UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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November 12, 2010

EA-10-207

Mr. Timothy S. Rausch
Senior Vice President and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Boulevard, NUCSB3
Berwick, PA 18603

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION – NRC INTEGRATED INSPECTION REPORT 05000387/2010004 AND 05000388/2010004; AND PRELIMINARY WHITE FINDING

Dear Mr. Rausch:

On September 30, 2010, 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 inspection results, which were discussed with Mr. Jeff Helsel and other members of your staff during an exit meeting on October 14, 2010.

This inspection examined activities completed 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.

Based on the results of this inspection, a finding was identified which, using the applicable Significance Determination Process (SDP), has preliminarily been determined to be of low to moderate safety significance (White). As described in this report, the finding involved inadequate procedures related to the maintenance and operation of the main condenser waterboxes and circulating water system which, on July 16, 2010, resulted in an internal flooding event, a manual reactor scram, and loss of the normal heat sink. Specifically, a maintenance procedure contained inadequate condenser waterbox gasket installation instructions which led to the event. Furthermore, operator response to the event was complicated and delayed by two inadequate off-normal procedures. One off-normal procedure contained an incorrect diagram that operators used to identify and isolate the leak and the other lacked specific instructions to isolate a leak associated with the condenser waterboxes. There were no impacts to safety-related equipment as a result of the flooding.

In accordance with NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The SDP encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before we make a final decision on this matter, we are providing you with an opportunity to (1) attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of your receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either, you fail to meet the appeal requirements stated in IMC 0609, Attachment 2, Section 2, "Prerequisites," and Section 3, "Limitations."

Please contact Mr. Paul Krohn at 610-337-5120 and in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence. In addition, please be advised that the characterization of the finding described in the enclosed inspection report may change as a result of further NRC review.

In addition, this report documents two NRC-identified findings and two self-revealing findings of very low safety significance (Green). Each of these findings was determined to involve a violation of NRC requirements. Additionally, three licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program (CAP), the NRC is treating these as non-cited violations (NCV), consistent with Section 2.3.2 of the NRC's Enforcement Policy. If you contest any 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 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 NRC, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Susquehanna Steam Electric Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at the Susquehanna Steam Electric Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

David C. Lew
Director
Division of Reactor Projects

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

Enclosures: Inspection Report 05000387/2010004 and 05000388/2010004
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U.S NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-387, 50-388
License No:	NPF-14, NPF-22
Report No:	05000387/2010004 and 05000388/2010004
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	Berwick, Pennsylvania
Dates:	July 1, 2010 through September 30, 2010
Inspectors:	P. Finney, Senior Resident Inspector J. Greives, Resident Inspector A. Rosebrook, Senior Project Engineer J. Furia, Senior Health Physicist D. Molteni, Operator Engineer R. McKinley, Senior Emergency Response Coordinator
Approved By:	Paul G. Krohn, Chief Projects Branch 4 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000387/2010004, 05000388/2010004, 07/01/2010 – 09/30/2010; Susquehanna Steam Electric Station, Units 1 and 2; Flood Protection Measures, Operability Evaluations, Identification and Resolution of Problems, Event Follow-up.

The report covered a three month period of inspection by resident inspectors and announced inspections by regional reactor inspectors. One preliminary greater-than-green finding was identified and four Green non-cited violations (NCVs) of very low safety significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects associated with findings are determined using IMC 0310, "Components Within The Cross-Cutting Areas," dated February 2010. 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Finding To-Be-Determined: A self-revealing preliminary White finding regarding procedure NDAP-QA-0008, "Procedure Writer's Guide," Revision 8, was identified following a July 16, 2010, flooding event in the Unit 1 condenser bay which resulted in a manual reactor scram and loss of the normal heat sink. There were three instances of inadequate procedures identified. The first instance involved maintenance procedure MT-043-001 which provided inadequate instructions regarding installation of the condenser waterbox gaskets and led to the event. In addition, two other off-normal procedures were inadequate in that they complicated operator response to the event. Specifically, operators used a diagram in off-normal procedure ON-100-003, "Chemistry Anomaly," to identify and isolate the leak which was incorrect, delayed leak isolation, and resulted in a manual reactor scram in anticipation of a loss of the normal heat sink. Finally, ON-142-001, "Circulating Water (CW) Leak," did not contain specific instructions to isolate a condenser waterbox leak which contributed to operators using ON-100-003 which was not intended to be used to isolate the condenser box during flooding conditions. PPL corrected the diagram error, dewatered and repaired affected equipment, and entered this issue into their CAP (1282128).

This finding was determined to be more than minor as it affected the Initiating Events cornerstone attribute of Procedure Quality and its objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operation. The finding was evaluated using Phases 1, 2, and 3 of the Significance Determination Process. The conclusion of the Phase 3 analysis was an estimated change in core damage frequency (CDF) of 1.1E-6/yr (White) and an estimated change in large early release frequency (LERF) of 2.6E-7/yr (White). The finding is related to the cross-cutting area of Problem Identification and Resolution, Corrective Acton Program, in that PPL did not thoroughly evaluate problems such that the resolutions address the causes and extent of condition, as necessary. Specifically, PPL did not appropriately evaluate and correct a known issue in an off-normal procedure or adequately evaluate previous CW system waterbox manway gasket leaks to ensure that future occurrences could be prevented. (P.1.(c)) (Section 1R06)

Cornerstone: Mitigating Systems

Green: The inspectors identified a Green NCV of Susquehanna Unit 1, TS 5.4.1, "Procedures," for an inadequate procedure to transfer water from the condenser area to the condensate storage tank (CST) berm. Specifically, the procedure failed to include a maximum level in the CST berm that was acceptable to limit interactions with other safety-related equipment. The NCV was identified following the July 16, 2010, Unit 1 manual reactor scram due to a non-isolable circulating water leak in the main condenser area. Operations personnel commenced dewatering efforts by transferring water from the condenser area to the CST berm using a "Liquid Radwaste Collection" operating procedure as a guide. Water was transferred to the berm to a level sufficient to cause water intrusion into cable conduit and junction boxes containing High Pressure Coolant Injection system (HPCI) and Reactor Coolant Isolation Cooling system (RCIC) CST low-level suction instrumentation which transfers HPCI and RCIC pump suction from the CST to the suppression pool. As a result, the low-level suction instrumentation became submerged affecting the reliability and capability of the HPCI and RCIC CST to suppression pool transfer function despite being required in Mode 3. The issue was entered into PPL's CAP (1297039).

This performance deficiency is more than minor as it affected the equipment performance and procedural quality attributes of the corresponding Mitigating Systems cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the low-level suction instrumentation was not designed for submergence. Transferring too much water from the condenser bay to the CST berm submerged the low-level suction instrumentation and affected the reliability and capability of the HPCI and RCIC CST to suppression pool transfer function. The finding was evaluated for significance using IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings." Since the finding did not result in a loss of safety function or the loss of a train for greater than its TS allowed outage time, and was not potentially risk significant due to external event initiators, the finding was determined to be of very low safety significance (Green). This finding was determined to have a cross-cutting aspect in the area of Human Performance, Resources, because PPL did not ensure that procedures were adequate to assure nuclear safety. Specifically, operating procedure OP-169-004, Revision 17, did not specify a maximum level that could be transferred to the CST berm to limit interactions with safety-related, HPCI and RCIC low-level suction transfer instrumentation. (H.2(c)) (Section 1R15)

Green: An NRC-identified, Green NCV of 10 CFR 55.46(c)(1), "Plant Referenced Simulators," was identified because the Susquehanna simulator did not accurately model RCIC system response when operated in automatic flow control at less than design basis full flow. While the licensee has not yet completed simulator modifications to routinely model RCIC control system instabilities when operating the system in automatic flow control at less than design basis full flow, the simulator does model instabilities resulting from a control system malfunction. The inspectors verified that licensed operators have trained on and responded to RCIC control system malfunctions during examinations. This issue was entered in PPL's corrective action process as CRs 1285503, 1287462, and 1286803.

The performance deficiency is more than minor because it is associated with the Human Performance attribute of Mitigating Systems and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the modeling of the Susquehanna simulator introduced negative operator training that could affect the ability of the operators to take the appropriate actions during an actual event. The finding was determined to be of very low safety significance because it is not related to operator performance during requalification, it is related to simulator fidelity, and it could have a negative impact on operator actions.

This issue was determined to not have a cross-cutting aspect. This was based on the age of the EPRI guidance (issued in 2002) applicable to the RCIC system flow instabilities and the lack of opportunities over the past three years to revisit this guidance. Therefore, this issue was not reflective of current performance. (Section 1R15)

Green: A self-revealing NCV of 10 CFR 55.46(c)(1), "Plant Referenced Simulators," was identified because the Susquehanna simulator did not accurately model integrated control system (ICS) response to reactor pressure vessel (RPV) level transients. This violation was due to an error in the simulator modeling that caused RPV level control in the simulator to respond more rapidly than the actual plant resulting in the simulation of a more stable response and smaller overall changes in RPV level during level transients in the simulator. This error contributed to the decision to proceed with an extended power uprate (EPU) required condensate pump trip test during reactor power ascension activities. As a result on May 14, 2010, when the condensate pump trip test was performed, the ICS system was unable to adequately control reactor vessel water level and operators inserted a manual reactor scram prior to a high level turbine trip at level 8. PPL completed corrective actions to update the simulator model to accurately reflect the feedwater flow component of ICS and has ensured that the simulator reflects actual plant performance and re-performed the condensate pump trip test. This issue was entered in PPL's corrective action process as AR/CR 1257781.

The performance deficiency is more than minor because it is associated with the Human Performance attribute of Mitigating Systems and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the modeling of the Susquehanna simulator introduced negative operator training that affected the ability of the operators to take the appropriate actions during an actual event. The finding was determined to be of very low safety significance because it is not related to operator performance during requalification, it is related to simulator fidelity, and it had a negative impact on the timeliness of operator actions during an actual plant transient. This finding has a cross-cutting aspect in the area of Human Performance, Resources, because PPL did not ensure that equipment and other resources were available and adequate to assure safety. Specifically, simulator fidelity was inadequate in that modeling information provided by the simulator vendor was not reviewed by PPL nor was an alternate methodology used to validate simulator performance prior to use in operator training and predictions of actual plant response. In addition, ICS adjustments made after the April 22, 2010, scram provided another opportunity to verify the validity of ICS gain settings. (H.2(d)) (Section 4OA2)

Cornerstone: Emergency Preparedness

Green: A Green self-revealing NCV associated with emergency planning standard 10 CFR 50.47(b)(4) was identified regarding inadequate indications for operators to determine if a threshold for an Alert Emergency Action Level (EAL) (OA7) declaration based on toxic gas concentrations immediately dangerous to life and health (IDLH) within a vital area had been met. Specifically, there were no meters (permanently installed or portable) available on site to measure Freon concentration, a toxic gas in high concentrations. This impacted the operator's ability to make an EAL declaration and operators had to rely on other indications such as personal ill effects from exposure. PPL entered this issue into its CAP as AR 1294109 and is evaluating the development of permanent corrective actions.

This performance deficiency is more than minor because it was associated with the Emergency Preparedness (EP) cornerstone attribute of Facilities and Equipment, and affected the cornerstone objective of ensuring that a licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. This finding was similar to an example of a green finding evaluated using IMC 0609, Appendix B, "Emergency Preparedness SDP," Sheet 1, "Failure to Comply." This finding is associated with a failure to meet or implement a regulatory requirement. The deficiency is not greater than Green because it did not result in the Risk-Significant Planning Standard Function being lost or degraded and was similar to an example of a green finding in that "the EAL classification process would not declare any Alert or Notification of Unusual Event that should be declared." Since the declaration of Alert OA7 based on toxic gas levels for Freon concentrations IDLH (defined as greater than 2000 ppm Freon) within a vital area could have been missed or delayed, this finding was considered consistent with the example provided and was determined to be of very low safety significance (Green). This finding is related to the cross-cutting area of Human Performance, Resources, because PPL did not ensure that equipment and other resources were available and adequate to assure safety. Specifically, PPL did not appropriately evaluate equipment necessary to effect a change to the emergency plan for an EAL classification related to toxic gasses in a vital area. PPL lacked adequate equipment to make an accurate EAL classification and had to rely on secondary means (personnel ill effects) for appropriately classifying a Freon leak in the Unit 1 RB that occurred on August 10, 2010. This was determined to be the most significant contributing factor to this issue. [H.2(d)] (Section 4OA3)

B. Licensee Identified Violations

Violations of very low safety significance, identified by PPL, were reviewed by the inspectors. Corrective actions taken or planned by PPL have been entered into PPL's CAP. These violations and corrective action tracking numbers 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 100 percent of its licensed reactor thermal power (RTP). On July 16, two condenser manway gaskets failed releasing approximately one million gallons of CW into the condenser bay before the condenser was isolated and the reactor was manually scrammed. Following repairs, a reactor startup was commenced on August 2 and full RTP was reached on August 7. On August 10, an Alert was declared based on a Freon leak from the 1A reactor building chiller. The unit remained at full RTP during this declaration and through the end of the inspection period.

Unit 2 began the inspection period at the authorized licensed power level of 94.4 percent RTP. On July 10, the unit was reduced to 79 percent RTP over six hours in support of condenser waterbox cleaning. The unit operated at 94.4 percent RTP for the remainder of the inspection period.

Note: The licensed RTP for both units is 3952 megawatts thermal. The Extended Power Uprate (EPU) License Amendment for SSES was approved in January 30, 2008, and was implemented for both units in accordance with the issued license conditions. For the purposes of this report and the remainder of the current operating cycle, the authorized power level for Unit 1 is 100 percent of the EPU licensed power limit. For the current operating cycle, the authorized power level for Unit 2 is 94.4 percent of the EPU licensed power limit.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

.1 Partial Walkdown (71111.04Q – 3 samples)

a. Inspection Scope

The inspectors performed partial walkdowns to verify system and component alignment and to identify any discrepancies that would impact system operability. The inspectors verified that selected portions of redundant or backup systems or trains were available while certain system components were out-of-service (OOS). The inspectors reviewed selected valve positions, electrical power availability, and the general condition of major system components. Documents reviewed are listed in the Attachment. The walkdowns included the following systems:

- Unit 1, Instrument Air (IA) system during “A” IA compressor maintenance;
- Unit 2, RCIC during HPCI system outage window (SOW); and
- Unit 2, “B” 125 VDC system.

b. Findings

No findings were identified.

.2 Complete Walkdown (71111.04S - 1 Sample)

a. Inspection Scope

The inspectors performed a detailed review of the alignment and condition of the Unit 2, Division II, emergency service water (ESW) system. The inspectors reviewed operating procedures, checkoff lists, and system piping and instrumentation drawings. Walkdowns of accessible portions of the system were performed to verify components were in their correct positions and to assess the material condition of systems and components. The inspectors evaluated ongoing maintenance and outstanding condition reports (CRs) associated with the ESW system to determine the effect on system health and reliability. The inspectors verified proper system alignment and looked at system operating parameters.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Fire Protection – Tours (71111.05Q - 5 samples)

a. Inspection Scope

The inspectors reviewed PPL's fire protection program to evaluate the specified fire protection design features, fire area boundaries, and combustible loading requirements for selected areas. The inspectors walked down these 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. The inspected areas included:

- Unit 1, Reactor Building (RB) 670' elevation (FZ 1-2A, 1-2B, 1-2C, 1-2D);
- Unit 1, RB heating and ventilation filter rooms, (Fire Zone 1-7A);
- Unit 2, Containment access area (Fire Zones 2-4A-N, W, and S);
- Common, Emergency Diesel Generator (EDG) "B" bay, (Fire Zone 0-41B); and
- Common, Engineering Safeguards Service Water (ESSW) pump house, (Fire Zone 0-51 and 0-52).

b. Findings

No findings were identified.

.2 Fire Protection – Drill Observation (71111.05A – 1 Sample)

a. Inspection Scope

On July 1, 2010, the inspectors observed an unannounced fire drill for the "E" shift conducted in the vicinity of the Motor Generator Area Load Center of the Unit 2 turbine building to evaluate fire brigade performance. The inspectors evaluated whether fire brigade members responded in the appropriate number, correctly donned the proper gear, carried and applied the proper fire protection equipment, and arrived at the scene in a timely manner. Further, the inspectors evaluated the fire brigade leader's command

and control as well as communications throughout the fire response organization. Finally, the inspectors observed the drill evaluators' conduct and control during the drill to include the post-drill critique and evaluation against established acceptance criteria. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R06 Flood Protection Measures

.1 Internal Flooding (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed documents, interviewed plant personnel, and walked down structures, systems and components (SSCs) to evaluate the adequacy of PPL's response to a Unit 1, condenser bay, internal flooding event that occurred on July 16, 2010. The inspection focused on PPL's response to the flooding event and the adequacy of maintenance and off-normal procedures. The inspectors also evaluated whether mitigation plans and equipment were consistent with design requirements and risk analysis assumptions and the material condition of credited components such as watertight plugs, floor drains, flood detection equipment, and alarms.

b. Findings

Introduction: A self-revealing preliminary White finding regarding procedure NDAP-QA-0008, "Procedure Writer's Guide," Revision 8, was identified following a July 16, 2010, flooding event in the Unit 1 condenser bay which resulted in a manual reactor scram and loss of the normal heat sink. There were three instances of inadequate procedures identified. The first instance involved maintenance procedure MT-043-001 which provided inadequate instructions regarding installation of the condenser waterbox gaskets and led to the event. In addition, two other off-normal procedures were inadequate in that they complicated operator response to the event. Specifically, operators used a diagram in off-normal procedure ON-100-003, "Chemistry Anomaly," to identify and isolate the leak which was incorrect, delayed leak isolation, and resulted in a manual reactor scram to isolate the condenser and stop the leak. Finally, ON-142-001, "Circulating Water (CW) Leak," did not contain specific instructions to isolate a condenser waterbox leak. PPL corrected the diagram error, dewatered and repaired affected equipment, and entered this issue into their CAP (1282128).

Description: On July 16, 2010, Unit 1 received a condenser area flood alarm followed by a condenser bay transfer sump high level alarm. Nuclear Plant Operators (NPOs) were dispatched to the area and identified a large leak on a CW inlet waterbox manway to the low pressure (LP) condenser. CW enters the condenser in a parallel flow arrangement through four waterboxes that are individually isolable by motor-operated butterfly valves on the inlet and outlet of the condenser for each waterbox. The off-normal procedure for flooding in the condenser bay, ON-142-001, Revision 17, "CW Leak," directed that, if the leak was within the isolation valve boundary, the leak be isolated, but provided no additional details on how that isolation should occur.

Using information provided by the NPOs regarding the location of the leak and comparing the information to a drawing depicting waterbox layout in ON-100-003, Revision 24, "Chemistry Anomaly," control room operators determined that 'B' waterbox was leaking and commenced isolation activities (note – the drawing in ON-100-003 was later determined to be incorrect). Despite complete closure of both the inlet and outlet isolation valves for the 'B' waterbox, NPOs reported that the leak continued (as the leak was actually coming from the 'D' waterbox). Control room personnel continued with their leak isolation by restoring the "B" waterbox and commenced isolation of the next waterbox. Following restoration of the "B" waterbox, NPOs reported that the leak had worsened. Control room operators continued making attempts to isolate other waterboxes, however they were unable to shut both the inlet and outlet valves on any of the remaining waterboxes due to wetting/submergence of the valve operators. Operators made the decision to manually scram the reactor, shut the main steam isolation valves (MSIVs), and isolate the main condenser so that CW could be secured and the leak stopped. Approximately 1,000,000 gallons of CW entered the Unit 1 main condenser bay, filling the bay to a depth of approximately 12 feet before the leak was isolated. Post event walkdowns identified that two waterbox manway gaskets on the 'B' and 'D' waterboxes had been extruded.

During a subsequent root cause analysis (RCA), three inadequate procedures were identified which contributed to the event. First, it was determined that the leak initiated from the 'D' manway cover gasket being partially extruded under normal system operating pressures. This was caused by an inadequate procedure to install the manway gaskets upon completion of maintenance. Specifically, manway gasket installation is performed in accordance with MT-043-001, Revision 14, "Main Condenser Leak Detection, Tube Pulling, Waterbox Inspection and Cleaning." This procedure lacked specific instructions to check the applied torque after sufficient time had elapsed to allow for gasket relaxation and creep. By not re-checking the torque, the resulting torque on the gasket hold down bolts dropped below the required value, resulting in inadequate preload on the gasket. In addition, MT-043-001 provided no guidance to torque the hold down bolts in a manner that ensured equal compression (i.e., "star pattern") of the neoprene gasket and the torque value specified in the work order (65 ft-lbs) was not sufficient to ensure the vendor recommended 50% gasket crush. Post event, the vendor recommended torquing the hold down bolts to 110 ft-lbs. Post event walkdowns discovered that a significant number of the cover hold down bolts were found to be 30 ft-lbs or less and some were only hand tight. The licensee concluded that this was due to relaxation of the neoprene gasket and creep. The licensee also identified that similar CW leaks associated with the manways and/or gaskets in April 2007 (AR 866034) and March 2008 (AR 1004556) were inadequately evaluated. Specifically, the two events provided opportunities to review the manway gasket installation procedure and make modifications, but the opportunities were not realized.

Second, the licensee identified that the attachment in ON-100-003, "Chemistry Anomaly," was incorrect. In particular, the 'D' waterbox was mis-labeled as 'B'. This led to operators in the field misidentifying the waterbox that was leaking and the operators in the control room selecting the wrong waterbox to isolate. Whereas the initial leak was on the 'D' waterbox, the pressure transient that was placed on the 'B' waterbox when it was isolated and unisolated was determined to have caused the 'B' waterbox gasket to be extruded which further complicated the leak isolation efforts. This was confirmed by the NPO's report that the leak worsened after the 'B' waterbox was restored. As the environment inside the main condenser bay degraded the motor-operated valves began to malfunction due to wetting/submergence. The operators determined the leak was not isolable and took action to isolated the main condenser in order to secure the CW system to stop the leak. Inaccuracies in the attachment to ON-100-003 had been identified in November 2009 and entered into the CAP (AR 1184479), but this was not evaluated as a condition adverse to quality. In March 2010, the AR was closed without correction to the procedure based, in part, on the drawing deficiencies not impacting chemistry department activities.

Finally, it was identified that procedure ON-142-001, "CW Leak," did not have specific instructions on how to isolate a condenser waterbox leak. The procedure was written to respond to an unisolable leak in other parts of the CW system and no guidance was provided to assist the operators in identifying the location and isolating leaks associated with the waterboxes.

NDAP-QA-0008, "Procedure Writer's Guide," Revision 8, Attachment A, states that "Off-Normal Procedures specify operator actions....to restore normal operating conditions following a perturbation. Such actions are invoked... which, if not corrected, could degenerate into a condition requiring action under an Emergency Procedure." It also states that procedures for performing maintenance "contain enough detail to permit the maintenance work to be performed correctly and safely." Based on these requirements, the inspectors determined that having the inadequate off-normal and maintenance procedures was a performance deficiency. Based upon the previous opportunities to identify and correct these inadequate procedures, it was determined that the performance deficiency was within PPL's ability to foresee and prevent.

Analysis: The inspectors determined that: 1) inadequate maintenance procedures for securing the condenser manway covers; and 2) inadequate off-normal procedures to locate and isolate an internal flooding event associated with the condenser waterboxes was a performance deficiency within PPL's ability to foresee and prevent. The inspectors screened the performance deficiency in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening." The performance deficiency was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of Procedure Quality, and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operation.

The inspectors evaluated the finding in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Initiating Events cornerstone. The inspectors answered "Yes" to the screening question, "Does the finding contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions will not be available?" since the condenser is listed as mitigation equipment for the Power Conversion System in the Phase 2 SDP Notebook for Susquehanna Steam Electric Station. Therefore, a Phase 2 SDP evaluation was performed using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations."

For the Phase 2 SDP evaluation, the senior reactor analyst (SRA) used the Risk-Informed Inspection Notebook, Revision 2.1a, for Susquehanna Steam Electric Station to evaluate the risk significance of the finding. The performance deficiency was evaluated to affect the main condenser, since a loss of the CW system would render the condenser unavailable as a heat sink. Using Table 2 of the SDP Phase 2 Notebook, the main condenser affected the power conversion system (PCS) steam cycle and required an evaluation of the SDP Phase 2 Worksheet for the "Transient with Loss of PCS (TPCS)" initiating event (i.e., Table 3.2).

The exposure time was taken to be 91 days (0.25 yr), since Unit 1 went critical on April 16, 2010 and the turbine building flooding event occurred on July 16, 2010. The exposure time corresponds to the time period that the condition being assessed was reasonably known to have existed (per the usage rules of IMC 0308, Attachment 3, Appendix A), which was from initial startup of Unit 1 from the last refueling outage. No condenser recovery credit as a heat sink was assumed in the evaluation.

Using the SDP Phase 2 Worksheet Table 3.2, Transient with Loss of PCS (TPCS), the initiating event likelihood (IEL) for the TPCS initiator was increased by one order of magnitude for the TPCS transient. This increase was based on IMC 0609, Appendix A, Attachment 2, "Site Specific Risk-Informed Inspection Notebook Usage Rules," Section 1.2, which states that if the increase in the frequency of an initiating event due to an inspection finding is not known, to increase the IEL for the applicable initiating event by one order of magnitude. Applying the above change to Table 3.2, resulted in a characterization of the finding as "White" in Phase 2 based on the change in core damage frequency. The two dominant sequences for core damage frequency involved a transient with loss of PCS initiating event with (1) loss of containment heat removal, failure of extended injection/CST makeup, and failure of late inventory, makeup; and (2) failure of RCIC and HPCI, and failure to depressurize the RPV.

Since the change in core damage frequency was greater than 1E-7/yr, the finding was then screened in Phase 2 for the potential risk contribution due to large early release frequency (LERF) using IMC 0609, Appendix H, "Containment Integrity Significance Determination Process." This resulted in a change in LERF of 9.9E-7/yr (White). The dominant sequence for change in LERF involved a transient with loss of PCS initiating event with failure of RCIC and HPCI, and failure to depressurize the RPV.

To evaluate whether the Phase 2 SDP evaluation was conservative, a Phase 3 SDP evaluation was performed. The SRA used the Standardized Plant Analysis Risk (SPAR) Model, Revision 3P (Change 3.45) for Susquehanna Steam Electric Station for the analysis.

For the Phase 3 SDP evaluation, the exposure time used was 91 days (0.25 yr) (i.e., the same as for the Phase 2 SDP evaluation). The following SPAR Model modifications were made:

- SPAR model basic event IE-loss of condenser heat sink (LOCHS), representing "Loss of Condenser Heat Sink" was set to a frequency corresponding to one event during the 91 days (i.e., to an initiating event frequency (IEF) of 4.0/yr for the 91 days). The basis for the IEF change was that the frequency of the TPCS event was assumed to be 1.0 over the 91 days (0.25 yr) of the performance deficiency.

  NOTE: This method of calculation is essentially equivalent to performing a conditional core damage probability (CCDP) for a LOCHS event and then subtracting the baseline core damage probability (CDP) (i.e., 0.2/year x 0.25 years = 0.05 baseline CDP for LOCHS) for a LOCHS event in the SPAR model (i.e., CCDP – CDP).

- In the SPAR model, on the anticipated transient without scram (ATWS) Event Tree, the "RUN BACK" top event was deleted to ensure that the correct cutsets were obtained (i.e., so that cutsets associated with failure to runback the turbine-driven reactor feedwater (RFW) pumps would not show up for a LOCHS event, in which the RFW pumps would not be available anyway).

The following influential assumptions were used:

- The IEF due to the performance deficiency was assumed to be a constant 4.0/yr for the 91 days of exposure time (i.e., there was a constant probability of failure of the condenser waterbox manways).

- Failure of the condenser waterbox manways would result in a loss of condenser heat sink initiating event.

- Nominal test and maintenance values were used.

The result was a total estimated change in core damage frequency of 1.1 E-6/yr (White). The two dominant core damage sequences involved a LOCHS initiating event, and (1) failure of RCIC and HPCI, and failure to manually depressurize the RPV; and (2) failure of suppression pool cooling (early), failure of containment spray, failure of PCS recovery, failure of containment venting, and failure of late injection.

The contributions to the risk estimates from external events (e.g., fire, flooding, and seismic) were determined to be low as discussed below.

For fires, no appreciable external risk contributions were identified. The Susquehanna IPEEE screens the turbine building (Fire Area T-1) as a risk contributor (reference Table 4.15, "Building Screening Criteria"), based on defense-in-depth, and that loss of the turbine building leaves all emergency core cooling system (ECCS) equipment functional.

Flooding scenarios were screened using IMC 0609, Appendix A, Table 3.1, "Plant Specific Flood Scenarios." The guidance lists SSCs important to internal flooding and it does not contain the main condenser.

The seismic risk contributions were screened using IMC 0609, Appendix A, Attachment 3, since the main condenser is not used to mitigate the consequences of a loss of the offsite AC power supply.

The SRAs used IMC 0609 Appendix H, "Containment Integrity Significance Determination Process" and NUREG-1765, "Basis Document for LERF SDP," to determine the potential risk contribution due to LERF. The finding was determined to be of Type "A" which is a finding that can influence CDF and also impact LERF.

Based on IMC 0609 Appendix H and NUREG-1765, for transient sequences, if the RCS is at high pressure at the time of core damage, the conditional probability is 0.3 that a Mark II containment will fail whether or not the drywell floor is flooded (i.e., the LERF Factor is 0.3 for high RCS pressure core damage sequences for a Mark II containment). For transient sequences with the RCS at low pressure at the time of core damage, the LERF Factor is zero. For ATWS sequences, the LERF Factor is 0.4 for a Mark II containment.

The dominant core damage sequences obtained with the SPAR analyses were separated into groups to reflect ATWS core damage sequences, high RCS pressure core damage sequences, and low RCS pressure core damage sequences. The result of the LERF Phase 3 analyses was a change in LERF of 2.6E-7/yr (White). The two dominant LERF sequences involved a LOCHS initiating event, and (1) failure of RCIC and HPCI, and failure to manually depressurize the RPV, and (2) failure of the reactor to scram (ATWS), a failure of the power conversion system, and failure to manually depressurize the RPV.

In summary, the conclusion of the Phase 3 analysis was an estimated change in core damage frequency of 1.1E-6/yr (White) and an estimated change in large early release frequency of 2.6E-7/yr (White).

The finding is related to the cross-cutting area of Problem Identification and Resolution, Corrective Acton Program, in that PPL did not thoroughly evaluate problems such that the resolutions addressed the causes and extent of condition for two of the three inadequate procedures. Specifically, PPL did not adequately: 1) evaluate previous CW system waterbox manway gasket leaks (April 2007 and March 2008) to ensure that future occurrences could be prevented; and 2) evaluate and correct a known issue in an off-normal procedure that complicated the operator's response to the event (November 2009). (P.1.(c))

Enforcement: NDAP-QA-0008, "Procedure Writer's Guide," Revision 8, specifies, in part, that off-normal procedures "specify operator actions for restoring an operating variable to normal operating conditions following a perturbation" and "such actions are invoked following an operator observation of an off-normal condition, which, if not corrected, could degenerate into a condition requiring action under an Emergency Procedure." Additionally, NDAP-QA-0008 stated that maintenance procedures "contain enough detail to permit the maintenance work to be performed correctly and safely."

Contrary to this requirement, three instances of inadequate procedures were identified. The first instance involved maintenance procedure MT-043-001, Revision 14 which provided inadequate instructions regarding installation of the condenser waterbox manway gaskets and led to the event. In the second instance, off-normal procedure ON-142-001, Revision 17, "CW Leak," did not have specific instructions on how to isolate a condenser waterbox leak. Finally, lacking specific guidance in ON-142-001, operators referred to ON-100-003, Revision 24, "Chemistry Anomaly," to identify and isolate the leaking manway. However, an incorrect diagram in off-normal procedure ON-100-003, led to complications in isolating the waterbox leakage. Operators subsequently decided to manually scram the reactor, shut the MSIVs, and isolate the condenser in order to secure the CW system and stop the leak. These issues are identified in the PPL's CAP as CRs 1285076, 1296863, and 1283470.

This finding does not involve enforcement action because no regulatory requirement was identified. Specifically, the circulating water system is not considered safety-related. Because this finding does not involve a violation but has preliminarily been determined to be of low to moderate safety significance (White), it is identified as **Preliminary White FIN 05000387/2010005-01, "Procedural Inadequacies Result in Reactor Scram and Loss of Normal Heat Sink."**

1R11 Licensed Operator Requalification Program

.1 Resident Inspector Quarterly Review (71111.11Q – 1 sample)

a. Inspection Scope

On July 6 and 9, 2010, the inspectors observed as-found licensed operator simulator performance. Specifically, the inspectors observed as found scenarios on July 6, 2010, and a remedial session July 9, 2010, for OP002-10-05-01A and OP002-10-05-01B. The inspectors compared their observations to Technical Specifications (TSs), emergency plan implementation, and the use of system operating procedures. The inspectors also evaluated PPL's critique of the operators' performance to identify discrepancies and deficiencies in operator training. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12 – 2 samples)

a. Inspection Scope

The inspectors evaluated PPL's work practices and followup corrective actions for selected SSC issues to assess the effectiveness of PPL's maintenance activities. The inspectors reviewed the performance history of those SSCs and assessed PPL's extent of condition determinations for those 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. Documents reviewed are listed in the Attachment. The following systems were reviewed:

- Unit 1, Standby Liquid Control (SBLC) valve performance; and
- Common, ESW check valves for 'E' EDG.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 4 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 Part 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 specified and appropriate risk management actions were identified.

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

- Unit 1, Yellow Risk during Division II Residual Heat Removal Service Water (RHRSW) SOW;
- Unit 1, Yellow Risk, Loss of T-10 due to switching error;
- Unit 2, maintenance risk assessment during SBLC flow verification; and
- Unit 2, HPCI inoperable due to leak on auxiliary oil filter.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15 – 5 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 TSs. In addition, the inspectors reviewed the selected operability determinations to evaluate whether the determinations were performed in accordance with NDAP-QA-0703, "Operability Assessments." The inspectors used the TSs, Technical Requirements Manual (TRM), Final Safety Analysis Report (FSAR), and associated Design Basis Documents as references during these reviews. Documents reviewed are listed in the Attachment. The issues reviewed included:

- Unit 1, core spray (CS) Division I loop discharge pressure;
- Unit 1, HPCI/RCIC inoperable due to pumping to CST berm;
- Unit 1, RCIC flow instabilities;
- Unit 2, HPCI exhaust line drain pot high level alarm; and
- Common, "E" EDG minimum frequency during LOOP/LOCA event scenario.

b. Findings

.1 Transfer of Water from Condenser Area to CST Berm Submerged HPCI and RCIC CST Low Level Suction Transfer Instrumentation

Introduction: The inspectors identified a Green NCV of Susquehanna Unit 1, TS 5.4.1, "Procedures," for an inadequate procedure to transfer water from the condenser area to the condensate storage tank (CST) berm. Specifically, the procedure failed to include a maximum level in the CST berm that was acceptable to limit interactions with other safety-related equipment. The NCV was identified following the July 16, 2010, Unit 1 manual reactor scram due to a non-isolable circulating water leak in the main condenser area. Operations personnel commenced dewatering efforts by transferring water from the condenser area to the CST berm using a "Liquid Radwaste Collection" operating procedure as a guide. Water was transferred to the berm to a level sufficient to cause water intrusion into cable conduit and junction boxes containing High Pressure Coolant Injection system (HPCI) and Reactor Coolant Isolation Cooling system (RCIC) CST low-level suction instrumentation which transfers HPCI and RCIC pump suction from the CST to the suppression pool. As a result, the low-level suction instrumentation became submerged affecting the reliability and capability of the HPCI and RCIC CST to suppression pool transfer function despite being required in Mode 3. The issue was entered into PPL's CAP (1297039).

Description: On July 16, 2010, the Unit 1 reactor was manually scrammed due to a large non-isolable circulating water (CW) leak in the main condenser area. The leak resulted in approximately 1,000,000 gallons of water entering the main condenser area. In an attempt to dewater the turbine building, operations personnel transferred water from the condenser area to the CST berm using temporary pumping equipment, while using an existing procedure, "Liquid Radwaste Collection," OP-169-004, Revision 1, as a guide. This procedure provided no guidance as to a maximum level that should be transferred to the berm to limit interactions with other safety- related equipment.

On July 17, 2010, the inspectors informed PPL that water was entering the buildings housing the 'B' and 'D' Emergency Diesel Generators through conduit and a junction box. The licensee determined that the junction box and associated conduit contained instrumentation cabling associated with suction transfer of HPCI and RCIC from the CST to the suppression pool. The prompt operability determination performed by operations personnel stated "the leaks do not appear to affect the diesel generators or HPCI and RCIC" and thus declared the systems operable, and an Engineering follow-up was requested. The Operability Follow-up Request (OFR) performed by engineering initially determined that the level switches were not impacted since it did not appear the switches were submerged. This supported operations' initial prompt operability determination. This assessment did not take into account the water intrusion in the junction box in the EDG building. It was later determined that the switches and associated unscheduled junction boxes in the berm had been submerged when water was transferred from the condenser area. These switches and associated junction boxes are not qualified for submergence. On July 20, 2010, RCIC swapped its suction from the CST to the suppression pool, with CST level at 24% (CR 1283258). Transfer should occur at a level of 7.5%. The OFR was updated to state that "the design of the switches is such that water intrusion via the conduit entrance to the switch could render the switch unreliable." At the time of the unintended swap, the plant had reduced pressure to the point that neither HPCI nor RCIC were required by plant TSs, so no further action was required.

Though non-safety related, the CST is the preferred source of water for both HPCI and RCIC operation. The CST provides an approximate capacity of 300,000 gallons of water for use by HPCI and RCIC, with an additional 680,000 gallons of water inventory available to be transferred to the CST from the Refueling Water Storage Tank. It was subsequently determined that the short-term failure mechanism of the level switches controlling the TS required function was a simulated low-level condition. Though this failure mode is a fail-safe condition, and ensures the Technical Specification safety function is preserved, it would render the preferred source of water for high pressure injection unavailable. Additionally, when the suction swap occurs on the HPCI system, operators are unable to override the signal to restore the suction to the CST.

During review of the event, the inspectors noted that an Engineering Work Request completed in 1992 following a similar condenser area flooding event evaluated the maximum level that the CST berm could be filled as approximately 6.8 feet. This level was based on meeting the UFSAR design basis for the CST berm of retaining water from a simultaneous rupture of the CST and RWST, without releasing any water to the environment. However, it cautioned that an engineering analysis documented that the functioning of HPCI and RCIC would be compromised if water level were allowed to submerge the associated level switches. These switches are located at a height of 6.5 feet. Though not implemented, the work request recommended that the procedure used to transfer water to the CST berm be updated to include a maximum level and a precaution that it should only be done as a last resort.

Analysis: Failure to have an adequate procedure for transferring water from the condenser area to the CST berm to limit interactions with other safety-related equipment is a performance deficiency which was reasonably within PPL's ability to foresee and correct. The finding was not subject to traditional enforcement because there were no actual consequences, it was not willful, and did not impact the NRC's ability to regulate. This issue is more than minor as it affected the equipment performance and procedural quality attributes of the corresponding Mitigating Systems cornerstone objective to ensure the reliability and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the low-level suction instrumentation was not designed for submergence. Transferring too much water from the condenser bay to the CST berm submerged the low-level suction instrumentation and affected the reliability and capability of the HPCI and RCIC CST to suppression pool transfer function. The finding was evaluated for significance using IMC 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings." Since the finding did not result in a loss of safety function or the loss of a train for greater than its TS allowed outage time, and was not potentially risk significant due to external event initiators, the finding was determined to be of very low safety significance (Green).

This finding was determined to have a cross-cutting aspect in the area of Human Performance, Resources, because PPL did not ensure that procedures were adequate to assure nuclear safety. Specifically, operating procedure OP-169-004, Revision 17, did not specify a maximum level that could be transferred to the CST berm to limit interactions with safety-related, HPCI and RCIC low-level suction transfer instrumentation. (H.2(c))

Enforcement: Susquehanna Unit 1 TS 5.4.1, "Procedures," requires that written procedures be established, implemented and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A requires procedures for operation of the liquid radioactive waste system. Contrary to the above, on July 17, 2010, safety-related level switches that controlled the suction transfer of both the HPCI and RCIC systems from the CST to the suppression pool were submerged when operators transferred water to the CST berm in accordance with procedure OP-169-004, Revision 17. Because this finding is of very low safety significance and has been entered into PPL's corrective action program (CR 1297039), this violation is being treated as an NCV consistent with section 2.3.2 of the NRC's

Enforcement Policy. **(NCV 05000387/2010004-02, Transfer of Water from Condenser Area to CST Berm Submerged HPCI and RCIC CST Low Level Suction Transfer Instrumentation.)**

.2 Failure to Accurately Model the Simulator for RCIC System Operation at Reduced Flow Rates in Automatic

Introduction: An NRC-identified, Green NCV of 10 CFR 55.46(c)(1), "Plant Referenced Simulators," was identified because the Susquehanna simulator did not accurately model RCIC system response when operated in automatic flow control at less than design basis full flow. While the licensee has not yet completed simulator modifications to routinely model RCIC control system instabilities when operating the system in automatic flow control at less than design basis full flow, the simulator does model instabilities resulting from a control system malfunction. The inspectors verified that licensed operators have trained on and responded to RCIC control system malfunctions during examinations. This issue was entered in PPL's corrective action process as CRs 1285503, 1287462, and 1286803.

Description: Following the July 16, 2010, reactor scram, PPL identified that RCIC system operation was unstable when attempting to operate the system in automatic flow control with the RCIC flow controller set below the design flow rate. PPL initiated EWR 1285503 to evaluate the system instabilities and any potential operability impact on the RCIC system. PPL documented that operation of the RCIC system below design flow while in automatic flow control was the most likely cause for the flow instabilities. PPL referenced the current EPRI Terry Turbine Maintenance Guide, "RCIC Application," Section 22.3, which identifies that operating the RCIC system at less than 75 percent of design flow was likely to cause system instabilities. PPL also identified a previous test performed by another licensee which allowed RCIC to inject to the RPV and confirmed that instability occurred as described in the EPRI Maintenance Guide.

The inspectors reviewed the evaluation provided in EWR 1285503 and determined that the EWR response was appropriate and acceptable. The inspectors determined that the examples cited within the EWR were appropriate to be applied to Susquehanna due to the similarities in design and application between PPL and other boiling water reactors (BWRs) which have a RCIC system installed. The inspectors noted that the guidance provided in the EPRI Maintenance Guide had been publically available since 2002. Interviews with PPL personnel identified that the instabilities in the RCIC system described in the EPRI Maintenance Guide and experienced by separate licensees was not previously known by PPL until EWR 1285503 was developed.

The incorrect modeling of the Susquehanna plant referenced simulator introduced negative operator training that could affect the ability of the operators to take the appropriate actions during an actual event. Specifically, the simulator training conditioned the operators to expect RCIC system operation to be stable at all selected flow rates when operated in automatic. As a result, during an actual event, the operators could misdiagnose the cause or means to correct unstable RCIC operation and eliminate an injection system to the RPV unnecessarily.

The inspectors then reviewed the simulator modeling of RCIC operation to determine if operator training was adequate to address control system instabilities while controlling the RCIC system in automatic at reduced flow rates. The Susquehanna simulator demonstrated stable operation at all flow rates when injecting to the reactor vessel in automatic. The inspectors determined that PPL did have instability of the RCIC system available as a failure mode of the system and used it periodically during training. Additionally, the inspectors reviewed the procedures used during RCIC operation and determined that the potential for unstable operation when RCIC is operated in automatic at reduced flow rates was not mentioned within the procedures nor was guidance provided to address the system instabilities if they occurred.

Corrective actions included, but were not limited to, the revision of procedures OP-1(2)50-001, "RCIC System," to identify that flow instability may occur at reduced flow rates and provide direction to place the flow controller in Manual mode if flow becomes unstable. The inspectors concluded that PPL did not ensure that the plant referenced simulator accurately modeled the expected plant response for RCIC operation in automatic at less than design flow rates, resulting in negative training of the licensed operators.

The RCIC flow instability was identified on July 16, 2010, and entered in the corrective action process under EWR 1285503. The evaluations developed within the EWR only addressed RCIC system operability and recommended testing to be performed due to extended system operation. Although PPL correctly characterized the RCIC system instabilities as potentially occurring any time the system is operating in automatic with flow set less than 75 percent of the design flow rate; PPL failed to address the differences between the actual plant response and the modeling of the simulator. Since the inspector identified the impact of this issue on the simulator this finding was characterized as NRC-identified.

Analysis: The inspectors determined that failing to model the simulator for RCIC instabilities at low flow rates is a performance deficiency that was within PPL's ability to foresee and correct. This finding is more than minor because it is associated with the Human Performance attribute of the Mitigating Systems cornerstone and affects the objective to ensure the availability, reliability, and capability of the RCIC system to respond to initiating events to prevent undesirable consequences. Specifically, the incorrect modeling of the Susquehanna plant referenced simulator introduced negative operator training that could affect the ability of the operators to take the appropriate actions during an actual event. The simulator training conditioned the operators to expect RCIC system operation to be stable at all selected flow rates when operated in automatic. As a result, during an actual event, the operators could misdiagnose the cause or means to correct unstable RCIC operation and eliminate an injection system to the RPV unnecessarily. The inspectors evaluated the finding in accordance with IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process." The finding was determined to be of very low safety significance (Green) because it is not related to operator performance during requalification, it is related to simulator fidelity, and it could have a negative impact on operator actions.

This issue was determined to not have a cross-cutting aspect. This was based on the age of the EPRI guidance (issued in 2002) applicable to the RCIC system flow instabilities and the lack of opportunities over the past three years to revisit this guidance. Therefore, this issue was not reflective of current performance.

Enforcement: 10 CFR 55.46(c)(1), "Plant Referenced Simulators," states, in part, that "a plant referenced simulator... must demonstrate expected plant response to... normal, transient, and accident conditions." Contrary to this, from 2002 until August 2010, the Susquehanna plant referenced simulator did not accurately demonstrate the expected plant response of the RCIC system in automatic flow control when operating at reduced flow rates, which could result in negative operator training. However, because the finding was of very low safety significance (Green) and has been entered into the CAP (ARs 1285503, 1287462 and 1286803), this violation is being treated as an NCV, consistent with section 2.3.2 of the NRC's Enforcement Policy. **(NCV 05000387/2010004-03; 05000388/2010004-03 – Failure to Accurately Model the Simulator for RCIC System Operation at Reduced Flow Rates in Automatic)**

1R18 Plant Modifications

.1 Temporary Plant Modifications (71111.18 – 1 sample)

a. Inspection Scope

The inspectors reviewed a temporary plant modification to determine whether the changes adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the FSAR, TSs, and assessed the adequacy of the safety determination screening and evaluation. The inspectors also assessed configuration control of the changes by reviewing selected drawings and procedures to verify that appropriate updates had been made. The inspectors compared the actual installation to the modification documents to determine whether the implemented change was consistent with the approved documents. The inspectors reviewed selected post-installation or removal test results as appropriate to evaluate whether the actual impact of the change or removal had been adequately demonstrated by the test. The following modification and document was included in the review:

- Unit 1, pumping water to cooling tower blowdown line.

b. Findings

No findings were identified.

.2 Permanent Plant Modifications (2 samples)

a. Inspection Scope

The inspectors reviewed the following two permanent plant modifications to determine whether the changes adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the FSAR, TSs, and assessed the adequacy of the safety determination screenings and evaluations. The inspectors also assessed configuration control of the changes by reviewing selected drawings and procedures to verify whether appropriate updates had been made. The inspectors compared the actual installations to the permanent modification documents to determine whether the implemented changes were consistent with the approved documents. The inspectors reviewed selected post-installation test results to evaluate whether the actual impact of the changes had been adequately demonstrated by the test. Documents reviewed are listed in the Attachment. The following modification and document were included in the review:

- Common, 4kV vacuum breakers; and

- Common, TS Bases change removes ventilation requirements for refuel exhaust radiation monitors.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 5 samples)

a. Inspection Scope

The inspectors observed portions of post-maintenance test (PMT) activities in the field to determine whether the tests were performed in accordance with the approved procedures. The inspectors assessed the test adequacy by comparing the test methodology to the scope of maintenance work performed. In addition, the inspectors evaluated acceptance criteria to determine whether the test demonstrated that components satisfied the applicable design and licensing bases and TS requirements. The inspectors reviewed the recorded test data to determine whether the acceptance criteria were satisfied. The following tests were reviewed:

- Unit 1, quarterly calibrations of HPCI and RCIC CST level switches following berm flood;
- Unit 2, 2B control rod drive (CRD) pump logic system functional after a 2 year preventive maintenance activity;
- Unit 2, HPCI stop valve;
- Unit 2, "A" loop CS; and
- Common, "D" EDG after five year overhaul.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20 – 1 sample)

.1 Unit 1 Forced Outage

a. Inspection Scope

A Unit 1 forced outage was conducted from July 16 to August 2, 2010 following condenser bay flooding. During the outage and through reactor startup, as appropriate, inspectors performed the activities below to verify PPL's controls over outage activities:

- Shutdown activities;
- Outage activity control – monitored or verified the following:
  1) Clearance activities;
  2) RCS instrumentation;
  3) Electrical power;
  4) Decay heat removal;
  5) Reactivity control;
  6) Fatigue management;
- Monitoring of plant heatup and startup activities;

- Identification and Resolution of Problems – reviewed CAP entries to verify an adequate threshold of issues and appropriate corrective actions.

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 4 routine surveillance samples)

a. Inspection Scope

The inspectors observed portions of selected surveillance test activities in the control room and in the field and reviewed test data results. The inspectors compared the test results to the established acceptance criteria and the applicable TS or TRM operability and surveillance requirements to evaluate whether the systems were capable of performing their intended safety functions. The observed or reviewed surveillance tests included:

- Unit 1, RCIC flow verification at ≤ 165 psig;
- Unit 2, HPCI comprehensive flow verification;
- Unit 2, 24 month logic system functional test (LSFT) for 2A RB chiller (LOOP/LOCA application); and
- Common, "B" CS CW flow verification and "B" control room emergency outside air supply (CREOAS) operability test.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed the combined functional drill scenario and observed selected portions of the drill in the emergency operations facility. The inspection focused on PPL's ability to properly conduct emergency action level (EAL) classification, notification, and protective action recommendation activities and on the evaluators' ability to identify observed weaknesses and/or deficiencies within these areas. Eight performance indicator (PI) opportunities were included in the scenario.

The inspectors attended the post-drill critique and compared identified weaknesses and deficiencies including missed PI opportunities against those identified by PPL to determine whether PPL was properly identifying weaknesses and failures in these areas. The drill evaluation sample included:

- Common, EP Drill (Blue Team), August 24, 2010.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational/Public Radiation Safety (PS)

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01 – 1 sample)

a. Inspection Scope

The inspectors reviewed the current status of the investigation into the cause of the water that collected inside the steam dryer storage facility (CR 1266343). The inspectors reviewed drawings of the facility, immediate corrective actions taken, and PPL's future plan of action to identify and address the root cause(s) of the event. The inspectors reviewed available data, including the facility design/construction and the data from sampling of the nearest monitoring well (MW-6), to verify that there appears to be no evidence that any water escaped the storage facility and entered the groundwater.

Inspector Planning

The inspectors reviewed PPL Performance Indicators for the Occupational Exposure Cornerstone for followup. The inspectors reviewed the results of radiation protection program audits (e.g., PPL's quality assurance audits or other independent audits). The inspectors reviewed reports of operational occurrences related to occupational radiation safety since the last inspection.

Instructions to Workers

The inspectors selected containers holding nonexempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and verified that they were labeled and controlled. For this activity, the inspectors reviewed a selection of outage equipment storage containers located on the turbine deck.

The inspectors reviewed radiation work permits (RWPs) used to access high radiation areas (HRAs) and identify what work control instructions or control barriers had been specified. The inspectors verified that allowable stay times or permissible dose for radiologically significant work under each RWP was clearly identified. The inspectors verified that electronic personal dosimeter (EPD) alarm set points were in conformance with survey indications and plant policy.

The inspectors selected occurrences where a worker's EPD noticeably malfunctioned or alarmed. The inspectors verified that workers responded appropriately to the off-normal condition. The inspectors verified that the issue was included in the CAP and dose evaluations were conducted as appropriate.

Contamination and Radioactive Material Control

The inspectors observed several locations where PPL monitors potentially contaminated material leaving the RCA, and inspected the methods used for control, survey, and release from these areas. The inspectors verified that the radiation monitoring instrumentation had appropriate sensitivity for the type(s) of radiation present. The inspectors used a sampling of instrumentation located at the Unit 1, Unit 2, and control building access points.

The inspectors reviewed PPL's criteria for the survey and release of potentially contaminated material. The inspectors verified that there was guidance on how to respond to an alarm that indicated the presence of licensed radioactive material.

The inspectors reviewed PPL's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters.

The inspectors selected a sample of sealed sources from PPL's inventory records that presented the greatest radiological risk. The inspectors verified that sources are accounted for and had been verified to be intact.

The inspectors verified that any transactions involving nationally tracked sources were reported in accordance with 10 CFR 20.2207. No transactions of this type have occurred in 2010.

<u>Risk-Significant High Radiation Area and Very High Radiation Area Controls</u>

The inspectors verified that PPL controls for all very high radiation areas (VHRA), and areas with the potential to become a VHRA, ensured that an individual is not able to gain unauthorized access to the VHRA. PPL accomplished this via a combination of procedures and key controls.

The inspectors discussed with the Radiation Protection Manager the controls and procedures for high-risk HRAs and VHRAs. The inspectors verified that any changes to PPL procedures did not substantially reduce the effectiveness and level of worker protection.

The inspectors discussed with first-line health physics supervisors the controls in place for special areas that have the potential to become VHRAs during certain plant operations.

<u>Problem Identification and Resolution</u>

The inspectors verified that problems associated with radiation monitoring and exposure control were being identified by PPL at an appropriate threshold and were properly addressed for resolution in PPL's CAP. In addition to the above, the inspectors verified the appropriateness of the corrective actions for a selected sample of problems documented by PPL that involve radiation monitoring and exposure controls. The inspectors determined that PPL was assessing the applicability of operating experience to their plants.

b. <u>Findings</u>

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

Verification of Dose Estimates and Exposure Tracking Systems

The inspectors selected as low as is reasonably achievable (ALARA) work packages and reviewed the assumptions and basis for the current annual collective exposure estimate for reasonable accuracy. The inspectors reviewed the applicable procedures to determine the methodology for estimating exposures from specific work activities and the intended dose outcome.

The inspectors verified that for the selected work activities that PPL had established measures to track, trend, and if necessary to reduce, occupational doses for ongoing work activities. The inspector verified that trigger points or criteria were established to prompt additional reviews and/or additional ALARA planning and controls.

The inspectors reviewed the results of the spring, 2010 Unit 1 refueling outage, and compared the dose estimates with the actual doses received.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope

Inspection Planning

The inspectors reviewed the plant final safety analysis report to identify areas of the plant designed as potential airborne radiation areas and any associated ventilation systems or airborne monitoring instrumentation. The inspectors reviewed the reported PIs to identify any related or unintended dose resulting from intakes of radioactive materials.

Engineering Controls

The inspectors verified that PPL used ventilation systems as part of its engineering controls, in lieu of respiratory protection devices, to control airborne radioactivity. The inspectors reviewed procedural guidance for use of installed plant systems, and verified that the systems were used, to the extent practicable, during high-risk activities. The inspectors selected installed ventilation systems used to mitigate the potential for airborne radioactivity, and verified that ventilation airflow capacity, flow path, and filter/charcoal unit efficiencies were consistent with maintaining concentrations of airborne radioactivity in work areas below the concentrations of an airborne area to the extent practicable.

The inspectors selected temporary ventilation system setups high-efficiency particulate air used to support work in contaminated areas. The inspectors verified that the use of these systems was consistent with PPL procedural guidance and ALARA.

The inspectors selected installed systems to monitor and warn of changing airborne concentrations in the plant. The inspectors verified that alarms and set-points were sufficient to prompt PPL/worker action to ensure that doses were maintained within the limits of 10 CFR Part 20 and ALARA. The inspectors verified that PPL had established trigger points for evaluating levels of airborne beta-emitting and alpha-emitting radionuclides.

Problem Identification and Resolution

The inspectors verified that problems associated with the control and mitigation of in-plant airborne radioactivity were being identified by PPL at an appropriate threshold and were properly addressed for resolution in PPL's CAP.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation (71124.05)

a. Inspection Scope

Walkdowns and Observations

The inspectors walked down area radiation monitors and continuous air monitors to determine if they were appropriately positioned relative to the radiation source(s) or area(s) they were intended to monitor. The inspectors selectively compared monitor response with actual area conditions for consistency.

The inspectors selected personnel contamination monitors, portal monitors, and small article monitors, and verified that the periodic source checks were performed in accordance with the manufacturer's recommendations and PPL's procedures. The inspectors reviewed source checks performed on these instruments during the month of June 2010.

Whole Body Counter (WBC)

The inspectors reviewed the methods and sources used to perform WBC functional checks before daily use of the instrument. The inspectors reviewed the control charts for the first quarter of 2010. The inspectors determined that check sources were appropriate and aligned with the plant's isotopic mix.

The inspectors reviewed WBC calibration reports completed since the last inspection to verify that calibration sources were representative of the plant source term and that appropriate calibration phantoms were used.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Mitigating Systems (71151 - 4 samples)

a. Inspection Scope

The inspectors reviewed PPL's PI data for the period of November 2009 through July 2010 to determine whether the PI data was accurate and complete. The inspectors examined selected samples of PI data, PI data summary reports, and plant records. The inspectors compared the PI data against the guidance contained in Nuclear Energy Institute (NEI) 99-02, Revision 5, "Regulatory Assessment Performance Indicator Guideline." The following performance indicators were included in this review:

- Units 1 and 2, Emergency Alternating Current (AC) Power Systems, MS06; and
- Units 1 and 2, Cooling Water Systems, MS10.

b. Findings

No findings were identified.

.2 Occupational Radiation Safety (1 sample)

a. Inspection Scope

The inspectors reviewed all of PPL's PIs for the Occupational Exposure Cornerstone (OR01) for follow-up. The inspectors reviewed a listing of PPL's action reports for the period December 1, 2009 through August 27, 2010 for issues related to the occupational radiation safety performance indicator, which measures non-conformances with HRA greater than 1R/hr and unplanned personnel exposures greater than 100 mrem total effective dose equivalent (TEDE), 5 rem skin dose equivalent (SDE), 1.5 rem lens dose equivalent (LDE), or 100 mrem to the unborn child.

The inspectors determined if any of these PI events involved dose rates >25 R/hr at 30 centimeters or >500 R/hr at 1 meter. If so, the inspectors determined what barriers had failed and if there were any barriers left to prevent personnel access. For unintended exposures >100 mrem TEDE (or >5 rem SDE or >1.5 rem LDE), the inspectors determined if there were any overexposures or substantial potential for overexposure. The inspectors determined that no PI events had occurred during the assessment period.

b. Findings

No significant findings or observations were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered into the CAP

a. Inspection Scope

As specified by Inspection Procedure (IP) 71152, "Identification and Resolution of Problems," (PI&R), and in order to help identify risk significant, repetitive, long-term or latent equipment failures, cross-cutting components or adverse performance trends for followup, the inspectors performed screening of all items entered into PPL's CAP. This was accomplished by reviewing the description of each new CR, attending management committee meetings, and viewing computerized CAP entries.

b. Findings

No findings were identified.

.2 Annual Sample: ICS Performance (1 sample)

a. Inspection Scope

In response to repetitive issues regarding the ICS during plant transients on April 22 and May 14, 2010, an annual sample was conducted to examine the modification's performance and compare it to expected plant response as well as design. Additionally, inspectors reviewed operator actions and responses as well as procedural guidance for the operation of the ICS during transient response situations. As specified by IP 71152, and in order to help identify specific human performance issues for follow-up, the inspectors reviewed several analyses and evaluations entered into PPL's CAP for the failures of the ICS to perform as expected.

b. Findings

Introduction: A self-revealing NCV of 10 CFR 55.46(c)(1), "Plant Referenced Simulators," was identified because the Susquehanna simulator did not accurately model integrated control system (ICS) response to reactor pressure vessel (RPV) level transients. This violation was due to an error in the simulator modeling that caused RPV level control in the simulator to respond more rapidly than the actual plant resulting in the simulation of a more stable response and smaller overall changes in RPV level during level transients in the simulator. This error contributed to the decision to proceed with an extended power uprate (EPU) required condensate pump trip test during reactor power ascension activities. As a result on May 14, 2010, when the condensate pump trip test was performed, the ICS system was unable to adequately control reactor vessel water level and operators inserted a manual reactor scram prior to a high level turbine trip at level 8. PPL completed corrective actions to update the simulator model to accurately reflect the feedwater flow component of ICS and has ensured that the simulator reflects actual plant performance and re-performed the condensate pump trip test. This issue was entered in PPL's corrective action process as AR/CR 1257781.

Description: As part of the team's followup on the issues in CR 1257781, the inspectors reviewed the root cause analysis developed to determine the cause of the reactor scrams, on April 22 and May 14, during testing of the ICS following the 1R16 refueling outage. The licensee determined that the root cause of the RPV level transients and subsequent reactor scrams was less than adequate engineering rigor that allowed inadequate/incorrect gains and tuning factors to be developed as part of the ICS modification.

The root cause analysis identified that a major contributor to the incorrect gain setting/tuning parameter in the plant was an error in the initial simulator programming for the ICS modification. Specifically, the vendor responsible for providing the simulator modeling had incorrectly programmed the MFWLC which resulted in an output gain approximately five times higher than it should have been. This error made the simulator more responsive to large RPV level transients than the actual plant. The modeling information provided by the simulator vendor was not reviewed by PPL nor was an alternate methodology used to validate simulator performance prior to use in operator training and predictions of actual plant response.

The inspectors found the root cause analysis to be adequate to address the reasons why the ICS had failed to respond as expected during start-up testing. The inspectors agreed with PPL's assessment that the simulator had failed to model the ICS system correctly since the initial installation of the modeling information on PPL's simulator approximately twelve months prior to finding the error. The inspectors also agreed with PPL's assessment that the simulator failed to model the actual plant during transient conditions because of the error introduced in the MFWLC programming. Additionally, the inspectors agreed with the conclusion that PPL's failure to validate the changes to simulator modeling and performance had allowed the error to exist until the failure resulted in additional reviews of the simulator by PPL. The inspectors identified that the incorrect simulator modeling had been used for licensed operator training prior to the 1R16 refueling outage and was used as an input into PPL's decision to proceed with a test of ICS which led to an RPV level transient and a manual reactor scram. PPL completed corrective actions to update the simulator model to accurately reflect the MFWLC programming in the simulator and has verified that the simulator reflects actual plant performance with the updated gains and tuning factors.

The inspectors also noted that following the April 22 scram during ICS testing from ~33% reactor power, it was identified that the ICS system had been slow to respond to error signals during the special test procedure. One of the corrective actions was to increase the ICS gains by a factor of 3 to ensure the system responded more quickly. This gain error was approximately the same magnitude as the gain error in the MFWLC circuit. Thus PPL missed a potential opportunity to identify and correct this error prior to the May 14 scram.

Analysis: The inspectors determined that PPL's inability to accurately model the performance of the ICS in the simulator is a performance deficiency that was within PPL's ability to foresee and correct. The finding is more than minor because it is associated with the Human Performance attribute of Mitigating Systems and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, since the simulator model did not reflect actual plant performance, the Susquehanna simulator introduced negative operator training that affected the ability of the operators to take the appropriate and timely actions during an actual event to prevent a plant scram. This inadequate simulation also influenced PPL's decision to proceed with the condensate pump trip test on May 14, 2010, since the simulator predicted the ICS would be able to satisfactorily respond to the loss of a condensate pump from full reactor power. The inspectors evaluated the finding in accordance with IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process." The finding was determined to be of very low safety significance (Green) because it is not related to operator performance during requalification, it is related to simulator fidelity, and it could have a negative impact on operator actions.

This finding has a cross-cutting aspect in the area of Human Performance, Resources, because PPL did not ensure that equipment and other resources were available and adequate to assure safety. Specifically, simulator fidelity was inadequate in that modeling information provided by the simulator vendor was not reviewed by PPL nor was an alternate methodology used to validate simulator performance prior to use in operator training and predictions of actual plant response. In addition, ICS adjustments made after the April 22, 2010, scram provided another opportunity to verify the validity of ICS gain settings. (H.2(d))

Enforcement: 10 CFR 55.46(c)(1), "Plant Referenced Simulators," states, in part, that "a plant referenced simulator... must demonstrate expected plant response to... normal, transient, and accident conditions." Contrary to this, from April through May 14, 2010, the Susquehanna plant referenced simulator did not accurately demonstrate the actual expected plant response of the ICS during level transients, which resulted in negative operator training. However, because of the very low safety significance (Green) and because it has been entered into the CAP (CR 1257781), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC's Enforcement Policy. **(NCV 05000387/2010004-04, Failure to Accurately Model the Simulator for RPV Level Control Using the Integrated Control System)**

.3 Annual Sample: Operator Workarounds (OWAs) (1 sample)

a. Inspection Scope

As required by IP 71152, "Identification and Resolution of Problems," the inspectors conducted a review of the OWA program to verify that PPL was identifying OWAs at an appropriate threshold, was entering them into the CAP and proposing or implementing appropriate corrective actions. Specifically, the review was conducted to determine if any OWAs for mitigating systems affected their safety functions or affected operators' abilities to implement abnormal or emergency operating procedures (EOPs). The inspectors also interviewed nuclear plant operators and reviewed operator rounds and logs to determine whether routine compensatory actions should be categorized as OWAs or operator challenges.

b. Findings and Observations

No findings of significance.

The inspectors determined that PPL was implementing their OWA program appropriately. The program identified issues and deficiencies which may present a challenge for the operators to complete their required actions during a plant event. The program appropriately evaluated the risk of each individual challenge and the overall plant risk due to all challenges. Corrective actions for these challenges were developed and scheduled with respect to their risk significance.

4OA3 Event Follow-up (71153 – 4 samples)

.1 Condenser Bay Flooding and Manual Scram

a. Inspection Scope

On July 16, 2010, at approximately 3:20 pm, Unit 1 received a condenser bay flood alarm. Plant operators verified that flooding was occurring into the 656' elevation of the condenser bay. Reactor power was reduced to 40 percent RTP via control rod insertions and a recirculation runback. Operator attempts to isolate condenser waterboxes remotely were unsuccessful. Unit 1 was subsequently manually scrammed, Main Steam Isolation Valves (MSIVs) were shut, and the main condenser was isolated so that the CW system could be shutdown. Concurrently, plant operators manually closed waterbox isolation valves and isolated the leak.

Plant response following the manual reactor scram was not as expected. The ICS FWLC is designed to switch to single element (1E) control on low main steam flow. Due to steam condensation and flashing on the flow instrument, measured main steam flow remained above the transition point and ICS FWLC remained in three element (3E) control. The effect of this was that while the 'B' and 'C' feedwater pumps automatically switched to the idle mode and level setpoint setdown occurred as expected, the 'A' feedwater pump underwent demand oscillations prior to its transition to Discharge Pressure mode. Inventory continued to be added to the reactor vessel until level reached the high level turbine trip setpoint and peaked at 55". Exceeding the setpoint resulted in a trip of all feedwater pump turbines, the HPCI turbine, the RCIC turbine and the main turbine. It took approximately 14 minutes for reactor vessel water level to steam down less than the trip setpoint. Once level was restored below the setpoint, the MSIVs were shut and HPCI and RCIC were manually initiated for pressure and level control respectively.

A resident inspector responded to the control room and observed the plant's response to the transient and associated operator actions during the response. An estimated 1,000,000 gallons of CW entered the condenser bay reaching a maximum height of 12' in the area. This water was pumped to multiple, 21,000 gallon tanks brought onsite that were subsequently radiologically tested prior to discharge via the cooling tower effluent blowdown line to the Susquehanna river. Some of the water was pumped to the Unit 1 CST berm. The water in the CST berm was discovered by the resident inspectors to have made its way via RCIC/HPCI CST level instrumentation conduits into the 'B' EDG basement via a junction box.

The inspectors reviewed the transient response post-event to include operability determinations for the HPCI, RCIC, and EDG due to water intrusion and wetting of system components, the EAL classifications made and the EAL matrix bases used, and the response of the ICS to the reactor scram and corrective actions taken by PPL to correct the identified failure of the ICS to perform as expected. Additionally, the inspectors reviewed operator response to the CW leak, the subsequent manual reactor scram, and the failure of ICS to adequately control RPV water level after the scram to include adequacy of training provided for ICS. Finally, the inspectors reviewed the fidelity of the simulator as it applied to operator training on ICS and its modeling of the expected plant response after the reactor scram.

b. Findings

Three findings in this report are related to this event. These findings included: the preliminary white finding for procedural inadequacies which caused and complicated the July 16, 2010 flooding event, (Section 1R06); a Green NCV for an inadequate procedure which allowed safety-related instrumentation for the HPCI and RCIC systems to be submerged, a condition for which the instrumentation was not designed (Section 1R15.1); and a Green NCV for failure to accurately model the simulator for RCIC system operation at reduced flow rates while in automatic (Section 1R15.2).

.2 Feedwater Level Control and Control Rod Mispositioning Human Performance Errors

a. Inspection Scope

On August 4, 2010, two human performance events occurred during a Unit 1 reactor startup. With the plant in Mode 2, a plant control operator (PCO) switched the operating reactor feed pump from discharge pressure control mode to flow control mode. The unit experienced a level transient and reactor level dropped from 35 inches to approximately 23 inches. The PCO took manual control of the reactor feed pump and restored level to the normal operating band. A review of ICS revealed that the automatic setpoint that was stored for flow control mode was 25.5 inches instead of the desired 35 inches. PPL determined that the setpoint had been changed during the July 16, 2010 flooding event and had not been restored prior to reactor startup. Another factor that contributed to this event was that the PCO did not verify the automatic setpoint prior to changing operating modes. In the second instance, with the plant at 32 percent RTP, a control rod was withdrawn from step 12, past the desired step 24, to step 36 during power ascension. PCOs identified the performance error prior to executing the next procedural step.

b. Findings

A Green NCV of TS 5.4.1, “Procedures,” was licensee-identified for failure to comply with a procedure for reactivity control and is documented in section 4OA7 of this report.

.3 Alert Declared on 1A Reactor Building Chiller Freon Leak

a. Inspection Scope

On August 10, 2010, the 1A RB chiller tripped and the 1B RB chiller automatically started. When plant operators went to verify the trip condition and normal status of the respective chillers, they identified a Freon leak on the 1A chiller. The leak continued despite closure of valves on the associated line. After a discussion in the control room, operators, maintenance technicians, and site safety representatives returned to the chiller area to assess the status of the leak. The maintenance technicians noted that the leak was still active and evacuated the area. After reporting to the control room, one of the technicians, a qualified Freon handler, reported to operations personnel that he felt ill from the effects of the Freon gas and had evacuated the area. An ALERT was declared at 9:22 am. PPL staffed their Technical Support Center and Emergency Operations Facility, and the NRC entered the Monitoring Mode and staffed the Region I Incident Response Center. The residents assessed PPL's handling of the event from onsite locations and relayed significant plant information to the NRC's Incident Response Center. The ALERT was terminated at 11:35 pm.

b. Findings

Introduction: A Green self-revealing NCV associated with emergency planning standard 10 CFR 50.47(b)(4) was identified regarding inadequate indications for operators to determine if a threshold for an Alert Emergency Action Level (EAL) (OA7) declaration based on toxic gas concentrations immediately dangerous to life and health (IDLH) within a vital area had been met. Specifically, there were no meters (permanently installed or portable) available on site to measure Freon concentration, a toxic gas in high concentrations. This impacted the operator's ability to make an EAL declaration and operators had to rely on other indications such as personal ill effects from exposure. PPL entered this issue into its CAP as AR 1294109 and is evaluating the development of permanent corrective actions.

Description: On the morning of August 10, 2010, Susquehanna operators discovered a Freon leak from the Unit 1 RB chiller. The area was evacuated and approximately 30 minutes later, operators, maintenance technicians, and site safety personnel were sent back into the space to evaluate the leak, identify the source and isolate it, if possible. The leak was identified to be coming from the elbow of a 1" copper line to a filter and appeared to be unisolable. The lead maintenance technician, a qualified refrigerant handler, instructed all personnel to evacuate the area after he felt ill from the effects of Freon exposure. Personnel exited the space and reported the condition to the control room. The shift manager evaluated the entry criteria for OU7, "Release of Toxic or Flammable Gases Deemed Detrimental to NORMAL PLANT OPERATIONS," and OA7, "Release of Toxic or Flammable Gases within or Contiguous to a Plant VITAL AREA which Jeopardizes Operation of Safety Systems Required to Establish or Maintain Safe Shutdown." Alert OA7 was declared at 9:22 a.m. due to toxic gas concentrations in a vital area (Unit 1 RB) in concentrations greater than IDLH based upon the indication of personal ill effects from exposure. The shift manager declared the Alert and ordered the entire Unit 1 RB to be evacuated. Inspectors evaluated operator performance during this event and determined that an appropriate and timely declaration was made based upon the information available at the time.

The EAL threshold value for OA7 per PPL procedure EP-TP-001, Revision 3, "EAL Classification Levels," is the "report or detection of toxic gas within or contiguous to a plant VITAL AREA in concentrations that may result in an atmosphere IDLH." Additionally, the EAL states that an atmosphere that is IDLH may be determined by: direct measurement; other indication of personal ill effects from exposure; or a judgment that respirators must be work for entry to the area. The Freon IDLH is greater than 2000 ppm Freon.

During the August 10, 2010, Alert declaration, PPL identified that it did not have any installed or portable means of determining Freon concentration. PPL possessed Freon "sniffers" which could detect the presence of Freon but could not accurately measure concentration. Portable meters from another nuclear site and a vendor were identified and a wall mounted meter in the Freon Handling Building was identified and evaluated for being temporarily moved to the RB. Without the ability to remotely measure Freon concentrations or measure Freon concentrations using a portable meter, PPL could not evaluate the atmospheres during a known Freon leak and was forced to rely upon personnel showing exposure effects to declare this event. Furthermore, PPL did not have the Freon measurement capabilities to determine if respirators were required. Thus, PPL did not have two of three methods for determining IDLH available to them for a known hazard. PPL did not have instrumentation available to preclude entry into the space via remote monitoring or the means to minimize exposure times via portable meters.

PPL procedure EP-TP-001, Revision 2, "EAL Classification Levels" was implemented in May 2009. This revision included specific gases and concentrations that should be considered for an atmosphere that is IDLH, as described above. When the revision was implemented, PPL failed to properly consider whether the site possessed the necessary equipment necessary to measure all the gases considered per the emergency plan procedure. This change was carried into Revision 3, which was in effect at the time of this event.

Analysis: The failure to provide adequate indication for assessment of EAL entry criteria that could impact the declaration of an emergency was considered a performance deficiency that was within PPL's ability to foresee and prevent and is contrary to 10 CFR 50.54(q) and 10 CFR 50.47(b)(4). Traditional enforcement does not apply because there were no actual safety consequences, the violation was not willful, and it did not impact the NRC ability to regulate because the NRC inspectors were present and aware of the event. This finding is more than minor because it was associated with the Emergency Preparedness (EP) cornerstone attribute of Facilities and Equipment, and affected the cornerstone objective of ensuring that a licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. This finding was evaluated using IMC 0609, Appendix B, "Emergency Preparedness SDP," Sheet 1, "Failure to Comply." This finding is associated with a failure to meet or implement a regulatory requirement. The performance deficiency is not greater than Green because it did not result in the Risk-Significant Planning Standard Function being lost or degraded. Section 4.4 of IMC 0609, Appendix B, provides examples for use in assessing EP related findings. One example of a Green finding states, "The EAL classification process would not declare any Alert or Notification of Unusual Event that should be declared." Since the declaration of Alert OA7 based on toxic gas levels for Freon concentrations at or greater than IDLH within a vital area could have been missed or delayed, this finding was considered consistent with the example provided and was determined to be of very low safety significance (Green).

This finding is related to the cross-cutting area of Human Performance, Resources, because PPL did not ensure that equipment and other resources were available and adequate to assure safety. Specifically, PPL did not appropriately evaluate equipment necessary to effect a change to the emergency plan for an EAL classification related to toxic gasses in a vital area. PPL lacked adequate equipment to make an accurate EAL classification and had to rely on secondary means (personnel ill effects) for appropriately classifying a Freon leak in the Unit 1 RB that occurred on August 10, 2010. This was determined to be the most significant contributing factor to this issue. [H.2(d)]

Enforcement: 10 CFR 50.54(q) requires that the facility licensee follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b). 10 CFR 50.47(b)(4) requires, in part, that emergency response plans include a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters. The emergency classification and action level scheme is required to be used by the nuclear facility licensee, and state and local response plans rely on information provided by facility licensees for determinations of minimum initial offsite response measures. Contrary to the above, from May 2009 until August 2010, PPL did not have adequate means of indication or procedures to support an EAL classification based on toxic gas concentrations greater than IDLH within vital areas for Freon. PPL entered this issue into its CAP as AR 1294109 and is evaluating the development of permanent corrective actions. Because this issue is of very low safety significance and has been entered into PPL's CAP, it is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000387; 05000388/2010004-05, Inadequate Equipment to Measure Freon Concentration and Assess Threshold for an EAL Declaration)**

.4 (Closed) URI 05000387/2010-003-06, Predicted Plant Response to Large Transient With ICS

a. Inspection Scope

On May 14, 2010, during the performance of a condensate pump trip test, Unit 1 experienced a level transient and a manual reactor scram. Despite simulator predictions that the ICS would handle the recirculation pump runback and associated level transient, the ICS did not respond as expected. PPL conducted an RCA of the scram event at the end of the last inspection period. Inspectors reviewed PPL's RCA and interviewed staff involved with the design, implementation, simulation and operation of the ICS modification. PPL created and commenced implementation of corrective actions from their RCA. Immediate corrective actions included correcting simulator and in-plant MFWLC gains ("GAP control") and long-term corrective actions include revision of procedures governing modification testing.

b. Findings

A Green, self-revealing NCV of 10 CFR 55.46(c)(1), "Plant Referenced Simulators," was identified and is documented under section 4OA2.2 of this report. Based on inspector review and the resulting NCV, this URI is closed.

4OA5 Other Activities

.1 Operation of an ISFSI at Operating Plants (60855.1)

The inspectors verified by direct observation and independent evaluation that PPL had performed loading activities at the independent spent fuel storage installation (ISFSI) in a safe manner and in compliance with applicable procedures. This included observing the loading of one canister of spent fuel into the ISFSI on August 10, 2010. The inspectors verified by direct observation that radiation dose and contamination levels were within prescribed limits after a dry cask storage system container had been installed at the ISFSI.

.2 Institute of Nuclear Power Operations (INPO) Training Accreditation Report Review

a. Inspection Scope

On September 8 and 9, 2010, the inspectors reviewed the Institute of Nuclear Power Operations Training Accreditation Reports for the Licensed Operator Training Program report. The review included the previous reports, self-assessments, root cause analyses, and corrective actions developed by PPL following in response to previous reports. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On July 2, 2010, inspectors presented inspection results to Mr. J. Helsel and other members of his staff. PPL acknowledged the findings.

On July 30, 2010, inspectors presented inspection results to Mr. J. Helsel and other members of his staff. PPL acknowledged the findings. No proprietary information is contained in this report.

On August 27, 2010, inspectors presented inspection results to Mr. J. Helsel and other members of his staff. PPL acknowledged the findings.

On October 14, 2010, inspectors presented inspection results to Mr. J. Helsel and other members of his staff. PPL acknowledged the findings. No proprietary information is contained in this report.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by PPL and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy, for being dispositioned as non-cited violations:

- On August 4, 2010, a control rod was withdrawn from step 12, past the desired step 24, to step 36 during power ascension. PCOs identified the performance error prior to executing the next procedural step. This human performance error is a violation of TS 5.4.1, "Procedures," for failure to comply with a procedure for reactivity control. The finding is more than minor because it affected the Barrier Integrity cornerstone objective of reactivity control and affected the configuration control attribute, specifically control rod position and its potential impact on the fuel cladding. The violation was evaluated using IMC 0609 Attachment 4, Table 4a, and determined to be of very low safety significance (Green) because it was associated with a degraded fuel barrier. The issue was entered in PPL's CAP as CR 1289395.

- On August 19, 2010, PPL identified that Emergency Procedure, EO-000-031, "Station Power Restoration," Revision 17, was inadequate for restoration of emergency 4kV busses from a station blackout (SBO). Procedural steps to energize the busses once offsite power was available would not result in a closed breaker. A y-coil prohibited breaker closure without energizing the synchroscope and taking the breaker switch to close vice the as-written directions to take the switch to the open position, thus matching the in-field condition permitting automatic closure. This issue was determined to be a violation of 10 CFR 50.63, "Loss of All AC Power," for a degraded capability to recover from a station blackout. The finding is more than minor because it affected the Mitigating Systems cornerstone objective of system availability to respond to initiating events to prevent undesirable consequences and affected the procedure quality attribute, specifically an EOP. The violation was not greater than green because it did not represent a loss of safety function and was not related to a design or qualification deficiency or external event. The issue was entered in PPL's CAP as CR 1294270.

- On August 19, 2010, PPL identified that the risk profile erroneously indicated that the equipment out-of-service (EOOS) risk for both units was green. Upon further review, it was identified that the EOOS status should have been yellow for both units due to the "A" EDG ventilation supply fan being unavailable. PPL determined that, with the fan out-of-service (OOS) and the diesel running, room temperature could reach 120 degrees F in about three minutes. This issue was determined to be a violation of 10 CFR 50.65 (a)(4), for failure to ensure emergent work was properly modeled and evaluated for online plant risk. This finding is more than minor because it is similar to example 7.e. in NRC IMC 0612 Appendix E, "Examples of Minor Issues." This example states, in part, that failure to perform an adequate risk assessment when required by 10 CFR 50.65 (a)(4) is not minor if the overall elevated plant risk would put the plant into a higher licensee established risk category. This finding was evaluated using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process." In accordance with flow chart 1, the finding was determined to be Green since the risk deficit did not exceed the threshold for incremental core damage probability or incremental large early release probability. The issue was entered into PPL's CAP as CR 1294583.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Rausch, Site Vice President
T. Illiadis, General Manager Operations
D. Walsh, Assistant Operations Manager
R. Klinefelter, Assistant Operations Manager
A. Fitch, Site Training Manager
J. Petrilla, Supervisor Nuclear Regulatory Affairs
V. Schuman, Radiation Protection Manager
F. Hickey, Health Physicist
G. Glaser, System Engineer
R. Bogar, Senior Engineer
C. Lehman, Supervisor Plant Analysis
I. Missien, Senior Emergency Planned Coordinator
S. Davis, Manager Nuclear Emergency Planning
M. Adelizzi, Senior Engineer
J. Goodbred Jr, Manager Nuclear Operations
G. Mahalick, Senior Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened		
05000387/2010004-01	FIN TBD	Procedural Inadequacies Result in Reactor Scram and Loss of Normal Heat Sink (1R06)
Opened/Closed		
05000387/2010004-02	NCV	HPCI and RCIC CST Low-Level Suction Transfer Made Inoperable Due to Transfer of Water from Condenser Area to CST Berm (1R15.1)
05000387; 388/2010004-03	NCV	Failure to Accurately Model the Simulator for RCIC System Operation at Reduced Flow Rates in Automatic (1R15.2)
05000387/2010004-04	NCV	Failure to Accurately Model the Simulator for RPV Level Control Using the Integrated Control System (4OA2.2)

05000387; 388/2010004-05	NCV	Inadequate Equipment to Measure Freon Concentration and Assess Threshold for an EAL Declaration (4OA3.3)

Closed

05000387/2010003-06	URI	Predicted Plant Response to Large Transient With ICS (4OA3.4)

LIST OF DOCUMENTS REVIEWED
(Not Referenced in the Report)

Section 1R04: Equipment Alignment

Condition Reports (* NRC identified):
193889, 1300331, 1294200*, 1294215* , 1233313, 1236279, 1236734, 1289228, 1266449, 1297475*, 1063829, 1300460, 1290726, 1300202*, 1301624*

Procedures:
SO-250-001, Monthly RCIC Alignment Check, Revision 15
CL-250-0011, Unit 2 System Electric Checklist, Revision 8
CL-250-0012, Unit 2 RCIC System Mechanical Checklist, Revision 15
OP-202-001, 125 VDC System, Revision 16
CL-202-0011, 125 VDC System, Revision 1
OP-118-001, Instrument Air System, Revision 28
OP-118-002, Instrument Air System Infrequent Operations, Revision 3
OP-054-001, ESW System, Revision 29
CL-054-001, Common ESW System – Electrical, Revision 14
CL-054-0012, Common ESW System – Mechanical, Revision 19
CL-054-0015, Unit 2 ESW System – Electrical, Revision 3
CL-054-0016, Unit 2 ESW System – Mechanical, revision 25
CL-054-0011, Common ESW System – Electrical, Revision 14

Drawings:
E-106255, RCIC P&ID, Revision 30
E-107160, Sheet 2, 125 VDC System One Line Diagram, Revision 27
E-106230, Sheet 1, Unit 1 P&ID Compressed Air System (Instrument Air), Revision 52
E-106216, Sheet 1, ESW Common P&ID, Revision 48
E-106216, Sheet 2, ESW Unit 2 "B" LOOP P&ID, Revision 7
E-106216, Sheet 3, ESW Unit 1 "B" LOOP P&ID, Revision 22
E-106216, Sheet 4, ESW Common P&ID, Revision 2

Other:

TM-OP-018-ST, Instrument Air, Revision 7
TM-OP-054-ST, ESW, Revision 8

Section 1R05: Fire Protection

Condition Reports (*NRC identified):
1278916, 1286081, 1277366, 1276868, 1294259*, 1299695*

Action Requests (*NRC Identified):
1286135*

Procedures:
FP-113-108, Fire Zone 1-2A/2-3C Pre-Fire Plan, Revision 4
FP-113-109, Fire Zone 1-2B/1-2D Pre-Fire Plan, Revision 4
FP-013-192, DG Bay "B", Fire Zone 0-41B, Elevations 677', 660', and 710', Revision 4
FP-213-248, Containment Access Area (11-401) Decontamination Stations (11-404, 405) Fire Zones 2-4A-N, 2-4A-W, 2-4A-S, Elevation 719'-1", Revision 5
FP-013-200, Pre-Fire Plan ESSW Pump House LOOP "A", Pump Room (E-1), Revision 4
FP-013-201, Pre-Fire Plan ESSW Pump House LOOP "B", Pump Room (E-2), Revision 4
FP-113-130, H&V Filter Rooms (I-700), (I-702) Access Area (1-704), Fan Room (I-703), Fire Zone 1-7A Elevation 799'-1", Revision 4

Drawings:
E-205950, Sheet 1, Unit 1 RB Elevation 670' Fire Zone Plan, Revision 5
E-205950, Sheet 2, Unit 1 RB Elevation 670' Fire Doors and Dampers, Revision 6

Work Order:

Other:
Fire Protection Review Report, Revision 18
NTP-QA-53.1, Susquehanna Fire Brigade Training Program, Revision 19
NSEI-AD-145, SFPE Responsibilities in Fire Brigade Program, Revision 7

Section 1R06: Flood Protection Measures

Condition Reports (* NRC identified):
845713

Procedures:
ON-142-001, CW System Leak, Revision 17

Drawings:
M-161, Sheet 3, Liquid Radwaste Collection, Revision 16
A-11, Sheet 1, General Floor Plans, Elevation 646'-0", 648"-0", and 656'-0", Revision 24

Other:
IN 2007-01, Recent Operating Experience Concerning Hydrostatic Barriers
EC-012-2864, Evaluation of Condenser Area Bay Doors, Revision 0

Section 1R11: Licensed Operator Requalification Program

Procedures:
EO-100-113, "Level/Power Control," Revision 8
EO-100-102, "RPV Control," Revision 8
EP-TP-1, "Emergency Classifications Level Manual," Revision 3
EO-100-112, "Rapid Depressurization," Revision 7

Section 1R12: Maintenance Effectiveness

Condition Reports:
1028121, 1300167*, 1300262*, 320683, 1251755, 855957, 1195381, 1243436, 1259153, 868163, 868164, 1268450, 1299699, 1296015, 1295849, 1295903, 1296646, 872056, 824522, 810442, 809503, 877828,

Procedures:
NDAP-QA-0412, Leakage Rate Test Program, Revision 12

Work Orders:
1181786

Other:
LER 05000388/2007-001-00
LER 05000387/2001-003-00
10 CFR 50 Appendix J
Regulatory Guide 1.163, Performance-based Containment Leak Test Program, September, 1995
Unit 1, 16 RIO Post-Outage Containment Leakage Testing Report
PLA-6392, SSES Amendment Request: Technical Specification Change to Technical Specification 3.6.1.3 to Increase the Maximum Allowable Secondary Containment Bypass Leakage Limit, July 31, 2008
System Health Report, 2010 2nd Quarter
Maintenance Rule Basis Document – System 54

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Condition Report:
1295164, 1302058, 1302070, 1302445*, 1305655, 1306005, 1305843, 1306078, 1305993, 1306214, 1307411, 1306092, 1306222

Procedures:
NDAP-QA-0340, Protected Equipment Program, Revision 6
NDAP-QA-1902, Maintenance Rule Risk Assessment and Management Program, Revision 2

Section 1R15: Operability Evaluations

Condition Reports and Action Requests (* NRC-identified):
636249, 863257, 1276460, 1275707, 1282142, 1282850, 1282633, 1282691, 1289723 1283258, 1284233*, 1282143, 1285438, 1282503*, 1282520*, 1282629*, 1287919, 1287062, 1302995, 1318476*

Calculations:
EC-051-0004, Core Spray TS Test Pressure, Revision 6

Procedures:
SO-151-A05, Core Spray Comprehensive Flow Verification Division I, Revision 4
SO-151-A02, Quarterly Core Spray Flow Verification Division I, June 30, 2010, Revision 15
FSAR 6.3.2.2.3
SO-150-004, RCIC Valve Exercising, Revision 27

SO-150-002, Quarterly Flow Suveillance, Revision 41
OP-169-004, Liquid Radwaste Collection, Revision 17
ON-142-001, CW System Leak, Revision 17
ON-169-001, Flooding in the Turbine Building, Revision 3
AR-214-001, HPCI Alarm Response Procedure, Revision 22

Drawings:
M-152, Core Spray, Revision 38

Other:
IN 2009-09, Improper Flow Controller Settings Renders Injection Systems Inoperable and Surveillance Did Not Identify
EPRI Terry Turbine Maintenance Guide, RCIC Application, dated November 26, 2002
EC-051-004, Core Spray Technical Specification Test Pressure, Revision 6
Regulation Guide 1.9, Application and Testing of Safety-Related DGs in Nuclear Power Plants, Revision 4
Inspection Report 05000482/2008004
TM-OP-052-ST, HPCI Student Text, Revision 2
FSAR 7.3.1.1a.1.3, 6.3.2.2.3
TS 3.8.1
FSAR 8.3
DBD004, HPCI Design Basis Document, Revision 5
E-105956, HPCI P&ID, Revision 26

Engineering Work Request:
1285503,
M00696, "CST Berm Area Penetration Leakage

Operability Followup Request:
1202850, 1282633, 1282691

Section 1R18: Permanent Plant Modifications

Condition Reports (* NRC identified):
1282489, 1285595, 1286057, 1302889*, 1294270, 1137664, 1137184, 654872, 1311423

Procedures:
TP-142-006, Pumping water to Cooling Tower Blowdown Line, Revision 0
EO-000-031, Station Power Restoration, Revision 17
5059-01-1412, Revision 0

Drawings:
FF-62098, Schematic Diagram 50DHP – VR 250V, 1200A Class IE VR Series Breaker, Sheet 3, Revision 0
E-103, 4.16kV Bus 1A Auxiliary Relay Control, Slot 2, Revision 28
E-103, 4.16kV Bus 1A Incoming Feeder Breaker from ESS Trans 101, Sheet 1, Revision 29

Other:
EC-486214, 4 kV Breaker Replacement Project, Revision 1
50.59 SD00919, Discharge Radioactive Water From Tanker Directly to Cooling Tower Blowdown Line, Revision 0

FSAR 11.5, 9.4.2, 15.7.4
TM-OP-079E-ST, Process Radiation Monitoring, Revision 3
LDCN 4823 and 4824, Refuel Floor Radiation Monitoring Ventilation Support Requirements

Section 1R19: Post-Maintenance Testing

Condition Reports (*NRC-identified):
1252833, 1252834, 1252004, 1251995, 1263386, 1262646, 1266374, 1268944*, 1268969*, 1287919, 1286704, 1286720, 1286694, 1287062, 1286719, 1280638, 1281379, 1281522, 1281850, 1280576, 1279833, 1284470, 1284837

Action Request:
580264

Procedures:
SI-152-308, Quarterly Calibration of CST Low Level Channels LSLL-E41- 1N002 and LSLL-E41-1N003, August 1, 2010, Revision 11
SI-150-315, Quarterly Calibration of CST Low Level Channels LSL-E51- 1N035 A & E, August 1, 2010, Revision 10
SE-255-001, 24 Month Division II CRD Pump 2P132B DC Control Automatic Transfer Logic, August 12, 2010, Revision 6
MT-GE-048, Cutler Hammer Type DHP-VR 4.16 kV Circuit Breaker and Switchgear Inspection and Maintenance, Revision 14
SO-252-002, Quarterly HPCI Flow Verification, July 15, 2010, Revision 45
SO-024-001, "D" DG. Monthly Operability Test, July 8, 2010, Revision 7
SO-251-A02, "92 DY Core Spray Flow Verification "A" LOOP," August 18, 2010, Revision 15

Work Orders:
1269857, 1269861, 1284370, 1283967, 1168836, 1222293, 975437

Other:
PLA-4677, R. G. Byram, "Relay to a Notice or Violation (50-387/97-03-02)"

Section 1R20: Refueling and Other Outage Activities

Condition Reports (* NRC identified):
1287225, 1289358, 1289361

Procedures:
GO-100-006, Cold Shutdown, Defueled and Refueling, Revision 43
GO-100-005, Plant Shutdown to Hot/Cold Shutdown, Revision 48
GO-100-010, ECCS/Decay Heat Removal in Mode 4, 5, or Defueled, Revision 17
GO-100-002, Plant Startup, Heatup, and Power Operation, Revision 66

Other:
Operations, Maintenance and Fire Brigade work Schedules for July 12 – 26, 2010
TRO 3.7.9
Unit 1, Cycle 17, Control Rod Sequence A1 1450

Section 1R22: Surveillance Testing

Condition Reports (* NRC identified):
1297015, 1296927, 1296405

Procedures:
SO-150-005, 24 Month RCIC Flow Verification, August 3, 2010, Revision 16
SO-252-006, HPCI Comprehensive Flow Verification, Revision 10
SO-030-B01, Monthly CREOAS "B" Operability Test, Revision 0
SO-030-B03, Quarterly CSCS Flow Verification LOOP "B", Revision 15
SE-234-003, 24 Month Division I Reactor Building Chiller Compressor 2K206A, Revision 3

Section 1EP6 Drill Evaluation

Condition Reports:
1296115, 1296125, 1296145, 1296160, 1296136, 1296315, 1188839, 1298964, 1300530, 1300535*

Procedures:
EP-TP-001, EAL Classification Levels, Revision 3

Other:
NEI 99-02, Regulatory Assessment Performance Indicator Guide List, Revision 6
EP-AD-005, SSES Drill and Exercise Performance, Revision 15

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Condition Reports:
1243791; 1244946; 1247024; 1249276; 1251178; 1251330; 1251628; 1252708; 1260056; 1263463, 1262901, 1267016, 1290224, 1291992

Section 2RS2: Occupational ALARA Planning and Controls

Condition Reports:
1246626; 1257667

Other:
Unit 1 16th Refuel and Inspection Outage Radiological Performance Report
Station ALARA Committee Agenda, July 1, 2010

Section 2RS3: In-Plant Airborne Radioactivity Control and Mitigation

Procedures:
HP-TP-240, Use and Operation of Portable Ventilation Units, Revision 18
HP-TP-500, Directions for Completing the weekly Vacuum Program, Attachment L, Revision 38

Section 2RS5: Radiation Monitoring Instrumentation

Condition Reports:
1243427; 1243633; 1245033; 1246184; 1246188; 1246361; 1258867

Other:
Health Physics Technical Basis 00-022
WBC System Performance Verification, March 30, 2010

Section 4OA1: Performance Indicator Verification

Condition Reports (* NRC identified):
1294189*, 1197631, 1238431, 1293601, 1246429, 1236279, 1177506, 1246136, 1253630, 1304133, 1239635

Other:
EDG System Engineer Unavailability Spreadsheet
Operator Logs from November 1, 2009 to July 31, 2010
NEI-99-02, Regulatory Assessment Performance Indicator Guidelines, Revision 5
PL-NF-06-002, Mitigating System Performance Index, Revision 5
MSPI Derivation Reports for UAI and URI of Units 1 and 2 Emergency AC Power System ending July 2010
MSPI Derivation Reports for UAI and URI of Units 1 and 2 Cooling Water System ending July, 2010

Section 4OA2: Identification and Resolution of Problems

Condition Reports (* NRC identified):
1302276, 1299656*, 1299642*, 1303423*, 1303422*, 1282662, 1243388, 1097100, 1132862, 1222656, 1109279, 1149714, 1220771, 1222895, 1246769, 1248262, 1257781, 1281156, 1282140, 1282503, 1285070, 1285072, 1286930, 1284526*, 1287462*, 1287465*, 1287467*

Procedures:
NDAP-QA-0300, Conduct of Operations, Revision 26
GO-100-002, "Plant Startup, Heatup and Power Operation", Revision 66
ON-145-001, "RPV Level Control System Malfunction", Revision 27
ON-245-001, "RPV Level Control System Malfunction", Revision 27
ON-131-003, "ICS Component Failure(s)", Revision 0
OP-AD-001, "Operations Standards for System and Equipment Operation", Revision 44
OP-AD-002, "Standards for Shift Operations", Revision 33
OP-AD-055, "Operations Procedure Program", Revision 12
OP-145-001, "RFP and RFP Lube Oil System", Revision 58
OP-150-001, "RCIC System," Revision 33
TP-145-028, "SAT – ICS – Initial Operation of FWLC, Recirculation Speed Control, and RFPT Speed Control", Revision 0
TP-145-029, "Feedwater Master Level Controller (MWLC) Tuneup", Revision 3
TP-145-030, "SAT-ICS – Initial Calibration of FWLC, Rx Recirculation Speed Control and RFPT Speed Control", Revision 3
TP-145-031, "ICS Startup and Tuneup in Condition 1 and 2 Less Than 40% RTP", Revision 1
TP-145-033, "Site Acceptance Test – integrated Control System (ICS) of FWLC, RRP Speed Control, RFP Speed Control (Pre-Outage), Revision 2
TP-145-034, "ICS Startup and Flow Control Tuning", Revision 0
TP-150-005, "RCIC Minimum Speed vs. Oil Pressure", Revision 0
TP-150-006, "RCIC Pump Performance Test", Revision 0

Work Orders:
1110756

Other:
OI-AD-096, Operator Work-Arounds/Challenges, Revision 6
OWA List as of August 29, 2010
Control Room Deficiencies List as of August 29, 2010 and Operator Challenges List
Trend Code Search Results for OWAs from June 30, 2009 – September 30, 2010
ANSI/ANS 3.5-1985, "Nuclear Power Plant Simulators for use in Operator Training"
OP002-09-01-15
OP002-09-02-04
OP002-09-06-06
OP002-10-01-03
OP002-10-01-04
OP002-10-03-02
OP002-10-03-03
OP002-10-04-04
OP002-10-04-05
OP002-10-04-09
OP002-10-05-01A
OP002-10-05-04
OP002-10-05-05
LOR Cycle 09-04 Job Sheet 1
LOR Cycle 09-05 Job Sheet 1
LOR Cycle 09-05 Rx Recirculation HMI
LOR Cycle 09-06 HMI #1 Worksheet
LOR Cycle 09-06 HMI #2 Worksheet
LOR Cycle 09-06 HMI #3 Worksheet
LOR Cycle 09-06 HMI #4 Worksheet
LOR Cycle 10-02 HMI #1 Worksheet
LOR Cycle 10-03 HMI #1 Worksheet
Annual Licensed Operator Requalification Schedule 2009
Annual Licensed Operator Requalification Schedule 2010

Section 4OA3: Event Followup

Condition Reports:
1284526*, 1284760*, 1284855*, 1289140, 1289395, 1289823, 1028121, 1300167*, 1300262*, 320683, 1251755, 855957, 1195381, 1243436, 1259153, 868163, 868164, 1268450

Procedures:
OP-145-001, RFP and RFP Lube Oil System, Revision 59
ON-155-001, Control Rod Problems, Revision 34
NDAP-QA-0412, Leakage Rate Test Program, Revision 12

Work Orders:
1181786

Other:
PPL Operations Department Stand Down Letter
Unit 1 Operator Logs for August 3 through 5, 2010

Unit 1 Core Performance Log – Predict Calculation for Rod Misposition
Unit 1 Cycle 17 Startup Control Rod Sequence A1

Section 4OA5: Other Activities

Other:
Area Surveys – ISFSI Facility dated: August 19, 2010; July 23, 2010; July 8, 2010; June 18, 2010; May 22, 2010; April 24, 2010; January 22, 2010; October 22, 2009; and July 25, 2009

Section 4OA7: Licensee-Identified Violations

Condition Reports:
1294583, 1295164, 1297347

LIST OF ACRONYMS

AC	Alternating Current
ADAMS	Agencywide Document and Access Management System
ALARA	As Low As Is Reasonably Achievable
ANS	Alert and Notification System
AR	Action Report
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CAP	Corrective Action Program
CCDP	Conditional Core Damage Probability
CDP	Core Damage Probability
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
CREOAS	Control Room Emergency Outside Air Supply
CST	Condensate Storage Tank
CW	Circulating Water
DEP	Drill and Exercise Performance
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOOS	Equipment Out-of-Service
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
EPD	Electronic Personnel Dosimeter
EPU	Extended Power Uprate
ER	Engineering Request
ERO	Emergency Response Organization
ESS	Engineering Safeguard System
ESW	Emergency Service Water
ESSW	Engineering Safeguard Service Water
EWR	Engineering Work Request

FEMA	Federal Emergency Management Agency
FIN	Finding
FSAR	[SSES] Final Safety Analysis Report
GE	General Electric
GEH	GE - Hitachi
GL	Generic Letter
HPCI	High Pressure Coolant Injection
HRA	High Radiation Area
HV	High Voltage
HVAC	Heating, Ventilation and Air-Conditioning
IA	Instrument Air
ICS	Integrated Control System
IEF	Initiating Event Frequency
IEL	Initiating Event Likelihood
I&C	Instrumentation and Controls
IDLH	Immediately Dangerous to Life and Health
IEEE	Institute of Electrical and Electronics Engineers
IN	Information Notice
IL	Instruction Leaflet
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	NRC Inspection Report
ISI	Inservice Inspection
IST	Inservice Testing
IVVI	In-Vessel Visual Inspection
IVVI/IST	In Vessel Visual Inspection/Inservice Testing
JP	Jet Pump
kV	Kilovolts
LCO	Limiting Condition for Operation
LDE	Lens Dose Equivalent
LEFM	Leading Edge Flow Meter
LER	Licensee Event Report
LERF	large Early Relief Frequency
LLD	Lower Limits of Detection
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LSFT	Logic System Functional Test
LTC	Load Tap Changer
MFWLC	Master Feedwater Level Controller
M&TE	Measuring and Test Equipment
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Testing
NCV	Non-Cited Violation
NDAP	Nuclear Department Administrative Procedure
NDE	Non-Destructive Examination
NDT	Non-Destructive Test
NEI	Nuclear Energy Institute
NERO	Nuclear Emergency Response Organization
NRA	Nuclear Regulatory Affairs
NRC	Nuclear Regulatory Commission
OA	Other Activities

OE	Operating Experience
OFR	Operability Followup Request
ON	Off-Normal
OOS	Out-of-Service
PARS	Publicly Available Records
PCO	Plant Control Operator
PEMA	Pennsylvania Emergency Management Agency
PF	Power Factor
PI	[NRC] Performance Indicator
PI&R	Problem Identification and Resolution
PIM	Plant Issues Matrix
PMT	Post-Maintenance Test
PPL	PPL Susquehanna, LLC
PRV	Pressure Relief Valve
PS	Planning Standard
PT	Penetrant Test
QA	Quality Assurance
RB	Reactor Building
RCA	Radiologically Controlled Area
RCA	Root Cause Analysis
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluents Technical Specifications
RFO	Refuel Outage
RFPT	Reactor Feedpump Turbine
RFU	Request for Followup
RG	[NRC] Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RMS	Radiation Monitoring System
ROP	Reactor Oversight Process
RPM	Radiation Protection Manager
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power
RVWL	Reactor Vessel Water Level
RWMU	River Water Make-Up
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SCWE	Safety Conscious Work Environment
SDE	Skin Dose Equivalent
SDHR	Supplemental Decay Heat Removal
SDP	Significance Determination Process
SE	Safety Evaluation
SOMS	Shift Operations Management System
SPAR	Standard Plant Analysis Risk
SRA	Senior Risk Analyst
SRV	Safety Relief Valve
SSC	Structures, Systems and Components

SSES	Susquehanna Steam Electric Station
SW	Service Water
TASA	Tapchanger Activity Signature Analysis
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescence Dosimeter
TRM	Technical Requirements Manual
TS	Technical Specifications
T20	T20 Startup Transformer
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Test
VHRA	Very High Radiation Areas
VT	Visual Examination
WBC	Whole Body Counter
WO	Work Order
WPS	Weld Procedure Specification