list dated 8/30/04
- Operator Challenge list dated 9/13/04

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17 - 1 Sample)

a. Inspection Scope

The inspectors reviewed the system design package and the associated design and licensing documents for the Unit 1 Fire Protection Cross Tie to Condensate Transfer System Modification. All functions and design attributes of the modification that could affect the plant specific SDP worksheets were reviewed. Field implementation activities were observed and compared to the design requirements and installation standards. The inspectors reviewed the results of post modification testing. The inspectors also reviewed the affected procedures and design basis documents to verify that the affected documents were appropriately updated. Condensate transfer system and fire protection system were reviewed to verify the modification did not interfere or negatively impact system functions. This inspection activity represented one sample. The following documents were included as part of the review:

Procedures and Documents

- Design Change Package (DCP) and Modification Safety Assessment No. 556794, and 556789
- NDAP-QA-1220, Modification Process
- EO-000-102, “RPV control”
- EO-000-113, “Level Power Control”
- ON-037-001, “Loss of Condensate Transfer System”
- EO-000-114, “RPV Flooding”
- ES-013-001, “Fire Protection System Cross-tie to RHR”

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 8 Samples)

1. Routine Review of Post Maintenance Testing

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. This inspection activity represented seven samples. The post maintenance testing activities reviewed included:

- Unit 1 control rod stroke timing after rod 18-23 triple notch withdraw, TP-055-010
- Fire diesel replacement of fuel supply solenoids, CR 592724
- “E” EDG overhaul, TP-024-149 & MT-024-024
- Unit 2 division 2 RHR room cooler and “D” RHR motor maintenance, SO-249-B02
- Fire system to Unit 2 condensate transfer cross-tie
- Control room emergency air supply fan logic test after Agastat relay replacement WO 522184
- Unit 2 SLC flow indicator (FI) FI-24814 replacement AR 599727

b. Findings

No findings of significance were identified.

2. Unit 2 13.8 KV stepdown transformer 2X250, 2X270 power lead replacement

a. Inspection Scope

On April 28, 2004, a plant transient resulted from an electrical fault on the 13.8 KV power cables to transformer 2X270. This fault damaged the 2X270 transformer enclosure and resulted in an Unusual Event classification. The fault caused a short duration voltage reduction on the 13.8 KV system and de-energized two 480 volt load centers. This resulted in the isolation of the condenser offgas system and reduced condenser vacuum, which required the operators to reduce reactor power. PPL determined that the fault was the result of insulation breakdown on the 13.8 KV power cables to transformer 2X270. PPL performed maintenance testing to determine the health of the transformer. Inspectors observed portions of post maintenance testing activities in the field to determine whether the tests were performed in accordance with the approved work documents. The inspectors reviewed the completed work package and compared it to the work plan and procedures. The inspectors reviewed the recorded doble test data to determine whether the acceptance criteria were satisfied prior to placing the transformer back in service. This inspection activity represented one sample.

b. Findings

Introduction. A Green NRC-identified finding was identified because PPL did not complete an evaluation of transformer 2X270 test data as required by work package instructions and procedure MT-IT-001, “AC Insulation Dielectric Loss and Power Factor Check.” The inspectors concluded that not completing the evaluation allowed a degraded transformer (2X270) to be returned to service which resulted in a plant transient.

Description. The NRC identified that, PPL did not follow maintenance work instructions and station procedure MT-IT-001, “AC Insulation Dielectric Loss and Power Factor Check,” which required additional testing and the evaluation of the Unit 2 13.8 KV stepdown transformer 2X270 doble test data. Specifically, the work instructions and the initial engineering review determined that testing after the application of heat was required. Not completing the additional testing and, therefore, not completing the evaluation of the degraded transformer before energizing the transformer resulted in an electrical over current fault on the transformer’s primary windings on July 29, 2004. This fault resulted in the loss of power to various reactor building ventilation fans and dampers, chillers, a spent fuel pool cooling pump, and the trip and automatic start of main generator auxiliary system pumps. This fault also resulted in isolation of the condenser offgas system which reduced condenser vacuum and required the operators to reduce reactor power to 78 percent.

Following the April 2004 event, PPL initiated work requests to test load center transformers 2X250 and 2X270 to evaluate the operational readiness of these transformers prior to reenergizing the transformers. The test results showed a tripling of the percent power factor over the baseline data, which was an unacceptable change that required further evaluation. The work package test results recommended applying concentrated local heat to the transformer windings and to retest the transformers to determine if the unexpected results were due to moisture or other degradation in the transformer.

PPL applied concentrated heat to dry out any potential moisture in the transformer windings. However, PPL did not perform a retest or an engineering evaluation to assess the condition of the transformer, which was required by procedure MT-IT-001, "AC Insulation Dielectric Loss and Power Factor Check." Not performing these actions resulted in energizing a degraded transformer. On July 29, 2004, several hours after the transformer was energized the primary windings on the transformer faulted which resulted in a Unit 2 transient.

Analysis. The finding is a performance deficiency because PPL did not follow station procedure MT-IT-001, "AC Insulation Dielectric Loss and Power Factor Check," in that they did no perform an engineering evaluation of an unacceptable change in percent power factor. Not completing this assessment resulted in a plant transient after this degraded transformer was energized. This finding is greater than minor because it adversely impacted the equipment performance attribute of the initiating events cornerstone and adversely affected the cornerstone objective in that the finding was associated with an event that upset plant stability. 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 was considered to have very low safety significance (Green), using Phase 1 of the significance determination process. The issue did not result in an increase in the likelihood of a loss of coolant accident (LOCA) initiator; and did not increase the likelihood of a reactor trip and the likelihood that mitigation functions would be lost. In addition, the finding did not increase the likelihood of a fire or flood events.

A contributing cause of this finding is related to the Human Performance cross-cutting area because PPL did not complete the required retest and engineering evaluation of transformer 2X270 prior to energizing the transformer.

Enforcement. There were no violations of NRC requirements. The inspectors determined that the finding did not represent a noncompliance because procedure MT-IT-001, "AC Insulation Dielectric Loss and Power Factor Check," is not a procedure that is referenced among the required procedures listed in NRC regulatory Guide 1.33, Revision 2, February 1978, Appendix A. Additionally, the 13.8 KV electrical system and associated balance of plant 480 volt electric distribution system are not safety related. The related inspection issue was entered into the Susquehanna corrective action program as CR # 596092. **FIN 05000388/2004004-03, "PPL Did Not Retest and Evaluate Transformer 2X270."**

1R22 Surveillance Testing (71111.22 - 8 Samples)

a. Inspection Scope

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

- Unit 1 Control Rod Coupling Full Indicator Check, SO-156-007
- “E” Diesel Generator Monthly Operation Test, SO-024-014
- Unit 2 RCIC flow surveillance, SO-250-002
- Pump Curve Development for Division II ESW pumps TP-054-066
- “C” EDG Start Time Testing, OP-024-005 and OP-024-001
- Unit 1 HPCI Flow Surveillance, SO-152-006
- ESW Flow Balance, TP-054-076
- Unit 2 Shift and Daily Surveillance, SO-200-006

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modification (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 Final Safety Analysis Report (FSAR), Technical Specifications, and assessed the adequacy of the safety determination screenings and evaluations. The inspectors also assessed configuration control of the temporary 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. This inspection activity represented two samples. The following temporary modifications and documents were included in the review:

- Temporary Modification (TMOD) # 574659, Feedwater Heater 3A & 4A Level Control

- TMOD # 581717, Reactor Building Chiller Discharge Gas Temperature Trip Elimination

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes (IP 7111404)

a. Inspection Scope

A regional in-office review was conducted of licensee-submitted revisions to the emergency plan, implementing procedures and EALs which were received by the NRC during the period of April - September 2004. A thorough review was conducted of plan aspects related to the risk significant planning standards (RSPS), such as classifications, notifications and protective action recommendations. A cursory review was conducted for non-RSPS portions. These changes were reviewed against 10 CFR 50.47(b) and the requirements of Appendix E and they are subject to future inspections to ensure that the combination of these changes continue to meet NRC regulations. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 4, and the applicable requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 1 Sample)

a. Inspection Scope

On August 3, the inspectors observed a control room simulator based training event. The inspectors assessed licenced operator response to simulated plant events and their use of emergency plan procedures. The inspectors observed PPL's critique of the training event to evaluate PPL's identification of weaknesses and deficiencies associated with event classification and notifications. The inspectors compared PPL's identified findings against the inspectors' observations to determine whether PPL adequately identified performance issues. The inspection activity represented one sample. The inspectors' review included the following documents and procedures:

- Susquehanna Emergency Plan, revision 45
- EP-PS-100, "Emergency Director Control Room"

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 9 Samples)

a. Inspection Scope

The inspector reviewed and assessed the adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 mrem committed effective dose equivalent (CEDE).

The inspector examined the licensee's physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools.

The inspector reviewed the licensee's self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection. The inspector determined that identified problems were entered into the corrective action program for resolution.

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

The inspector reviewed licensee documentation packages for all performance indicator (PI) events occurring since the last inspection.

The inspector selected jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (<1 R/hr) for observation. The inspector reviewed all radiological job requirements and observed job performance with respect to these requirements. The inspector determined that radiological conditions in the work area were adequately communicated to workers through briefings and postings. The jobs reviewed and observed included the removal and replacement of the filter elements in the 2B condensate filtration system (CFS) filter.

The inspector discussed with first-line health physics (HP) supervisors the controls in place for special areas that have the potential to become very high radiation areas (VHRA) during certain plant operations. The inspector determined that these plant operations required communication beforehand with the HP group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

This inspection activity represented nine samples. The documents reviewed are provided in the attachment - "List of Documents Reviewed."

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector reviewed the licensee's self-assessments, audits, and special reports related to the As Low As Is Reasonably Achievable (ALARA) program since the last inspection. The inspector determined that the licensee's overall audit program scope and frequency (for all applicable areas under the Occupational Cornerstone) meet the requirements of 10 CFR 20.1101(c)).

The inspector 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. This inspection activity represented two samples. The documents reviewed are provided in the attachment - "List of Documents Reviewed."

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (7112103 - 2 Samples)

a. Inspection Scope

The inspector reviewed licensee self-assessments, audits, and Licensee Event Reports and focused on radiological incidents that involved personnel contamination monitor alarms due to personnel internal exposures.

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

The inspector reviewed documents related to the licensee's processing of thermoluminescent dosimeters (TLDs) to measure personnel doses of record. Documents reviewed included the most recent laboratory testing (Personnel Dosimetry Performance Testing Report dated 9 January 2004) and laboratory audit (On-Site Assessment 100554-0, February 2003) of the licensee's program and facility by the National Voluntary Laboratory Accreditation Program (NVLAP). This inspection activity represented two samples. The documents reviewed are provided in the attachment - "List of Documents Reviewed."

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152 - 4 samples)

1. Identification and Resolution of Problems - Occupational Radiation Safety

a. Inspection Scope

The inspector selected issues identified in the condition report system for detailed review. The issues were associated with occupational radiation safety performance during 2004. The inspector met with the plant radiation protection manager to discuss these reports. The documented reports for the issues were reviewed to ensure that the full extent of the issues was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. This inspection activity represented one sample.

b. Findings and Observations

No findings of significance were identified.

2. Annual Sample Review - Emergency Diesel Generator Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed PPL's initial evaluation and associated corrective actions for 50 condition reports (CRs) related to emergency diesel generator (EDG) problems between November 1999 and June 2004. This sample was selected due to the potential for an adverse trend with regard to resolving potentially repetitive EDG maintenance issues. The review evaluated PPL's threshold for identifying and resolving problems. This inspection activity represented one sample. The documents reviewed during this inspection are listed in the Supplemental Information Attachment (back of this report).

b. Findings and Observations

No findings of significance were identified.

Observations

The inspectors evaluated PPL's problem identification and resolution efforts for EDG issues for the previous five years. The inspectors determined that, in general, PPL was effective at identifying problems and placing them in the corrective action program. PPL evaluated many of the issues as only affecting the "test mode" of EDG operations, and concluded that the EDG would have been able to perform its safety function. As a result, PPL was slow to identify adverse trends in EDG maintenance effectiveness; PPL identified an adverse trend in EDG reliability in July 2003. The inspectors noted examples where problem resolution was narrowly focused or lacked sufficient technical analysis to prevent recurrence. As a result, PPL's EDG maintenance effectiveness remained weak during the 5-year period reviewed.

The inspectors identified a high failure rate for EDG post maintenance tests (PMTs) performed following EDG overhauls (9 failures out of 13). In each case, additional unplanned EDG unavailability and TS LCO time was incurred. In one instance, the 72-hour TS LCO expired, and a reactor shutdown was commenced.

In addition to a high post-overhaul test failure rate, the inspectors identified multiple examples, during the 5-year period, where weak maintenance practices resulted in EDGs becoming inoperable or unavailable during surveillance testing or standby service. The most significant issues included:

- January 2004, with the "B" EDG in standby, an emergency service water leak identified that half the bolts on the lube oil cooler heat exchanger were loose; attributed to a degraded gasket. Subsequent investigation identified multiple fastener issues on all five EDGs, and resulted in an NRC Special Inspection (IR 50-387,388/2004-007).
- January 2004, the "A" EDG failed a surveillance test due to an oil spray, which identified loose governor mounting bolts; attributed to improper torque when last replaced during the August 2003 overhaul.
- July 2003, with the "C" and "D" EDGs in standby, PPL identified that the generator outboard bearings had been improperly torqued when last assembled; attributed to an inadequate procedure, which was not previously identified because the mechanics relied on skill-of-the-craft for bearing assembly, not procedural step-by-step compliance.
- May 2003, the "A" EDG tripped during a surveillance test; attributed to a loose wire on a relay base. The relay base had been replaced in October 2002, as a corrective action for a similar EDG trip.
- May 2003, the "C" EDG failed a surveillance test because load could not be increased above 100 kW. This was attributed to a loose connection in the test circuitry.
- March 2003, the "D" EDG failed a surveillance test when the mechanical governor became disconnected from the fuel rack linkage because a connecting bolt fell out; initially, PPL attributed the failure to excessive vibration. The inspectors subsequently identified that the bolt had not been torqued when the governor was last replaced during the June 2002 overhaul.
- January 2003, the "B" EDG tripped during a PMT because a jumper had not been reinstalled on a motor-operated potentiometer, which had just been replaced.

PPL has recently downgraded the EDG overall system health because of a negative trend in system performance. PPL developed a plan to improve the overall system health and documented this in CR 580429. This plan "The Road to Green," will addresses maintenance issues by creating a multi-discipline Diesel Review Team which will address issues related to the maintenance of EDG components.

3. <u>Annual Sample Review - Diesel Driven Fire Pump Lack of Engine Cooling</u>

a. <u>Inspection Scope</u>

The inspector reviewed PPL's evaluations and associated corrective actions for selected condition reports related to the diesel fire pump (DFP). The DFP was selected due to the numerous condition reports written on the component and the risk insights with respect to late reactor vessel injection. The inspectors completed a detailed walkdown of the DFP engine and controls. The inspectors reviewed the FSAR, Operating procedures, Fire Protection Review Report, and the Individual Plant Evaluation (IPE) to ensure the equipment is operated and maintained consistent with the design and risk significance of the system. Inspectors also reviewed equipment vendor manual instructions to assess the significance of operation beyond established limits. This inspection activity represented one sample. The documents reviewed are provided in the report attachment.

b. <u>Findings and Observations</u>

<u>Introduction</u>. The NRC identified a Green finding for failing to adequately evaluate and correct high engine operating temperatures and repetitive overheating of the DFP after shutdown, in accordance with NDAP-QA-0702, "Action Request and Condition Report Process." For over three months, PPL did not evaluate or initiate specific action to correct the high temperature condition which resulted in repetitive maintenance and unavailability time for the DFP.

<u>Description</u>. During May, June, and July 2004, the inspectors observed several instances where the DFP was operated with cooling water temperatures exceeding the vendor's recommended high temperature of 195EF. The inspectors also observed instances in which the vendor's maximum engine coolant temperature of 200EF was exceeded after the engine was shutdown. The observations are listed below.

C May 2004, during post maintenance testing the inspectors observed the DFP engine overheat to greater than 220EF and the overflow of coolant immediately following engine shutdown.

C June 1, 2004, the DFP engine cooling water temperatures were above the operating procedure limit of 185EF.

C July 14, 2004, during troubleshooting and post maintenance testing, for a failure of the DFP to automatically start, inspectors observed high engine coolant temperatures greater than 195EF.

C July 21, 2004, the inspectors again observed high engine operating temperatures of 199EF and DFP engine temperature increased to 225EF when shutdown with the overflow of approximately two pints of engine coolant onto the engine. PPL did not initiate corrective action reports for the July 21, 2004 event, the plant operators referenced previously issued CRs in the work package.

Prior to the NRC inspectors questioning DFP availability and reliability, PPL did not evaluate the cause of the elevated engine temperatures and had not taken effective actions to correct the condition or evaluate the severity of the engine coolant overflow and engine overheating following shutdown. The DFP was removed from service for corrective maintenance numerous times without any action to restore engine cooling capability. The vendor manual states that if the engine is stopped suddenly, the turbocharger temperature can rise as much as 100EF. The Turbocharger is cooled by lubricating oil which is cooled by the engine coolant (water/glycol). A high degree of heat in the turbocharger can cause seizer of bearings, burned O-ring oil seals and the distortion of the bearing housing. Inspectors also determined that the overflow of engine coolant over the engine components including the starting solenoids has the potential to reduce component reliability.

On July 21, 2004, the inspectors discussed, with PPL, the repeated coolant overflow and engine overheating following shutdown as well as the elevated engine temperatures above procedure limits during pump operation. On July 27, 2004, PPL took corrective action to change the pressure control valve setpoint to increase cooling water flow to the DFP engine and bring engine temperatures back into the normal operating band. The cause of the inadequate cooling water flows was attributed to previously establishing a pressure control valve setpoint that was based on the temperature conditions in November but was not adequate for summer season operation. There was also instrument drift found in the pressure control valve setpoint that decreased flow to the engine heat exchanger.

Analysis. The finding is a performance deficiency because PPL failed to correct a degraded equipment condition that was captured in the corrective action system, specifically, the elevated temperatures and engine overheating of the DFP. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC’s regulatory function and was not the result of any willful violation of NRC requirements or PPL procedures. This issue is greater than minor because the DFP is utilized as a mitigation system and elevated temperatures, coolant overheating and coolant discharge, reduced the reliability and the availability of the system. Thus, the finding affected the Mitigating Systems cornerstone objective.

This finding was assessed in accordance with NRC Manual Chapter 0609, Appendix A, Attachment 1, "Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations," and was determined to be of very low safety significance (Green) based on a Phase 1 analysis. The inspectors determined that this condition did not represent an actual loss of the equipment designated as risk-significant for greater than 24 hours, because there was no loss of the reactor vessel late injection capability. Additionally, the finding does not increase the potential or risk of a seismic event, flood or severe weather event.

A contributing cause of this finding is related to the Problem Identification and Resolution (PI&R) cross-cutting area. PPL did not sufficiently evaluate the condition to identify and correct the reduced cooling water flow to the DFP engine. This resulted in ineffective corrective actions because the DFP was removed from service several times without taking action to correct the DFP high engine coolant temperature issue.

Enforcement. No NRC requirements were violated. The Diesel Driven Fire pump is not safety related and thus not covered by 10 CFR 50, Appendix B. PPL failed to correct a degraded condition on a system utilized to mitigate station risk. The issues associated with this finding are entered into the corrective action program under CR 594877.
FIN 05000387, 388/2004004-04, "Diesel Driven Fire Pump Lack of Engine Cooling."

4. Annual Sample Review - N2J and N1B Nozzle Weld Indications

a. Inspection Scope

The inspectors reviewed conditions reports associated with a crack at the weld root in each of the N1B and N2J safe end to nozzle welds in the Unit 1 recirculation system. These cracks were located during a scheduled Inservice Inspection (ISI) using computer based ultrasonic examination techniques. The condition reports were reviewed to ensure a complete and accurate identification of the issue, an appropriate root cause evaluation was performed, extent of condition was considered, and corrective actions were implemented. In addition to the condition reports, the inspector reviewed design calculations, welding procedures, and process qualifications used to support the repair effort.

The inspector also conducted interviews with personnel responsible for the nondestructive processes used to identify the indications, performance of the root cause evaluation, development of the repair plan, implementation of the plan and verification the repair met the specified acceptance criteria. This inspection activity represented one sample. The documents reviewed during this inspection are listed in the Supplemental Information Attachment (back of this report).

b. Findings and Observations

No findings of significance were identified.

Observations

PP&L discovered significant indications in the Unit 1 N1B and N2J safe end to reactor vessel nozzle welds through ultrasonic testing performed during refueling outage 13RIO (Spring 2004). Ultrasonic test data was evaluated and the indications identified were characterized as cracks which had initiated at the root of the welds and propagated into the weld thickness. Both indications were found to be acceptable within the ASME Section XI, IWB 3640 design requirements. However, PP&L elected to install a weld overlay on both welds to alleviate any concerns regarding crack growth with potential for penetration through the pressure boundary during subsequent operating cycles.

The inspector noted that these welds had been previously inspected at established ASME Code intervals using both surface (penetrant test) and volumetric (ultrasonic test) techniques. Tests were performed using qualified techniques and personnel as required by the ASME Section XI Code. Although indications were identified during these previous inspection periods, a number of the indications were not identified, sized or characterized as “cracks.” As a result of the failure to identify and characterize these indications as linear or, as cracks, no further flaw evaluations were performed since the indications were reported as “non-relevant” or “root geometry.” This was consistent with the limitations of the equipment and examination techniques in effect at the time of these previous examinations.

Ultrasonic examinations performed beginning in 13RIO (April 2004) were completed using Performance Demonstration Initiative (PDI) qualified procedures and examiners. The procedures, equipment and examiners qualified for this testing represent the most current available technology and equipment coupled with rigorous training and testing of personnel to assure optimum results. This examination system was used for the first time during 13RIO to examine the N1B and the N2J welds and had not been available for previous examinations.

The inspector reviewed the previous results of surface and volumetric tests performed on both the N1B and the N2J welds in an effort to determine why the indications were not identified, characterized and evaluated as “cracks” prior to the Spring 2004 examination. Based on this review, the inspector concluded that the failure to identify and adequately characterize the indications in the N1B and N2J nozzle to safe-end welds during previous examinations was the result of a combination of procedure, equipment and examiner limitations. However, the inspector noted that at the time the previous examinations were performed, the procedures, equipment and examiners were qualified to the requirements of ASME Section XI. Examinations were implemented and results interpreted in accordance with the Code requirements in effect at the time of the examinations. Indications noted were found to meet the acceptance criteria provided in the ASME Section XI Code.

5. Routine PI&R Review

a. Inspection Scope

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

b. Findings and Observations

No findings of significance were identified.

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

Section 1R12 describes a finding where PPL had numerous condition reports that discussed problems with the equipment floor drain system (EFDS) but did not identify the design significance of this system and did not specify adequate corrective actions to maintain the functionality of the system.

Section 4OA2.3 describes a finding where PPL did not sufficiently evaluate the overheating of the diesel fire pump (DFP) to identify and correct this degraded condition.

4OA3 Event Follow-up (71153 - 3 Samples)

1. (Closed) LER 05000387/2004001-00 Automatic Actuation of ‘A’ Emergency Diesel Generator when Operator Removed Incorrect 4.16 KV Bus Fuse

On March 7, 2004, the ‘A’ Emergency Diesel Generator (EDG) started due to a detected under-voltage condition on the 1A201 4.16 KV Engineering Safeguards bus. The bus undervoltage condition was created when an in-plant operator removed fuses from an incorrect bus breaker cubicle during equipment alignment.

The inspectors reviewed the event and documented their assessment in NRC Inspection Report 50-387,388/2004-002, Section 1R14 Item 2. The documented assessment included a very low safety significance (Green) NCV that was identified because a non-licensed plant operator did not implement the electrical system shutdown procedure as written. (NCV 05000387,388/2004002-01).

The inspectors reviewed the related information and no additional findings were noted. This issue was documented in PPL's corrective action program as CR 555676. This LER is closed.

2. (Closed) LER 05000387/2004002-00 Loss of Safety Function - Control Structure Chillers Inoperable due to Blank Flanges Being Installed in Wrong Emergency Service Water Loop (Common)

On March 9, 2004, PPL identified that both control room emergency outside air supply system (CREOAS) were inoperable because the “A” and “B” loops of emergency service water (ESW) system that supplied the cooling to the CREOAS system were inoperable and the associated TS limiting condition of operation had not been entered. Specifically, on March 5, 2004, during a refueling outage with irradiated fuel movements in progress, the emergency supply of cooling to “A” CREOAS was out of service when maintenance personnel incorrectly removed a blank flange in the Unit 1 “B” ESW loop. Removal of the blank flange resulted in the “B” ESW piping being in an unanalyzed seismic condition and was declared inoperable in accordance with TS. During the one hour period, both CREOAS trains were inoperable because the “A” ESW supply was not available and the “B” ESW supply was not fully seismically qualified. Fuel moves were not suspended as required by TS 3.7.3.F and 3.7.4.E. PPL determined the cause to be a lack of a pre-job brief and a poor turnover between maintenance shifts. Corrective actions included returning the ESW and CREOAS systems to an operable status and including this event into station training.

This finding is more than minor because it affected the human performance attribute of the containment barrier cornerstone. The finding was considered to have very low safety significance (Green) using SDP Phase 1 screening. Because this finding only represented a degradation of the radiological barrier function provided for the control room because the CREOAS system did not have a fully seismically qualified safety-related supply of cooling water for one hour. This licensee-identified finding involved a violation of TS 3.7.3 and 3.7.4. The enforcement aspects of this violation are discussed in Section 4OA7. This LER is closed.

3. (Closed) LER 05000387/2004003-00 Manual Scram Following Turbine High Vibration

On April 21, 2004, Unit 1 was operating at 17% power when the reactor was manually scrammed and condenser vacuum was broken to rapidly reduce turbine speed in response to main turbine bearing high vibrations. The high vibration was the result of component rubs that were expected, in part, due to the newly installed turbine and not setting the turbine vibration monitoring limits and turbine trip limit low enough to prevent high vibrations. PPL revised OP-1(2)93-001, “Main Turbine Operation,” to include clear guidance on turbine vibration monitoring and lower vibration limits to allow quicker detection of elevated turbine vibration. The LER was reviewed by the inspectors and no finding of significance were identified. PPL documented this condition in CR 573728. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

Cross-References to Human Performance Findings Documented Elsewhere

Section 1R19 describes a finding where PPL did not complete a required retest and engineering evaluation of a degraded transformer prior to returning the transformer to service.

4OA5 Other

1. Operation of an Independent Spent Fuel Storage Installation (60855)

a. Inspection Scope

The inspector observed selected spent fuel dry cask loading operations for dry shielded canister (DSC) 29 which were conducted in accordance with procedure ME-0RF-023, “Dry Fuel Storage - 61BT Dry Shielded Canister.” Specifically, installation and welding of the inner top cover, canister blowdown, and initial vacuum drying operations were observed. Vacuum drying and Helium leak test results for DSCs 26, 27, and 28 were reviewed with respect to nuclear horizontal modular storage (NUHOMS) 52B and 61BT Technical Specifications (TS) 1.2.2, 1.2.4 and 1.2.4a criteria, respectively. Radiological work practices and exposure rates were discussed with technicians responsible for ongoing work. Conformance to the requirements of TS 1.2.11 and 1.2.12, “Transfer Cask Dose Rates” and “Maximum DSC Removable Surface Contamination,” criteria, was evaluated. Personnel exposures were reviewed and radiation work permit (RWP) 2004-0200, “Dry Fuel Storage Activities On The Refuel Floor” was examined.

The inspector discussed with cognizant licensee representatives the procedural controls in place that ensured only designated fuel assemblies were properly selected and loaded into NUHOMS 52B and 61BT Casks. A review of the spent fuel assembly move sheets and verification records required by RE-081-042, “Fuel and Core Component Transfer Authorization Sheet (FACCTAS) Preparation Guidelines,” and RE-081-043, “Selection and Monitoring of Fuel for Dry Storage,” was conducted. The inspector observed a video tape of final fuel configuration in NUHOMS 52B DSCs 26 and 27, and NUHOMS 61BT DSCs 28 and 29, which indicated fuel assembly serial numbers. Fuel characteristics including enrichments, burn-up, post irradiation cooling time, heat generation, and known structural defects, were reviewed and evaluated against the NUHOMS TS 1.2.1 limits.

The inspector reviewed 10CFR72.48 safety evaluations generated since the last spent fuel transfer campaign in 2002, including a 72.48 screen (7248-01-117, “Convert to 61BT Dry Shielded Canisters”) which evaluated the effect of Amendment 4 to Certificate of Compliance 1004 on the NUHOMS Storage System at Susquehanna. The inspector also reviewed Evaluation SE 72-1778: Use of Wrong Gas in DSC Canister.

Training and qualifications of selected personnel involved with dry cask storage work was reviewed to ensure adherence to the general license criteria in 10CFR72.212(b)(6). This review included personnel responsible for rigging and cask handling, welding, vacuum drying, helium backfill operations, and helium leak testing. Inclusion of operating experience was also evaluated.

Effectiveness of licensee management oversight, quality assurance (QA) audits and self assessments, and corrective action program as applied to the dry cask storage program was evaluated. A management readiness review, conducted prior to the start of this campaign and approved by the station PORC, was reviewed. The inspector discussed operational oversight with a QA inspector on the refuel floor during operational activities associated with DSC 29.

A tour of the ISFSI pad and enclosed area was conducted to ensure proper housekeeping and conformance to combustible loading limits.

b. Findings

No findings of significance were identified.

2. (Closed) SL-III Violation 05-387, 388/2002005-02, Spent Fuel Cannister Filled with Wrong Gas

On July 26, 2002, PPL filled a spent fuel storage canister (DSC) with Argon and Helium gases instead of using all Helium gas as required by the NUHOMS Technical Specification. A Severity Level III violation (with no civil penalty) was issued to PPL for this event in inspection report 2002-005. During the inspection, PPL's corrective actions to prevent the use of the Argon gas were reviewed. The inspector verified procedure changes were completed and training on the revised procedures conducted prior to the start of the next dry cask storage campaign. The inspector observed equipment set up on the refuel floor and determined that adequate controls were instituted to prevent recurrence of the violation. This item is considered closed.

4OA6 Meetings, Including Exit

On October 7, 2004, the resident inspectors presented the inspection results to Mr. B. McKinney, Vice President - Nuclear Operations, and other members of your 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.

C TS 3.7.3.F and 3.7.4.E requires that fuel moves be immediately terminated when both trains of CREOAS are inoperable. Contrary to this on March 5, 2004, both trains of CREOAS were determined to be inoperable due to a maintenance error on a support system and fuel moves were not suspended. This was identified in PPL's corrective action program as CR 556923. This finding is of very low safety significance because even though the “B” ESW system was not fully seismically qualified, PPL determined that the system remained capable of supplying water to the necessary heat loads during this period.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINT OF CONTACT

Licensee Personnel

S. Ingram, Senior Health Physicist
E. McIlvaine, Jr., Health Physics Operations Foreman
V. Schuman, Radiological Operations Supervisor
L. Wolf, Health Physics Operations Foreman

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

050000387/2004004-01	URI	Equipment Hatch Floor Plugs are not Watertight as Indicated in the FSAR

Opened and Closed

05000387/2004004-02	NCV	Reactor Building Floor and Equipment Drains Not Fully Scoped into the Maintenance Rule
05000388/2004004-03	FIN	PPL Did Not Retest and Evaluate Transformer 2X270
05000387, 388/2004004-04	FIN	Diesel Driven Fire Pump Lack of Engine Cooling

Closed

05000387/2004001-00	LER	Automatic Actuation of 'A' Emergency Diesel Generator When Operator Removed Incorrect 4.16KV Bus Fuses (Common)
05000387/2004002-00	LER	Loss of Safety Function - Control Structure Chillers Inoperable due to Blank Flanges Being Installed in Wrong Emergency Service Water Loop (Common)
05000387/2004003-00	LER	Manual Scram Following Turbine High Vibration
05000387, 388/2002005-02	VIO	Spent Fuel Cannister Filled with Wrong Gas.

LIST OF BASELINE INSPECTIONS PERFORMED

7112101	Access Control	2OS1
7112102	ALARA Planning and Controls	2OS2
7112103	Radiation Monitoring Instrumentation	2OS3

LIST OF DOCUMENTS REVIEWED

(Not Referenced in the Report)

<u>1R06: Flood Protection Measures</u>

<u>Internal Flood Protection</u>

EC-FLOD-0500,	Review of MISC-037 calculation for Determining Correct Flood Depths for Reactor Building Elevation 683 Foot
EC-FLOD-0001,	Moderate Energy Pipe Breaks - Floods
EC-012-6047	Removal of Floor Plugs in Conditions 4 or 5
EC-RISK-0539	Internal Flooding Analysis for PRA
PPL DBD 0010	
Section 2.10,	Internal Flooding Functional Requirements
FSAR Section 3.4.1	Water Level Design / Flood Protection
FSAR Section 3.6	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
FSAR Section 3.12	Separation Criteria for Safety-Related Mechanical and Electrical Power Equipment
FSAR Section 6.3	Emergency Core Cooling Systems
CR 600070	Operability Determination, Rev. 0
CR 600070	Operability Determination, Rev. 1

PPL Evaluation of Water Intrusion into ECCS Rooms

Regulatory Guide 1.102, Rev. 1, Flood Protection for Nuclear Power Plants

<u>2OS1: Access Control to Radiologically Significant Areas</u>

<u>Section 2: Radiation Safety</u>

<u>Audit Reports</u>

AR 542112, Y2003 Trend Evaluation
AR 551986, Radiation Protection Activities Observed during the Unit 1 13th RIO

Self-Assessments

HPS-03-003, Training Program Self-Assessment
HPS-03-04, Comprehensive RP Assessment (10CFR20.1103c)
HPS-03-05, Assessment of Batelle PNL Irradiation Services
HPS-03-06, Radioactive Material Control

Condition Reports

559453, 569363, 569364, 569688, 573291, 574278, 577880, 578045, 578630, 579754, 580133, 580364, 580984, 581740, 581960, 582250, 582256, 582262, 582272, 582277, 582279, 582660, 589439, 590163, 590181, 590286, 591947, 593065

4OA2: Identification and Resolution of Problems

Annual Sample Review - EDG Maintenance Effectiveness

Procedures

NDAP-QA-0702, "Action Request and Condition Report Process"
IC-024-001, "Electronic Tune-up for Diesel Generator Governors"

Technical Specification (TS) 3.8.1, surveillance requirements, and TS Basis

Miscellaneous

"A" EDG Speed Control Troubleshooting Report, MPR Associates Inc., dated December 13, 1999
"Unexpected "A" Diesel Generator Operation Root Cause Analysis Report," ANNA Inc., dated January 18, 2000
"Memorandum to ESI Source Inspection Report 2000-20," Engine Systems Inc., dated July 18, 2000
"Laboratory Examination of Failed Bearing Temperature Sensor for Solder Coverage," PPL Metals Lab Report 2001-094, dated September 18, 2001
"Failure Analysis of Cracked Lower Cylinder Liner Expansion Sleeve," PPL Metals Lab Report 2002-003, dated January 18, 2002
"EDG Governor Component Failure Analysis Report MPR-2513," MPR Associates Inc., dated March 26, 2003
"DG Reliability Report," PPL Trend Review Team, dated September 11, 2003

Condition Reports

585913, 579510, 564942, 544874, 543502, 543496, 543172, 532012, 532461, 512613, 512609, 508825, 505495, 488000, 487476, 484120, 478691, 475852, 474234, 460227, 454959, 446398, 445656, 445506, 445489, 445379, 445361, 445171, 445139, 430146, 428743, 408025, 376231, 362560, 353773, 353093, 340936, 316629, 290686, 274297, 273269, 270141, 269851, 265604, 265556, 265475, 218433, 216004, 212940, and 212844

Annual Sample Review - N2J and N1B Nozzle Weld Indications

Condition Reports,

561319, 567886, 561319, 567886, 573890, 592892, 58437, 573883, 573882, 564420, 563116, 573889, 573893, 562164, 573887, 562886, 567886, 570394, 565923, 566347

Miscellaneous

Level 1 Root Cause Evaluation for Unit 1N1B and N2J Nozzle Flaws
Overlay Weld Procedure WPS 03-08-T-801 Rev 0
Weld Overlay Repair Plan SIR-04-038 Rev 0
Summary Reports of UT Exam of Unit 1 N1B and N2J Nozzle to Safe End Weld
MSIP Performance and Verification Record for Unit 1 N1B and N2J Welds
Engineering Calculation EC-062-1083 Rev 0
PCN 2004-6463, 6465, N1B and N2J Overlay Thickness Verification
DCP 567820, Rev 0, Unit 1 Weld Overlay Installation Instructions

LIST OF ACRONYMS

ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CFS	Condensate Filtration System
CIG	Containment Instrument Gas
CR	Condition Report
CREOAS	Control Room Emergency Outside Air Supply
DFP	Diesel Fire Pump
DSC	Dry Shielded Canister
EAL	Emergency Action Level
ECCS	Emergency Core Cooling Systems
EDG	Emergency Diesel Generator
EFDS	Equipment Floor Drain System
EP	Emergency Preparedness
EF	Degrees Fahrenheit
FACCTAS	Fuel and Core Component Transfer Authorization Sheet
FI	Flow Indicator
ESW	Emergency Service Water
FSAR	[SSES] Final Safety Analysis Report
HP	Health Physics
HPCI	High Pressure Coolant Injection
HRA	High Radiation Area
HSM	Horizontal Storage Module
ICM	Interim Compensatory Measures
IMC	[NRC] Inspection Manual Chapter
IPE	Individual Plant Evaluation

ISFSI	Independent Spent Fuel Storage Installation
KV	Killovolts
kW	Killowatts
LCO	[TS] Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LER	Licensee Event Report
MSIV	Main Steam Isolation Valve
NCV	Non-cited Violation
NDAP	Nuclear Department Administrative Procedure
NPO	Nuclear Plant Operator
NRC	Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Modular Storage
NVLAP	National Voluntary Laboratory Accreditation Program
PCO	Plant Control Operator
PI	[NRC] Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post Maintenance Test
PPL	PPL Susquehanna, LLC
PSV	Pressure Safety valve
QA	Quality Assurance
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RG	[NRC] Regulatory Guide
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RSPS	Risk Significant Planning Standards
RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SDP	Significant Determination Process
SLC	Standby Liquid Control
SOW	System Outage Window
SPDS	Safety Parameter Display System
SPING	System Site Particulate Iodine and Nobel Gas
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
TBCCW	Turbine Building Closed Cooling Water
TLD	Thermoluminescent Dosimeter
TMOD	Temporary Modifications
TS	Technical Specification
URI	Unresolved Issue
VHRA	Very High Radiation Area
WO	Work Order